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Nuclear Criticality Safety Experiments, Calculations, and Analyses—1958 to 1982

Compilation of Papers from the Transactions
of the American Nuclear Society

Volume 2: Summaries

Brian L. Koponen and Viktor E. Hampel

October 21, 1982

Lawrence
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UCRL--53369-Vol.2

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Gentlemen:

Enclosed is a compilation of papers on nuclear criticality safety experiments, calculations, and analyses originally published in the Transactions of the American Nuclear Society. This compilation in UCRL-53369 is one of a series of computer-generated reports released from the Nuclear Criticality Information System (NCIS) at the Lawrence Livermore National Laboratory (LLNL).

The Technology Information System (TIS) at LLNL operates the host computer for NCIS. The NCIS project is funded by the U. S. Department of Energy's Nuclear Facility Safety Division, Office of Nuclear Safety.

The information in these publications also can be searched interactively by authorized users from remote terminals by telephone dial-up, toll-free WATS lines, over the Advanced Research Projects Agency (ARPA) Computer Network, and over the International TYMNET.

We invite your comments about this publication and those that you may wish to see in the future. Your professional contributions and collaboration are welcome. Inquiries about NCIS should be directed to:

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Nuclear Criticality Safety Experiments, Calculations, and Analyses—1958 to 1982

**Compilation of Papers from the Transactions
of the American Nuclear Society**

Volume 2: Summaries

Brian L. Koponen and Viktor E. Hampel

Manuscript Date: October 21, 1982

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Nuclear Criticality Safety Experiments, Calculations, and Analyses—1958 to 1982

Abstract

This compilation contains 688 complete summaries of papers on nuclear criticality safety as presented at meetings of the American Nuclear Society (ANS). The selected papers contain criticality parameters for fissile materials derived from experiments and calculations, as well as criticality safety analyses for fissile material processing, transport, and storage.

The compilation was developed as a component of the Nuclear Criticality Information System (NCIS) now under development at the Lawrence Livermore National Laboratory.

The compilation is presented in two volumes: **Volume 1** contains a directory to the ANS Transaction volume and page number where each summary was originally published, the author concordance, and the subject concordance derived from the keyphrases in titles. **Volume 2** contains—in chronological order—the full-text summaries, reproduced here by permission of the American Nuclear Society from their Transactions, volumes 1-41.

Introduction

The summaries on nuclear criticality safety experiments, calculations, and analyses presented herein were originally published in the proceedings of the semiannual meetings of the American Nuclear Society. They are reproduced here by permission of the ANS as a quick reference for those concerned with the nuclear criticality safety of fissile materials.

This compilation is one of a series of computer-generated publications released from the Nuclear Criticality Information System (NCIS)

at the Lawrence Livermore National Laboratory (LLNL).¹⁻⁶ A previous publication contained references to 1067 reports involving nuclear criticality experiments.⁷ The information in both publications can be searched interactively from remote terminals by telephone dial-up, toll-free WATS lines, over the Advanced Research Projects Agency (ARPA) computer network, and over the international TYMNET. The Technology Information System (TIS) at LLNL operates the host computer for NCIS.⁸⁻¹²

Contents of the Compilation

Papers in this compilation were selected from the view of a criticality safety specialist who needs accurate information and data to determine safe limits for processing, transporting, and storing fissile materials. Papers related directly to power

and research reactors have been omitted, as have those dealing with the determination of neutron reaction rates (i.e., cross sections and other physics parameters). Papers pertaining to calculational methods and results of criticality safety analyses

Table of ANS Transactions selected for compilation.

Volume	Year	Papers selected
1	1958	10
2	1959	13
3	1960	19
4	1961	39
5	1962	23
6	1963	9
7	1964	11
8	1965	16
9	1966	14
10	1967	19
11	1968	29
12	1969	40
13	1970	41
14	1971	41
15	1972	25
16	1973	23
17	1973	18
18	1974	10
19	1974	19
20	1975	2
21	1975	6
22	1975	13
23	1976	14
24	1976	9
25	1976 Special meeting	0
26	1977	6
27	1977	31
28	1978	22
29	1978 Special meeting	0
30	1978	13
31	1979 Special meeting	13
32	1979	19
33	1979	17
34	1980	9
35	1980	20
36	1981 Special meeting	0
37	1981 Special meeting	0
38	1981	10
39	1981	43
40	1982 Special meeting	0
41	1982	25
42	1982	15 ^a

^a These papers had not been published at the time of this publication and are not included in the compilation.

are included. Some of the papers deal with criticality safety training, safety standards, quality assurance, and related administrative aspects of the national criticality safety programs.

The 688 papers describing work carried out by the international nuclear criticality safety community span more than two decades of work with fissile materials and have contributed to a safety record unmatched by any other industry. Although a total of 819 authors contributed to this body of knowledge, 32 specialists are credited with having published 347, or half, of all the papers, indicating that experimental work centers around a relatively small number of facilities. Some 550 authors have published only once. These statistics underscore the importance of organizing and disseminating nuclear criticality information to those responsible for the safety of others.

As calculational methods have improved in recent years, more than half of the critical experiment facilities have been closed. It is gratifying to note, however, that the number of nuclear safety publications has continued to rise (see figure).

Accurately measured and well reported criticality experiments represent a timeless and valuable reference. However, readers who want to establish the publication date of a particular summary can refer to the following table of ANS Transactions, which shows the year and number of publications selected for this compilation.

Other NCIS publications scheduled for release include a general bibliography to the nuclear criticality literature and an update to the previous

compilation on simple critical experiments. We also hope to release numeric data describing nuclear criticality safety experiments that may serve as benchmarks for calculations.

Corrections, comments, or additions to this compilation are welcomed. Address all communications to:

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How to Use the Compilation

This compilation is presented in two volumes to allow their convenient and independent use:

Volume 1, the directory, is organized in chronological and alphabetical order:

- Index to volume and page number of ANS Transactions.
- Author concordance.
- Author index.
- Subject concordance.
- Subject index.

The author and subject concordances list, respectively, in chronologically ascending order, the relevant publications associated with a particular author or a keyphrase taken from the titles to the publications. The volume and page numbers of

ANS Transactions shown in the concordances lead to the full-text papers reproduced in Volume 2.

The author and subject indexes are concise listings of all authors and keyphrases and refer to the page numbers of Volume 1 where the expanded concordance for the relevant publications is cited.

Volume 2 contains the full-text reprint of the selected nuclear criticality safety publications taken directly from the ANS Transactions. The top of each page bears the original volume and page number for each paper. The name of the session is indicated only before the first paper in the session. When more than one paper is reproduced on a page, more than one range of page numbers appears at the top of the page.

Creating the Compilation by Computer

Since 1968, we have been collecting contributions for a library of criticality safety documents at LLNL. This collection has been augmented by LLNL's Technical Information Department (TID) through selective searching and retrieval of pertinent reports from the literature. More recently, we have used the large, computer-based information system of the National Technical Information Service (NTIS) of the U.S. Department of Energy (DOE/RECON) to obtain citations to the literature pertinent to nuclear criticality safety. This search and retrieval pertains especially to RECON's Nuclear Science Abstracts (NSA) and to its Energy Data Base (DOE/EDB).

Originally, we used the MASTER CONTROL database management system to create the databases, to verify the text of the citations, and to search interactively.^{13,14} The system also permits direct correlation with the corresponding numeric data of criticality experiments and printing in a variety of different formats. Interactive search and retrieval with MASTER CONTROL provides LLNL staff with immediate and up-to-date insights about criticality experiments, or shows the results of relevant computer calculations. During the past 12 years, we developed a sophisticated set of postprocessing routines that take computer output generated by MASTER CONTROL and

format it, with a variety of fonts and type options, into camera-ready copy for printing.^{15,16} We used the high-resolution hardcopy cameras of the FR80 output devices, linked directly with LLNL computers,* to create earlier compilations of criticality safety papers. The use of MASTER CONTROL and OCTOPUS is limited to LLNL.

To make this valuable resource available to others, descriptive and numeric information and

data files are now being transferred from the OCTOPUS computer network to the TIS for interactive access and use nationwide by criticality safety personnel at other national laboratories and by special arrangement at universities and in industry.

*OCTOPUS network of four CRAYs and four CDC-7600s.

Acknowledgments

We gratefully acknowledge the assistance of the many experimenters and nuclear criticality safety personnel who encouraged us to proceed with this work. Of special note are DOE's Nuclear Facilities Safety Division, Office of Nuclear Safety, under whose auspices and sponsorship this work was carried out, and O. Clinton Kolar, head of the Criticality Safety Office at LLNL. Dr. Kolar has established the long-range planning and support required to make the unique resources of LLNL available to the criticality safety community at large. The creation and expansion of the online interactive NCIS, in particular, is due in part to his initiative.

We would also like to thank Richard K. Hunt of TID, who carried out the required support functions, Ginger Affolter and Carol Ann Foret of Staffbuilders, who contributed their expertise to the preparation of the camera-ready copies, and June A. Canada and Barry R. Smith of TID, who provided editorial guidance and had overall responsibility for the publication.

Our particular thanks go to Michael Diekman of the American Nuclear Society, who allowed us to reprint the selected papers on nuclear criticality safety as a concise and timely reference.

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Critical Assembly Experiments

Critical-Mass Measurements on Beryllium-Reflected, Enriched-Uranium, Spherical Systems, by O. Kolar and H. R. Ralston (UCRL-L). A series of critical mass determinations has been made for spherical beryllium reflected systems. Cores were enriched (93.17% U-235) uranium spheres ranging in mass from 10.6 to 32.6 kg. The corresponding reflector thicknesses were 20.3 to 2.2 cm. All results were normalized to a core density of 18.6 g/cc. Some Los Alamos data are included for completeness and have been similarly normalized. These two sets of experiments were found to be in good agreement. The experiments and results are discussed, together with the fit to the experimental data obtained with the spherical S_4 , 5-group code in use at UCRL.

The experimental data are as follows:

<u>Percent U-235</u>	<u>Oralloy density, g/cc</u>	<u>Critical mass, kg</u>	<u>Beryllium critical thickness, cm</u>
93.17	18.49	10.6†	20.27 ± 0.41
93.5*	18.8	15.4†	10.16
93.17	18.58	16.187	9.27 ± 0.18
93.17	18.59	21.700	5.44 ± 0.07
93.5*	18.8	22.8†	5.08
93.17	18.59	27.911	3.264 ± 0.033
93.5*	18.8	31.9†	2.54
93.17	18.59	32.574	2.22 ± 0.066
93.8*	18.8	52.4†	0.000

*G. A. Graves and H. C. Paxton, "Critical Masses of Oralloy Assemblies," *Nucleonics* **15**, 90-92 (June, 1957).

†Corrected to a U-235 concentration of 93.17% and an oralloy density of 18.6 g/cc.

Interaction of Subcritical Components,* by H. K. Clark (DuP-SR). A simple, generally conservative, and reasonably accurate method is developed for computing the interaction between fissile units or between such a unit and a reflector. For a given unit the albedo provided by its surroundings is expressed in terms of the fractions, ρ , of the neutrons emitted by the other units, or by reflectors, that reach the unit in question. Simplifying assumptions are made so that these fractions may be determined by purely geometrical calculations independent of the flux distributions. A conservative expression is used for the albedo which relates it to the reflector saving provided by the surroundings and to an experimentally determined extrapolation distance obtained by equating the material and geometric bucklings of unreflected isolated critical units.

An expression for ρ is derived for rectangular slabs at right angles to each other. From this expression values of ρ for parallel rectangular slabs with identical faces are obtained. These results are used for cylinders as well as for slabs by approximating circular cylinders by square cylinders with the same base area and the same axis-to-axis separation.

Equations are derived for multicomponent systems where the components need not be identical or equally spaced. Special precautions are necessary when the interacting faces are of unequal area, and a procedure is outlined for such situations. A procedure is outlined for allowing for the fraction of neutrons that is intercepted by intervening units.

An expression for ρ for the self-interaction in an annulus is derived and evaluations of the expression are plotted. Expressions for ρ for the case of a disc or of an infinite slab of finite height parallel to an infinite plane reflector are derived and evaluated. These results may be applied to any slab by taking whichever ρ is smaller, that for a disc of the same area or that for an infinite slab of the same height. Expressions are also derived for ρ for the situation in which a fissile unit is completely surrounded by a reflector.

Comparisons with experiment are made on the basis of calculated and experimental values of the reflector saving provided by the surroundings. Agreement is fairly good and generally conservative. The cases where the calculations are non-conservative by appreciable amounts are large diameter cylinders and groups of cylinders at small spacings. In the latter case, the non-conservatism is expected as the result of the assumption of a symmetrical flux distribution in each cylinder.

Experiments in Thoria-Urania-D₂O Criticals, by J. W. Armstrong, J. A. Beidelman, S. G. Kaufmann, and K. E. Plumlee (ANL). Initial results obtained in an experimental investigation of the nuclear properties of a thoria-uranium-D₂O system were presented at the second winter meeting of the ANS. Recent experiments have explored the effect of deviations from an ideal $\frac{1}{2}$ -in. triangular lattice spacing of the fuel pins on the buckling and migration area. Measurements of the disadvantage factor, reflector savings and photoneutron intensity are also reported.

The first system studied was a $\frac{1}{2}$ -in. triangular lattice of one-quarter inch diameter fuel having a 25 to 1 atom ratio of thorium to U-235, loaded into element holders which fitted among a pattern of control rod guide channels duplicating the EBWR structure. Subsequently the fuel rods in each element were clustered at its center with a $\frac{1}{2}$ -in. spacing, and finally the entire core structure was replaced by three layers of perforated sheet which allowed the assembly of a $\frac{1}{2}$ -in. lattice free of interruptions due to the control rod guide channels and the structure of the fuel rod holders. The bucklings for these three geometries were 1.89, 1.78 and $1.99 \times 10^{-3} \text{ cm}^{-2}$, respectively. The modified migration area, $\tau + (L^2/1 + L^2B^2)$, was determined by the differential water worth method to be 188 cm².

Calculation of the above bucklings was based on the observed correspondence between the changes in height and radius to maintain criticality and a knowledge of the axial reflector savings. The reflector savings was determined by fitting a cosine function to the central portion of the axial thermal flux distribution as measured by both gold wires and a miniature fission counter. The above values are based on $\lambda_{\infty} = 9.0 \text{ in.}$, but more recent measurements indicate a somewhat lower value.

Detailed flux distribution measurements throughout a representative cell have yielded a value of 1.10 ± 0.02 for the disadvantage factor for a single fuel pin in a $\frac{1}{2}$ -in. triangular lattice. Similar measurements in the clustered geometry gave 1.4.

An investigation was conducted into the validity of the inhour curve using only delayed neutrons from fission. The contribution to the effective delayed neutron fraction from photoneutrons generated in the heavy water by induced or fission gamma activity was determined experimentally by observing the flux decay after shutdown and matching theoretical expressions to the observed decay curves. A ratio of 0.10 for delayed neutrons from photo disintegration to those from fission permitted a good fit to the experimental data. For infinite dilution of U-235 in D₂O the reported value is 0.159.

Critical Assemblies and Research Reactors

Dilute Fast Critical Assemblies—Experimental Results, by J. K. Long, F. S. Kirn, R. L. McVean, and R. A. Porter (ANL). Within a twelve-inch U-238 blanket, a number of square cylindrical cores of different compositions have been brought to criticality in the ZPR-III critical machine. The compositions of these cores were selected to test the cross section sets and multigroup methods of calculation used in the fast power reactor field. Thus, assemblies with U-238 to U-235 ratios varying from 1:1 to 7:1 were examined. Graphite was then added to a series of assemblies in order to simulate the spectrum of somewhat more dilute high-power reactors. Critical mass and dimensions, fission ratios, and Rossi-alpha were determined for each assembly. Reactivity coefficients for selected materials were determined on several of the assemblies. Most of the techniques used in this series of assemblies were the same as described previously.^{1,2,3} A few of the significant results are given in the accompanying table. The patterns of these results are discussed from the experimental point of view.

Each core investigated was surrounded by a blanket at least 12 inches thick which consisted of 83.3% U-238 and 7.3% stainless steel. The assumed density of carbon in the core was 2.25. The fission ratios and reactivity coefficients for the first case listed in the table were determined at the center of a core having the same composition as listed, but with a low density inner blanket.

References

1. ANS Meeting, October 1957.
2. ANS Meeting, December 1956.
3. B. C. Cerutti, H. V. Lichtenberger, D. Okrent, R. E. Rice and F. W. Thalgott, "ZPR-III, Argonne's Fast Critical Facility," Nuc. Sci. and Eng. I, 126 (May, 1956).

CRITICAL ASSEMBLY EXPERIMENTAL RESULTS

Core Composition, Volume %					Critical Mass, Kg U-235	Central Reactivity Coefficient $\Delta k/k$ per Kg Relative to Void				Fission Ratios				Prompt Neutron Lifetime From Rossi alpha (sec $\times 10^4$)
U-238	U-235	SS	Al	C		C	B-10	U-235	Pu	U-238	U-234	U-233	Pu-239	
15.7	14.0	12.3	31.5	—	136.0	0.0022	-0.062	0.0071	0.013	0.074	0.56	—	1.24	8.65 $\pm$ 0.4
38.0	11.7	15.0	21.6	—	151.8	—	—	—	—	0.051	0.43	—	1.23	8.1 $\pm$ 0.4
57.9	11.8	19.3	—	—	155.8	—	—	—	—	0.044	0.33	1.61	1.22	6.75 $\pm$ 0.4
71.7	9.51	9.18	—	—	240.6	-0.00081	-0.079	0.0056	0.0097	0.036	0.30	1.60	1.17	7.1 $\pm$ 0.4
34.9	9.39	9.18	—	23.7	176.8	0.00085	-0.101	0.0070	0.0118	—	—	—	—	10.7 $\pm$ 0.5
20.5	9.42	9.18	—	33.8	156.5	—	—	—	—	—	—	—	—	—
Neg.	9.31	9.18	—	47.3	—	—	—	—	—	—	—	—	—	—

Not available

Analysis of Recent Dilute Fast Neutron Critical Experiments, by W. B. Loewenstein and J. Schultheis (ANL). Recent critical experiments at the Argonne ZPR-III facility were performed to obtain information on assemblies containing fast neutron spectra which are similar to those expected in the larger more dilute fast reactor systems. These experiments contained much smaller amounts of primary fissionable material than would be found in the actual systems of primary interest. A conceptual fast power breeder reactor with an 800 liter core volume would require a critical mass of about 450 kg of Plutonium or 700 kg of Uranium-235. In the experimental program no critical assembly contained more than 245 kg of Uranium-235 in the core. This was accomplished primarily by the use of higher density systems with graphite replacing the sodium.

Prior to the performance of such experiments calculations are done to determine the general properties of such assemblies. These include a realistic estimate of the expected critical condition. The optimum cylindrical cores, those with minimum critical mass requirements for a given composition satisfy the relationship $0.8 < L/D < 1$ where L is the core height and D is the core diameter.

Calculations were made primarily in spherical geometry using the S_n method.¹ These results were then interpreted to cylindrical geometry using reasonable shape factors based on both calculation and experiment. The shape factor (S.F.) is defined as the ratio of the critical mass of a spherical core to that of a cylindrical one. Both cores have the same composition and similar reflector parameters. For many such systems a shape factor such that $S.F. = 0.95 \pm 0.01$ appears to be reasonable in the L/D range of interest.

Fission ratios calculated at the center of the core are used to compare with those experimentally observed. The major current implication of some of these results is that the assumption of a fission spectrum flux above the U-238 fission threshold may not be justified, particularly for those systems with the more degraded spectra. The shape of the spectrum determines the scattering of neutrons from above the U-238 threshold to energies below 1.4 Mev. Graphite has virtually no nonelastic cross section until one attains very much higher energies. For this material the degradation discussed above is due largely to elastic effects. The energy loss per collision is quite well known, but to quantitatively predict the over-all effect one must determine the spectrum. The calculations were done using the fission spectrum and it can be seen that the measured $\sigma_f^{238}/\sigma_f^{235}$ are consistently lower than those calculated. It is also true that for systems containing large amounts of U-238, a neutron above 1.4 Mev is worth more towards maintaining the chain reaction than one somewhat below 1.4 Mev. The latter effect may explain the rather low calculated critical masses.

Reference

1. B. G. Carlin, "Solution of the Transport Equation by S_n Approximations," LA-1891 (Feb. 1955).

Physics Calculations on Four Los Alamos Graphite Moderated Critical Assemblies, by H. Iskenderian (ANL) and C. Byers (LASL). Diffusion theory calculations were made on four Los Alamos uranium-graphite critical assemblies. A condensed table of these assemblies is shown below. The disadvantage factor values given are the averages for all the foil-graphite configurations in each assembly and were obtained by P-3 calculations. The critical bucklings were evaluated by two group diffusion theory, by iteration. To allow for the fissioning of epithermal neutrons, three group calculations were made for the equivalent bare reactors. The agreement between the calculated and experimental loadings was satisfactory, allowing for the uncertainty in the impurities in the graphite used. In assembly #2 there was a uniform distribution of slots: 0.54 in. $\times$ 2.9 in. cross section in every other fuel and running the entire length of the core, but not into the reflector. Allowance was made for streaming of neutrons with the aid of Behren's formula, accounting approximately for the non-slotted reflector.

Using the same method of calculations, the critical loading of the cold clean TREAT reactor, without its slots, was evaluated.

CONDENSED DATA OF LOS ALAMOS CRITICAL ASSEMBLIES
(Graphite CS-312 Grade, $d = 1.67$)

Assembly	Dimensions (inches)	Volume (cu. ft)	Kgs of Oralloy	C/O _U	d_{eff}	Calculated Averaged Disadvantage Factors
1	48×48×48	64	7.985	6650	1.495	1.09
2*	45×45×48 plus 3 fuel tubes	57	9.727	4366	1.342	1.068
3	39×42×48	45.5	7.634	4937	1.494	1.042
4	36×36×40	30	7.921	3140	1.4955	1.029

*The additional 3 fuel tubes were averaged over the reactor.

Notes:

- 1) Effective graphite density in reflector = 1.547 g/cc.
- 2) Volume fraction of aluminum (2S) in core and reflector = 0.0614.
- 3) Thickness of reflector for all assemblies = 1 ft.

Zero Power Test Studies on Martin Power Reactor, by W. E. Osmeier and H. B. Rosenthal (MARTIN). Experimental results have been obtained from the Zero Power Test performed on the Martin Power Reactor core. The reactor is a heterogenous water-moderated, fully enriched core using tubular fuel elements. The core is cylindrical in shape and contains approximately 18.6 kilograms of U-235 with a total reactivity of about 15%. In the core are six "Y" shaped rods which are prototypes of the control rods in the Martin Power Reactor. The Zero Power Test reactor is controlled by three blade-shaped control rods (critical experiment rods). In addition, two or three "Y" shaped rods are used for safety depending on experimental requirements.

The design of the Zero Power Test core required that the system first be built up with dummy fuel bundles. Multiplication runs were performed by replacing these bundles with fuel bundles. The system was brought critical based on the reciprocal multiplication curve and achieved criticality with 700 fuel elements or a critical mass of 7.9 kilograms with no Martin Power Reactor rods in the core.

After the initial criticality, the core had to be built up to its design diameter of 21 inches. This was accomplished by alternately inserting prototype rods and fuel within the limits of the Critical Experiment control system. Since more prototype rods were required than calculations had indicated, the experimental program had to be revised to obtain the optimum amount of control.

Reactivity was determined as a function of water height to evaluate the end reflector. The Critical Experiment control system was calibrated.

Temperature coefficients of reactivity were measured for the system, both with and without a pressure vessel. Total reactivity of the core was determined, both with and without a pressure vessel.

To determine optimum control of the Martin Power Reactor "Y" rods, various arm lengths were studied as well as various orientations of these rods. Having obtained the optimum arm length and orientation, flux and power distributions were studied with this optimum core.

The KAPL High Temperature Critical Assembly, by J. A. Bistline, C. S. Hofmann, and R. G. Luce (KAPL). The KAPL High Temperature Critical Assembly makes possible for the first time critical experiments on water moderated reactors at operating temperatures up to 550°F.

The facility is being used to measure temperature coefficients, flux distribution, reactivity coefficients etc. on a variety of slab type reactors. These fundamental nuclear data are of great importance in establishing analytical methods. The core components temperature control system and the general facility will be described in detail. Briefly, the core is variable in 2 in. x 4 in. units up to a 36 in. x 44 in. array. Core heights up to 60 in. can be accommodated. The present array of 2 in. x 4 in. x 40 in. Zircaloy-3 cans contain spacer assemblies of Zircaloy. These spacer assemblies position the fuel, additional Zircaloy strips, the control rods and boron steel strips as necessary for the desired mockup. The fuel is enriched uranium metal .0013 in. thick clad with 10 mils of Zircaloy-2. This present core structure can be varied to give metal to water ratios from 2.1-1 down to 1-1.7. The maximum fuel density is 0.22 gm/cc. A total of 19 control rods can be used in the assembly. They are modified SIW mechanisms. Two remotely controlled sample withdrawing devices are used to remove material in a distributed manner from the core. Nuclear monitors are located near the core through instrument tubes in the head of the assembly. Chambers capable of operating at operating temperatures have been developed.

The pressure vessel is a right circular cylinder 60 in. I.D. and 150 in. high with an elliptical dished bottom head and a 19 in. thick flat top head. Almost all penetrations into the vessel are through the head and all components inside the vessel are supported from the head. This allows the pressure vessel to be lowered away from the core to permit changes in the core structure and loading while retaining a control safety margin with cocked safety. Because of the top head thickness, limits on the rate of change of water temperature are required to keep thermal stresses to a value that affords an acceptable cyclic life.

The external loop is similar to pressurized-water reactor loops with a pressurizer, coolant circulation pumps, a steam water heat exchanger, electrical heaters, an air water cooler, a hold tank, ion columns, fill pumps, a makeup and test pump, and blow-off tank. A flow of 180 gal./min. is used in the system. The heating and cooling cycles are automatically programmed to maintain the required heating and cooling rate. The steady state temperature is likewise maintained in the pressure vessel at the set temperature $\pm 0.3^\circ\text{F}$ by automatic control. The maximum operating temperature and pressure is 550°F and 1200 psi. The assembly may be operated cold without making a pressure seal on the vessel.

Initial Experiments in the KAPL High Temperature Critical Facility, by R. G. Luce, W. Skolnik, and J. Bistline (KAPL). Up to the present time there have been very few critical experiments performed at other than low temperatures. There is now in operation at KAPL a critical assembly facility capable of operating with water temperatures up to 550°F. This paper describes the initial experiments conducted in this facility. The first experiments have been directed toward the study of critical reactors of simple geometrical configuration. Such data is of great importance for comparing with the existing analytical methods now used in Naval Reactor design.

The first reactor studied in the high temperature facility was an essentially uniform slab with low buckling in two directions and high buckling in the third direction, i.e., 30 in. high by 30 in. wide by 8 in. thick. This reactor had high leakage and hence its properties were very temperature dependent. A fuel load was chosen which made the core critical near room temperature. Then by increasing the fuel load in steps, the critical mass was determined as a function of temperature. Each step was obtained with control rods completely withdrawn from the core. At some of the critical steps measurements of the temperature coefficient of reactivity were made and thermal flux traverses along the axes of the core were obtained.

The core under study contained zirconium and 93% enriched uranium only. The core structure is made up of 2 in. x 4 in. cells. Each cell consists of a zirconium box containing a many layered sandwich of zirconium plates each .029 in. thick. The fuel is zirconium clad fully enriched uranium, the meat thickness being 0.0013 in., which can be inserted into the zirconium sandwich. Metal-water ratio was adjusted by the addition of zirconium sheets to the sandwich.

Comparisons will be drawn between the experimental results and theoretical calculations using few-group diffusion theory. Of prime importance is the comparison of the variation of reactivity with temperature and the comparison of the thermal flux traverses. For the few-group calculations the material in a cell was homogenized by transport theory calculations and these will be discussed. In addition, the temperature coefficient measurements will be compared with the results of a few-group adjoint theory calculation.

Critical and Exponential Experiments

Thoria-Uranium-D₂O Exponential Measurements in a 3-Foot Diameter Tank, by W. J. Sturm, W. C. Redman (ANL). A series of material buckling measurements, begun earlier^a in a 5 foot diameter tank, has been transferred to and completed in a smaller facility. With increasing fuel concentration, neutron multiplication and fuel requirements were calculated to become excessive in the larger tank well before the survey of the core assemblies would be completed. Consequently, a facility of smaller 3-foot diameter was fabricated to extend the range of measurements.

The newer system, provided with automatic traversing and print-out equipment for radial and axial neutron distribution determinations, used a B¹⁰F₃ proportional counter for neutron detection and Sb-Be as a source of low energy photoneutrons. In view of the extreme sensitivity of an exponential assembly to externally generated neutrons, source storage during neutron background evaluation had to be very effective; a storage geometry which permits safe, effective, and rapid transfer from the exponential has been developed.

The fuel was a mixture of oxides of thorium and uranium-235 (THUD fuel), a material expected to have high physical and chemical stability in a reactor atmosphere, and to offer possibilities of breeding in a boiling water power system. Small cylinders of this mixture were pressed, sintered and stacked inside aluminum to form a fuel pin 60 inches long and $\frac{5}{16}$ inch in diameter. Two versions of this fuel were available in quantity, the first having a thorium to uranium-235 atomic ratio of 25:1; the second an atomic ratio of 50:1.

Measurements consisted of the determination of material buckling over a range of fuel to moderator ratios for each of the fuels available. Lattices were assembled to form a hexagonal array of fuel pins with center to center spacings varying from $\frac{3}{4}$ inch to 3 inches, which is equivalent to moderator to fuel ratios between 9.6 and 179, and traversed the region of maximum buckling in each case. Data evaluation was complicated somewhat by the rather high intensity of photoneutrons generated in the lattice by γ -rays from the fuel itself; the resulting background of neutrons had to be accurately measured and involved in the calculations at each data point.

Final values of B_m^2 for the lattices measured in this series are tabulated below; B_m^2 is reported in metric units of 10^{-6} cm^{-2} and an estimated probable error of $\pm 40 \times 10^{-6} \text{ cm}^{-2}$ applies to all results.

TABLE I

Lattice	Spacing	V_m/V_f	B_m^2 (25:1)	B_m^2 (50:1)
2a	$\frac{3}{4}$	9.58	1963	555
3a	$1\frac{1}{8}$	23.7	2012	812
4a	$1\frac{1}{2}$	42.6	1750	581
6a	$2\frac{1}{4}$	100.	1120	202
7a	$2\frac{3}{4}$	137.	887	---
8a	3	179.	571	49

A radial neutron distribution was measured for each lattice and the extrapolation distance was found to increase monotonically with increasing fuel concentration. Material buckling evaluation was made from the measured radial distribution in each case.

Since all buckling measurements are lower than those predicted theoretically, some effort has been devoted to a search for possible systematic experimental causes for the discrepancy. Effects of various loadings near the tank edge, as well as effects due to photoneutron production by shortlived fission products have been measured but do not account for the variation.

Reference

- a. W. J. Sturm, W. C. Redman and M. Copic, "Exponential Studies of a Ceramic Mixture of Thoria and Urania in Heavy Water Moderator", Paper 11-2 ANS, Oct. 1957.

Subcritical Experiments

Neutron Multiplication Measurements of Heterogeneous 3.1 Per Cent Enriched Uranium-Water Systems, R. C. Lloyd, R. B. Smith and E. D. Clayton (GE-H). The design of efficient systems for processing spent power reactor fuel elements requires information on the critical masses involved. To obtain some of the needed information, approach-to-critical experiments were carried out with light-water-moderated and -reflected hexagonal-lattice assemblies using enriched solid uranium fuel elements. The U^{235} content of the uranium was 3.063 wt %. Experiments were made for fuel elements of 0.925-, 0.600-, and 0.300-in. O.D. with the H_2O/U ratios for lattice spacings to span maximum buckling and minimum critical mass.

The experiments were carried out in a 4-ft-dia by 5-ft-deep tank of water placed in the reactor room of the Hanford Thermal Test Reactor. The lattice frameworks were so placed in the tank as to provide effective infinite water reflection on all sides. Two or three 8-in.-long fuel elements were loaded into Lucite tubes and placed in the lattice framework.

A pencil-size neutron source, 0.22 gm RaBe, was placed between the fuel rods at the center of the lattice during approach measurements. Three BF_3 counters were used to monitor the neutron multiplication. Two safety channels with control and trip circuits from the TTR were used in conjunction with four safety and two control rods for each experiment. A fast water-moderator dump system was also provided.

Loadings were taken to 96% of the critical mass as determined by inverse multiplication plots of the three BF_3 channels.

The critical number of fuel rods, N_c , was obtained by a linear least-squares fit to the $\sqrt{N}/CR$ -vs- N plot of the center counter. The fit was made to the region for N between 85 and 96% of the critical number of rods. The poisonous effects of the center counter, source and holder, and safety rod guide tubes were evaluated during the critical approaches.

The experimental results are given in the table together with the values of the minimum critical mass and maximum buckling for each rod size obtained by interpolation of the data.

Lattice Spacing (in.)	Rod Diameter (in.)	H_2O/U (vol. ratio)	Loading Length (in.)	No. of Critical Rods	Critical Mass (cylindrical, lb)	Critical Mass (spherical, lb)	Buckling ($cm^{-2} \times 10^{-6}$)
1.500	0.925	1.89	16	72.1	531	498	13,099
			24	60.0	663		
1.600	0.925	2.29	16	63.5	468	441	13,329
			24	52.4	579		
1.700	0.925	2.72	16	58.6	431	412	13,137
			24	49.3	544		
1.800	0.925	3.17	16	56.7	418	395	12,793
			24	46.9	518		
1.930	0.925	3.89	16	60.4	445	428	11,385
			24	48.6	537		
1.000	0.600	2.06	16	152.1	471	439	13,732
1.100	0.600	2.71	16	118.6	367	339	14,588
1.200	0.600	3.41	16	104.5	324	299	14,351
1.300	0.600	4.18	16	98.9	306	285	13,647
1.420	0.600	5.18	16	100.3	311	293	12,345
1.600	0.600	6.84	16	122.0	378	357	9,826
0.600	0.300	3.41	16	387.5	299	274	14,956
0.700	0.300	5.00	16	296.5	229	207	16,131
0.800	0.300	6.84	16	272.3	210	193	14,539
0.900	0.300	8.92	16	285.7	221	206	12,396

-	0.925	3.3	-	-	-	394 ^a	
-	0.925	2.3	-	-	-	-	13.330 ^b
-	0.600	4.5	-	-	-	284 ^a	-
-	0.600	2.8	-	-	-	-	14.600 ^b
-	0.300	7.0	-	-	-	193 ^a	-
-	0.300	5.0	-	-	-	-	16.130 ^b

^aMinimum critical mass by interpolation.

^bMaximum buckling by interpolation.

The bucklings were calculated using published¹ reflector savings values for slightly enriched uranium.

¹Kouts. H. C., et al, "Buckling Measurements with 1.027% Enriched Rods". BNL Log C-7592, November 5, 1953.

k_{∞} Measurements of 2 Per Cent U^{235} Enriched UF_4 in Paraffin, V. I. Neeley (GE-H). Values of k_{∞} for low enriched materials are of interest in the processing and handling of power reactor fuels.

The measurement of k_{∞} for 2 per cent U^{235} enriched UF_4 paraffin moderated at a hydrogen-to- U^{235} ratio of 195 is of special interest because critical mass studies of the same material have been made at Oak Ridge National Laboratory and a comparison of these results may be made.¹

In these experiments a neutron absorbing agent was added to the material such that for an infinite medium of the material the multiplication factor would be unity. A finite assembly of the material was then placed in the cavity of the Physical Constants Testing Reactor. The central portion of this assembly was then used as the test sample; the rest of the material then acting as a buffer region to obtain the correct incident neutron energy spectrum on the test sample. The criteria were that with the correct incident neutron energy spectrum and correct poisoning the removal and replacement of the test sample with a void would produce no perturbation on the reactor.² The incident flux and poison were then adjusted to meet these criteria. The change in k_{∞} from the poisoned to the unpoisoned case is assumed to be only a function of the change in the thermal utilization.²

Then for the unpoisoned system

$$k_{\infty} = \eta \epsilon p f$$

and for the poisoned system

$$k'_{\infty} = \eta \epsilon p f' = 1$$

then

$$k_{\infty} - 1 = \frac{k_{\infty} - k'_{\infty}}{k'_{\infty}} = \frac{f - f'}{f'} = \frac{(\sum_a V)_{\text{poison}}}{\sum_i (\sum_a V)_i}$$

where

$$f' = \frac{(\sum_a \phi V)_{\text{fuel}}}{\sum_i (\sum_a \phi V)_i + (\sum_a \phi V)_{\text{poison}}}$$

$$f = \frac{(\sum_a \phi V)_{\text{fuel}}}{\sum_i (\sum_a \phi V)_i}$$

since the material has been assumed to be homogeneous. The assembly was 24 x 24 x 30 in. with a 6 x 6 x 12-in. central region as the test sample. The material was in the form of blocks of various dimensions (4 x 4 x 4 in., 2 x 2 x 4 in., 4 x 4 x 1 in., etc.) A total of 2909 lb of material containing 18.4 kg of U^{235} was used. The poison consisted of 0.005-in.-thick film of boron carbide-impregnated polyethylene containing 4 wt % boron carbide. When this film was wrapped around the individual blocks the multiplication factor of the assembly was unity. Flux traverses were made in the test sample and in the buffer region with cadmium covered and bare gold foils 0.25 x 0.5 x 0.005 in. in dimensions. These traverses are also used to insure that the correct neutron energy spectrum was incident on the test sample.

Further measurements were made to observe the change in k_{∞} as a function of cadmium ratio in the test sample. The purpose of these measurements was to determine any change in k_{∞} due to a spectral change produced by the addition of poison and any mismatch of the flux between buffer region and test sample. The cadmium ratio in the unpoisoned test sample using 0.005-in. thick gold and 0.040-in. thick cadmium was 2.37, and in the poisoned test sample it was found to be approximately 2.33. The cadmium ratio in the poisoned test sample was varied over the range 2.31 to 2.35 by changing the poison in the buffer region and by changing the PCTR loading configuration. The change in $k_{\infty}-1$ over this range of cadmium ratios was found to be less than 0.5% of $k_{\infty}-1$. An error of 1% of $k_{\infty}-1$ was then assigned to the final result to cover any error introduced in this manner.

The value of k_{∞} for 2% U^{235} enriched UF_4 paraffin-moderated at a hydrogen-to- U^{235} atomic ratio of 195 was found to be 1.216 ± 0.013 .

¹J. J. Lynn, J. T. Mihalczo, W. C. Connelly, "Homogeneous Hydrogen Moderated Critical Assemblies with 2 Per Cent U^{235} Enriched Uranium," ORNL-2609 (1958).

²D. J. Donahue, D. D. Lanning, R. A. Bennett, R. E. Heineman, "Determination of k_{∞} from Critical Experiments in the PCTR.," Nuc. Sci. and Eng. 4, 297 (1958).

Critical Experiments

Critical Dimensions of Unreflected BeO Systems Fueled by Thin Enriched Uranium Foils, F. A. Kloverstrom, R. M. Deck and A. J. Reyenga (UCRL-L). A series of critical measurements on unreflected parallelepipeds built of enriched (93.2%) uranium foils and BeO blocks have been made for five fuel densities. The object of this program was to provide data for normalization of machine criticality calculations, and the systems were therefore made "clean." The only material present other than BeO and U was a small amount of Teflon (foil coatings) and a steel diaphragm, the effect of which was experimentally determined.

The assembly machine consists of a horizontal 0.020-in. steel diaphragm suspended above a vertically moving table. The major portion of an assembly is built on the table and the remainder on the diaphragm. The two parts are assembled remotely by lifting the table. During a measurement, the base dimensions of a system are fixed, the height being the variable. Critical heights are measured by extrapolation of inverse multiplication curves. The effect of the diaphragm is found by measuring the effects of additional thicknesses of steel and extrapolating to zero diaphragm thickness.

Enriched uranium foils having nominal thicknesses of 0.001 and 0.002 in. were sandwiched between 1-in.-thick BeO blocks, forming horizontal foil planes through the systems. The overall fuel density was variable by changing the foil thickness and/or spacing. Several measurements have been made in which the average fuel density is kept constant while the foil thickness is changed to provide information on self-shielding and flux depression effects of the foils.

The thermal neutron absorption cross section per BeO atom has been found to be 12 ± 2 mb by a pulsed neutron source experiment. The results obtained to date are summarized in the table. "BeO/U²³⁵" refers to the ratio of the numbers of Be to U²³⁵ atoms actually present in the assemblies. The BeO density is 2.86 g/cm³. Critical heights are accurate to within ± 0.2 in.

Critical Dimensions of BeO Moderated Unreflected Systems

(ρ_{BeO} = 2.86 g/cm³)

Uranium Foil thickness (in.)	Spacing between foil planes (in.)	BeO/U ²³⁵	Average U ²³⁵ density g/cm ³	System base (in.)	Critical Height (in.)
0.008	1	247	0.109	24 x 24	20.2
0.004	1	493	0.0545	24 x 24	22.0
0.008	2	493	0.0545	24 x 24	22.3
0.002	1	986	0.0272	24 x 24	25.0
0.004	2	986	0.0272	24 x 24	25.8
0.006	3	986	0.0272	24 x 24	27.5
0.001	1	1920	0.0140	30 x 30	21.1
0.002	2	1920	0.0140	30 x 30	21.6
0.002	2	1920	0.0140	30 x 24	26.0
0.003	3	1920	0.0140	30 x 24	27.3
0.001 ^a	1	3826	0.00702	36 x 36	22.9
0.002 ^a	2	3826	0.00702	36 x 36	24.2
0.001	2	3826	0.00702	36 x 36	22.7
0.002	4	3826	0.00702	36 x 36	24.2

^a These fuel foils have a triangular shape rather than square.

Clustering Experiments with $\text{ThO}_2\text{-UO}_2$ Fuel in D_2O , W. C. Redman and J. A. Beidelman (ANL). The program of critical experiments with lattices of thorium-uranium fuel in heavy water has been extended from a study of uniform lattices to the determination of properties of clustered fuel arrangements. Such data are of interest for the design of heavy water power reactors. The fuel, contained in 5/16-in.-OD aluminum tubes, has a diameter of 0.23 in., a density of 8.4 gm/cm^3 and a thorium to U^{235} atom ratio of 25.

The first phase of the clustering experiments related the parameters of importance, namely, cluster spacing, rods per cluster, and buckling, to the bucklings for uniform systems. For successive triangular lattice spacings of 3, 3-3/4 and 5 in., the available fuel pins were inserted with 3/8 in. triangular spacing in clusters around the lattice points. The critical water heights were observed for thirty configurations achieved by uniform stepwise removal of fuel pins from the periphery of the clusters. For each sequence of measurements at constant pitch, an assembly of maximum buckling was achieved. Some representative results from these and related observations are shown in the figure.

The effect of clustering was also investigated in a core arrangement suitable for the proposed heavy water operation of EBWR. Into each of thirty-six 4×4 in. clusters positioned to match the central region of the EBWR core were placed 28 fuel pins, spaced in a triangular pattern first at 3/4 in., next at 3/8 in. Several nuclear characteristics of these two core arrangements are compared in the table.

Properties of Lattices with EBWR Geometry

Pattern	$B_g^2 (\text{in.}^{-2})$	Center $\bar{\phi}$	Edge $\bar{\phi}$	$\bar{\phi}_{\text{Mod}}/\bar{\phi}_{\text{Fuel}}$
Slightly clustered (3/4-in.- Δ)	0.0122	0.88	0.94	1.10 ± 0.02
Fully clustered (3/8-in.- Δ)	0.0115	0.89	0.84	1.4 ± 0.1

Geometrical Buckling vs. Degree of Clustering for the Moderator to Fuel Ratio of a Uniform 3/4-in.-Triangular Lattice.

An investigation of the nuclear properties of several alternate forms of a pressure-tube-type heavy-water system is now in progress. Fuel pins are clustered in a 3/8-in.-triangular pattern within a 2 3/8-in.-ID, 2 1/2-in.-OD aluminum tube. Square lattices of 4 1/2 in. and 5 7/64 in. are used. In addition to heavy water, measurements include the substitution of light water and void in the fuel-bearing region inside the 2 3/8-in.-ID aluminum tubes.

Reactivity and Flux Measurements in Highly Reflected U^{235} -Bismuth-Graphite Critical Experiments, T. C. Engelder, H. W. Giesler and J. P. Farrar (B & W). The Liquid Metal Fuel Reactor Experiment (LMFRE) research and development program includes a series of critical experiments on U^{235} -bismuth-graphite cores, highly reflected by graphite. These experiments have a two-fold purpose: (1) to provide experimental data on highly reflected cylindrical cores of relatively simple geometry that require two-dimensional analysis, and (2) to supply nuclear and engineering parameters from a specific engineering mock-up of the LMFRE which may include such features as fuel plenums, fuel in the end and side reflectors, test holes, etc. Data from the first phase of this program are presented in this paper.

The LMFRE critical experiment is designed to provide a high degree of flexibility. It employs a movable table assembly driven by a combination electric and pneumatic drive through a lead screw and nut. When closed the assembly presents a level surface, 10 ft square, designed to support a total load of about 60 tons. The core region of the critical assembly is a pseudo-cylinder 56 in. in diameter by 8 ft high consisting of a vertical graphite honeycomb with a 4-in.-square modulus. Around the outside of the honeycomb are stacked solid graphite blocks, also 4-in. square. The core is filled with square bars of bismuth and graphite, and fuel in the form of 0.010 in. thick strips of U-Al alloy, containing approximately 18 wt % highly enriched uranium. With various combinations of fuel, bismuth, and graphite the volume ratio of bismuth to graphite in the core can be varied from zero to 2.0 and the atom ratio of U^{235} to bismuth can range up to 1200×10^{-6} in increments of 150×10^{-6} . Four vertically driven control rods are provided, which can be positioned at any core channel.

In the first set of experiments, the uranium concentration was varied, keeping the side and end reflector 2 to 3 ft thick. For each concentration several pseudo-cylinders were constructed with slightly different core cross-sectional areas, and their excess reactivities determined by calibrated control rods. Cores of different cross section but equivalent cross-sectional area were found to have the same reactivity within five cents. Therefore, the critical diameter for each concentration can be reported.

Assembly	N_{235}/N_{Bi} ($\times 10^{-6}$)	V_{Bi}/V_G	Reflector Thickness (in.)		Core Height (in.)	Critical Dia (in.)
			End	Side		
A	600	0.5	24	34	48.6	46.9
B	750	0.5	24	30	48.6	54.5
C	1200	0.5	24	39	48.6	36.6
CA	1200	0.5	30	37	36.4	41.4
CB	1200	0.5	18	40	60.8	34.4

In addition, radial, axial, and diagonal flux traverses with bare and cadmium covered gold foils were made in each assembly. Other factors influencing the accuracy and reliability of the data were measured. They include rod reproducibility, structural effects, temperature, room reflections, and table gap.

The intra-cell thermal flux structure was investigated in Assemblies B and C using calibrated Dy-Al foils and gold sector foils. Flux contour diagrams were developed and, for Assembly C, $\bar{\phi}_{23}/\bar{\phi}_{Bi}/\bar{\phi}_G = 1.000/1.004/1.037$ was found. Other measurements made on some of these assemblies include rod worths, radial fuel importance, $d\rho/dr$, and the effect of homogeneity and anisotropy.

The second set of experiments was devoted to measurements of reflector and geometrical effects. A large number of assemblies with $N_{235}/N_{Bi} = 1200 \times 10^{-6}$ and $V_{Bi}/V_G = 0.5$ were constructed with different height to diameter ratios and different end and side reflector thicknesses. Measurements on each assembly include the critical dimensions and radial and axial flux traverses with bare and cadmium covered gold and Dy-Al foils.

Neutron Spectra

Calculation of the Neutron-Leakage Spectra and Neutron Dose from Bare Critical Systems,* J. R. Knight (ORGDP). An accidental accumulation of sufficient fissionable material to produce a critical reaction results in the release of potentially serious amounts of radiation, and a method of determining the neutron dose that may be received by individuals near the reaction is of considerable importance. A method for calculating the neutron spectra and neutron dose for highly-enriched unreflected uranium solutions at various H/U²³⁵ atomic ratios has been developed. This method uses the output from a code on the IBM-704 for criticality calculations, and determines the average leakage-neutron energy, the neutron-leakage spectra, the first-collision neutron dose and the total neutron dose in rads per incident neutron flux. The calculations cover a range of H/U²³⁵ atomic ratios from 0 to 2000, which is essentially the entire range for which criticality is possible.

The calculations show that the average energy of the leakage neutrons decreases as the atomic ratio increases, with the total decrease over the range from 0 to 2000 being about 16%. The fraction of the leakage neutrons that are thermal increases from 0 to 11% as the atomic ratio increases from 0 to 2000, while the fraction of those less than 1000 ev increases from 0 to 23%. The first-collision neutron dose and the total neutron dose, in rads/neutron/cm², are essentially the same for each concentration. The first-collision dose decreases by about 21% over the range of atomic ratios studied, decreasing from 3.4×10^{-9} rads/neutron/cm² at an atomic ratio of 0, to 2.6×10^{-9} rads/neutron/cm² at an atomic ratio of 2000. It appears that the neutron dose is not a sensitive function of uranium concentration, and exact knowledge of the conditions at the accident would not be necessary to obtain reasonable dose estimates. The results of the calculations are summarized in the table.

Results of Neutron Leakage and Dose Calculations

H/U ²³⁵ Atomic Ratio	Average Leakage Neutron Energy (Mev)	Thermal Leakage (%)	First Collision Dose (rads/n/cm ²) $\times 10^{-9}$
0	2.14	0.00	3.36
20	2.05	0.04	3.00
200	1.99	1.58	2.90
2000	1.80	11.43	2.59

*Sponsor: H. F. Henry

Criticality and Lattice Theory

Calculation of Lattice Parameters of H₂O Moderated UO₂ Fueled Critical Assemblies, W. H. Arnold, Jr. (WAPD). A system, consisting largely of hand calculations, is described which predicts to within 1% in reactivity the critical masses of fourteen separate UO₂ - H₂O critical assemblies. Eight of these were the Al clad 1.3% enriched cores done by the Bettis TRX group.¹ One was an experiment on steel clad 4% enriched UO₂ performed under the Nuclear Ship Savannah program.¹ The other five were performed by WAPD—three on steel clad 2.7% UO₂ Yankee criticals,² two on steel clad 4.4% BR-3 criticals.³ Table 1 contains a brief description of these experimental results.

TABLE 1
Description of Experimental UO₂ - H₂O Cores

Case	Laboratory	Description	W/U	Critical B ² , m ⁻²
1	W Bettis	1.3%, 7.53 g/cm ³ , .6" dia., Al clad	3	28.37 ± .06
2	W Bettis	1.3%, 7.53 g/cm ³ , .6" dia., Al clad	4	30.17 ± .06
3	W Bettis	1.3%, 7.53 g/cm ³ , .6" dia., Al clad	5	29.06 ± .07
4	W Bettis	1.3%, 7.52 g/cm ³ , .388" dia., Al clad	4	25.28 ± .10
5	W Bettis	1.3%, 7.52 g/cm ³ , .388" dia., Al clad	5	25.21 ± .10
6	W Bettis	1.3%, 10.53 g/cm ³ , .383" dia., Al clad	2.9	32.59 ± .15
7	W Bettis	1.3%, 10.53 g/cm ³ , .383" dia., Al clad	3.6	35.47 ± .18
8	W Bettis	1.3%, 10.53 g/cm ³ , .383" dia., Al clad	4.9	34.22 ± .13
9	W APD	2.7%, 10.7 g/cm ³ , .300" dia., SS clad	2.2	40.7 ± .4
10	W APD	2.7%, 10.7 g/cm ³ , .300" dia., SS clad	2.9	53.2 ± .7
11	W APD	2.7%, 10.7 g/cm ³ , .300" dia., SS clad	3.9	63.3 ± .4
12	W APD	4.4%, 10.7 g/cm ³ , .300" dia., SS clad	2.8	79.3 ± .8
13	W APD	4.4%, 10.7 g/cm ³ , .300" dia., SS clad	3.7	84.7 ± 1.4
14	E-W NSS	4.0%, 9.45 g/cm ³ , .444" dia., SS clad	3.5	85.6

The calculation is based on a scheme similar to the simple four factor model, except that an additional factor is added to account for the effect of U-235 resonance fissions on reactivity. This additional factor is derived from a feedback loop analysis of resonance fissions, which yields a reactivity formula as follows:

$$k = \eta f \epsilon p_{25} p_c P_{NL} \beta = \left[\frac{\eta f \epsilon p_{25} p_c}{(1 + L^2 B^2)(1 + \tau B^2)} \right] \left[\frac{p_{25}}{1 - \epsilon P_{NL1} \eta_{25}^R p_{25}^U (1 - p_{25})} \right]$$

The product ηf , as well as L^2 , is calculated using the SOFOCATE Wigner-Wilkins code whose input number densities are modified by transport theory unit cell flux factors as determined by the method of Amouyal and Benoist.³ ϵ is calculated assuming a homogeneous model: fast capture in U-238 is included entirely in p_{25} . p_{25} is obtained from the work of Hellstrand,⁴ to which a Dancoff correction modified by the work of R. J. French⁵ is applied. An arbitrary split in p_{25} is assumed: p_{25}^U is that part of p_{25} above the 25 fission resonances. τ is obtained from equivalent one group fast neutron transport and removal cross sections fitted to experimental values of age in water and various mixtures. p_c represents all non-uranium neutron capture above 0.625 eV (the upper energy limit used in the SOFOCATE code), and is calculated using a homogeneous model and calculated and measured resonance integrals which include $1/v$ absorption. p_{25} is also calculated assuming homogeneity, but the resonance integral was calculated as a function of energy self shielding from the measured resonance parameters (and a $1/v$ term) and plotted as a function of barns of scattering per 25 atom. η_{25}^R , the resonance η , was calculated in a similar manner as a function of Σ_S/N_{25} by dividing capture by fission resonance integrals to obtain an α value. P_{NL1} is taken equal to $(1 + \chi \tau B^2)^{-1}$, where χ is a number less than 1 which expresses the fraction of slowing down taking place above the 25 resonances. This term brings about an interaction between leakage and resonance fission.

In the present calculation, χ was taken as 0.8 and p_{25}^U as $p_{25}^{0.6}$. The results are presented in Table 2 for both calculated reactivity at the measured critical buckling and for those microscopic lattice parameters which have been measured.

TABLE 2
Comparison of Calculated and Measured Lattice Parameters

Case	Calculated k_{eff}	$-\partial\rho/\partial B^2, \text{ cm}^2$		ϕ_{H_2O}/ϕ_{UO_2}		β^a	
		Calculated	Measured	Calculated	Measured	Calculated	Measured
1	1.009	46.6	48 ± 2	1.136	$1.09 \pm .03$	.076	$.071 \pm .010$
2	1.008	44.2	46 ± 2	1.146	$1.14 \pm .03$	.066	$.059 \pm .009$
3	1.003	42.8	41 ± 2	1.157	$1.16 \pm .03$	.047	$.051 \pm .004$
4	.996	48.3	47 ± 2	1.086	$1.10 \pm .01$	.066	$.063 \pm .003$
5	.994	46.1	48 ± 2	1.090	$1.10 \pm .01$	.047	$.054 \pm .003$
6	1.010	38.6	41 ± 2	1.123	$1.10 \pm .01$	.080	$.078 \pm .004$
7	1.006	37.4	39 ± 2	1.129	$1.13 \pm .01$	.065	$.070 \pm .004$
8	1.001	36.7	36 ± 2	1.138	$1.13 \pm .01$	.048	$.057 \pm .003$
9	1.010			1.160		.100	
10	1.013			1.167	$1.16 \pm .05$	.078	$.076 \pm .002$
11	1.008			1.175	$1.16 \pm .03$	.061	$.060 \pm .005$
12	1.000			1.256	$1.25 \pm .07$	.070	$.070 \pm .005$
13	.996			1.271	$1.25 \pm .02$	.054	$.060 \pm .004$
14	.998			1.365	1.40	.060	

^aKouts, et al., Peaceful Uses of Atomic Energy, Vol. 12, P/1851, 1958.

^bDavison, et al., YAE-94, "Yankee Critical Experiments."

^cAmouyal and Benoist, J. Nuclear Energy, 6, 79 (1957).

^dE. Hellstrand, J. App. Phys. 28, 1493 (1957).

^eR. J. French, The Resonance Escape Probability in Clustered Fuel Rod Lattices, Trans-Am. Nuclear Soc., Vol. 2, No. 2.

Experimental and Theoretical Reactivity Studies of Stainless Steel Fuel Elements, A.H. Spano and D.R. Metcalf (PHILLIPS). Clean critical experiments have been conducted on a number of light water-moderated and reflected slab configurations of Spert III fuel elements. The objective of these experiments has been to obtain criticality data on relatively simple lattices which are either rod-free or single-rodded. These data may be used to verify separately the reactor constants and the control rod analysis used in a previous four-group multi-region calculation for the operational Spert III core.¹

In the earlier analysis of the stainless steel core, flux-weighted, two-dimensional radial diffusion solutions were combined with a one-dimensional axial solution for the eigenvalue and flux distributions over the core.² The results of these extensive calculations, performed for a variety of reactor conditions, showed the calculated k_{eff} to be generally about 3% greater than the corresponding experimental value. However, application of similar analytical methods to Zr-H₂O systems yields eigenvalue results which are in the neighborhood of 1% of criticality. This discrepancy in calculational results between stainless steel and zirconium reactor systems suggests the use of either incorrect stainless steel constants or an inadequate control rod theory. These possibilities were investigated by means of the unrodded and single-rodded experiments.

Water height experiments were performed with fuel elements (638 gm U-235; fuel cell size = $3 \times 3 \times 36$ in.; metal-to-water ratio = 0.298) arranged in various slab configurations, either rod-free or with a single, fully inserted, B¹⁰-SS, box-shaped control rod. Reactivity control was by water height only and measurements included those of critical height, h_c , and incremental water height worth. Experimental results are given in the table, where lattices are described by the array of circles for elements and crosses for poison rods.

Table of Experimental Results

Loading Pattern	Critical Height (inches)	Incremental Water Worth at $\bar{h}$ (cents per inch)	$\bar{h} \left(= h_c + \frac{\Delta h}{2} \right)$ (inches)
ooooo	23.49	47.2	23.61
ooooo	27.75	29.3	27.90
ooooo		28.7	28.03
ooooo	34.04	15.5	34.30
ooooo		15.0	34.46
oooo oooo ^a	28.33	28.9	28.54
oooo oooo		28.3	28.65
oooooooooooo	23.64	45.1	23.76
oooooooooooo	31.26	21.9	31.52
oooooooooooo		21.6	31.72
oooooooooooo	28.12	29.3	28.24
oooooooooooo		29.2	28.41
oooooooooooo	31.41	20.9	31.62
oooooooooooo		20.6	31.86
ooooo	34.26	14.4	34.68
ooooo		13.0	34.98
ooooo	29.37	28.9	29.58
ooooo		27.2	29.72

^a Rod-free split core case of two 3 x 4 slabs separated by a 3 in. water gap.

¹ A. H. Spano and D. R. Metcalf, "Quarterly Progress Report, Reactor Projects Branch," April, May, June, 1958, G. O. Bright ed., IDO-16489, p. 56 (January 19, 1959).

² J. E. Meyer, "Synthesis of Three-Dimensional Power Shapes - Flux Weighting Technique," Bettis Technical Review, Vol. 1, No. 4 (1957).

A Comparison of Some Criticality Calculations with Experiment,* C. W. Nestor, Jr. (ORNL). Recently several critical experiments (described in reference 1) have been done at ORNL with light water solutions of uranyl nitrate in an essentially bare sphere 27 inches in diameter. Criticality calculations were performed for the two experiments described in Table 1 with four procedures to obtain information concerning their reliability in calculations for simple reactors.

Two basic methods were utilized. First, hand calculations were done with bare reactor theory, using the empirical fast nonleakage probability formula developed by Trubey, Moran and Weinberg.² Also, two versions of multigroup diffusion theory calculation were employed: the Los Alamos FIRE code³ and the General Motors GM-GNU II code.⁴

There are three items of considerable importance in the calculation of light water moderated reactors, namely the neutron cross sections (particularly in the thermal group), the age to thermal and the boundary conditions on the flux and current. The cross sections were obtained from the Hughes-Schwartz compilation.⁵ The two multigroup codes use the Selengut-Goertzel (or Coveyou-Macauley) treatment of slowing down; as this procedure overestimates the age, the hydrogen transport cross sections were adjusted to give a calculated age to indium resonance of 30.2 cm², and this age was also used in the hand calculations. The conventional extrapolated boundary condition was used in the hand calculations; the FIRE calculations were done with an energy-dependent buckling; and the GNU calculations were done with an energy-dependent buckling and with a single buckling.

In general, good agreement was obtained between theory and experiment. The hand calculations and the multigroup calculations with a single buckling predict a lower multiplication factor than is actually observed, as a result of an overestimate of fast leakage. Results of the calculations are shown in Table 2.

TABLE 1
Reactor Characteristics and Fuel Solution Compositions
(concentrations in atoms/barn cm.)

Diameter:	69.2 (computed from volume)				
Container:	0.32 cm aluminum sphere				
Reactor	Fuel	Fissile U*	H	O	N
1	U ²³³	4.35×10^{-3}	6.62×10^{-2}	3.31×10^{-2}	1.17×10^{-4}
2	U ²³⁵	4.81×10^{-3}	6.63×10^{-2}	3.315×10^{-2}	1.75×10^{-4}

*Both reactors were critical with these concentrations.

TABLE 2
Calculated Effective Multiplication Factors

Method	Reactor 1	Reactor 2	
	k_e	k_e	
Hand	0.972	0.987	
FIRE	0.999	0.997	
GNU	0.985	0.977	constant B ²
GNU	0.998	0.992	energy-dependent B ²

¹D. W. Magnuson and Reginald Gwin, "Comparison of Critical Experiments for the Determination of Eta of U-233," Transactions of the American Nuclear Society, 2, 146 (June 1959).

²D. K. Trubey, H. Moran and A. M. Weinberg, An Estimate of the Non-Leakage Probability for Bare Aqueous Homogeneous U-235 Reactors, ORNL memorandum CF 57-11-15 (1957).

³F. W. Brinkley and C. B. Mills, One-Dimensional Diffusion Intermediate Reactor Code (Fire Code), Trans. Am. Nuc. Soc. 2, 234 (June 1959).

⁴C. L. Davis, J. M. Bookston and B. E. Smith, GNU II - A Multigroup One-Dimension-Diffusion Program for the IBM-704, General Motors Report GMR 101 (1957).

⁵D. J. Hughes and R. B. Schwartz, Neutron Cross Sections, Brookhaven National Laboratory Report BNL-325 (Second Edition).

Critical and Exponential Experiments

Multiplication Measurements with Slabs of Enriched Uranium, J. T. Mihalczo and J. J. Lynn (ORNL). A series of neutron multiplication measurements with arrays of 1 by 8 by 10 in. slabs of 93.4% U-235 enriched uranium metal have been made to provide data from which safety criteria for the storage of these fissile units can be established. Each slab contained 22.9 kg of U-235. A maximum of 125 units was assembled. The arrays studied were cubic lattices of the units and were usually parallelepipedal in shape. Arrays were both unmoderated and Plexiglas-moderated and were surrounded in most cases by a 1-in.-thick Plexiglas reflector. The lattice densities (ratio of fissile unit volume to lattice cell volume) were between 0.023 and 0.06.

Unmoderated lattices with a density of 0.06 would require 146 ± 5 units for criticality, while those with a density of 0.023 would require 250 ± 30 units. In lattices in which the fissile units are separated by 1 in. of Plexiglas approximately 27 units would be required for a critical array with a lattice density of 0.06 and about 75 units for a density of 0.023. Distributing Foamglas (containing 2% boron) throughout a moderated array increased the critical number of fissile units by a factor of 6, while Styrofoam had a small effect.

If the number of fissile units required for high subcritical multiplications in a specific lattice type is known as a function of the lattice density, and if a good estimate of the critical number of fissile units for one lattice density is available, the critical number of fissile units for other densities of that lattice type may be predicted. The number of fissile units required to produce a given inverse multiplication for the unmoderated and moderated arrays as a function of lattice density is shown in Figs. 1 and 2. (See figures on next page).

A Thorium-Uranium Exponential Experiment, C. H. Skeen and W. W. Brown (AI). Because of uncertainties in the knowledge of the nuclear properties of thorium fuel and lattices containing this fuel, an experimental study was made of a thorium based fuel that is to be loaded into the Sodium Reactor Experiment (SRE) in the near future. An exponential experiment was performed with a square-celled lattice of 7-rod elements ($\frac{3}{4}$ -in. diameter rods) spaced 9.5 in. apart in graphite. The fuel is a Th-U-235 alloy containing 7.6% uranium by weight which is 93.13 atomic per cent U-235. The fuel elements were 5 ft. long. The subcritical lattice was placed on the thermal column of a water boiler reactor which served as the source of neutrons for the assembly.

The thermal-neutron diffusion length of a solid block (47.5 x 47.5 x 63 in.) of the graphite used was first measured. The thermal-neutron distributions along the vertical and the horizontal axes of this block were determined from the activations of sets of bare and cadmium-covered indium foils. The value of the diffusion length obtained was 49.9 ± 0.2 cm. This value agrees with the value 50.8 ± 0.8 cm. obtained previously¹ for this block.

The material buckling of the lattice was determined from the thermal-neutron flux distribution measured parallel to the vertical axis of the assembly. A cell of the lattice is shown in cross section in the figure. From this distribution and the extrapolated width of the assembly, a value of the buckling was found to be $10.89 \pm 0.05/\text{m}^2$.

Some Physical and Geometrical Properties of a Unit Cell.

The thermal utilization of the lattice was determined using the known volume fractions of the constituents of a unit cell and the corresponding macroscopic cross sections along with the thermal-neutron distributions in these constituents. The distributions were obtained from activations of sets of bare and cadmium-covered dysprosium foils placed in the various parts of the central cell at a distance of 18 in. above the bottom of the assembly where the cadmium ratio no longer varied with height. The value of f obtained was 0.9445 ± 0.0009 .

The resonance escape probability (actually the probability that neutrons escape capture in the Th-232 present while slowing down to cadmium cut-off) was measured directly by activating Th-234 foils in the fuel of the central cell. Sets of bare and cadmium-covered foils were irradiated at a distance of 18 in. above the bottom of the assembly. The activity was determined by counting the 89.5-kev internal conversion x-ray from Pa-233 which follows the Th-232 (n, γ) reaction. The resonance escape probability p was then obtained from the equation

$$\frac{C}{B - C} = \frac{1 - p}{pF},$$

where F is the measured fraction of the thermal neutrons that were captured in the Th-232 of the fuel and B and C are the activities of the bare and cadmium-covered foils, respectively. The value of p obtained is $0.856 \pm .009$. The value of the effective resonance integral deduced from this measurement is 9.5 ± 0.8 barns (including $1/v$ capture). The surface-to-mass ratio of the thorium present in the fuel was $0.0858 \text{ cm}^{+2}/\text{gm}$. This value of the effective resonance integral can be compared with those obtained by extrapolation of measurements or calculations made elsewhere for this surface to mass ratio. Extrapolation of Sher's work² to our value of surface-to-mass ratio gives 12 barns, Pettus³ gives 10 barns and Davis⁴ 13 barns. Vernon⁵ calculates 13.5 barns for this quantity. Use of age-diffusion analysis together with the values of buckling and thermal utilization given above results in a value of 11 ± 2 barns for the resonance integral. It has been estimated⁵ that about 0.8 barns may have to be subtracted from this last value to account for the small amount of U-238 present in the fuel.

¹W. W. Brown, F. L. Fillmore and B. L. Scott; NAA-SR-3096 (1959).

²R. Sher, BNL 483 (C-26) 1958.

³W. G. Pettus, BNL 483 (C-26) 1958.

⁴M. V. Davis, priv. comm.

⁵A. R. Vernon, priv. comm.

Fig. 1. Number of Unmoderated Fissile Units as a Function of the Ratio of Fissile Unit Volume to Lattice Cell Volume for Various Inverse Multiplications.

Fig. 2. Number of Plexiglas-Moderated Fissile Units as a Function of the Ratio of Fissile Unit Volume to Lattice Cell Volume for Various Inverse Multiplications.

Graphite - Uranium Exponential Measurements for Large Fuel Elements, D. E. Wood and G. W. R. Endres (GE-H). Exponential measurements at Hanford on graphite-uranium lattices now include a wide range of physical parameters. Studies have been made for lattice spacings from $4\frac{3}{4}$ in. to 15 in., solid rod sizes from 0.926-in. diameter to 2.5-in. diameter, and fuel configurations of solid rod, tubular, tube-and-tube, and cluster elements. For some of the fuel geometries, different water-to-uranium and -aluminum ratios have been measured, and several studies of enrichments other than natural uranium have been made. Results obtained since the last previous reports^{1,2} include bucklings for 2.5-in. and 1.92-in. solid elements, 2.5-in. tube-and-tube elements, and 2-cluster elements, all natural uranium.

The bucklings measured for the large fuel elements are listed in Table I along with the corresponding volume ratios and extrapolation distances, λ .

The exponential piles used were 4 ft long, 5 to 6 ft wide and about 8 ft high, depending on the lattice spacings involved. Traverses were taken with BF₃ counters in all three directions in each pile. Bucklings were determined from

$$B^2 = \left(\frac{\pi}{a}\right)^2 + \left(\frac{\pi}{b}\right)^2 - B_{11}^2,$$

where a and b are the effective horizontal dimensions as determined from the horizontal traverses, and B_{11} is the reciprocal of the vertical relaxation length. Harmonic corrections were applied in the usual manner, but fast source and spontaneous background corrections were not needed since the cadmium shutter technique was used.

The horizontal dimensions of the exponential assemblies were small enough so that the error involved in the extrapolation distance, λ , introduces a significant error in the buckling. This does not affect the original purpose of the series of measurements, which was to survey many lattices as economically as possible, obtaining useful values on a relative basis at the sacrifice of precision in the absolute value of each buckling. It is intended that any lattice which looks useful for further work could be studied in a larger exponential or in the Physical Constants Test Reactor³ (PCTR). However, the data from this series could be useful for theoretical correlations and checks of calculational methods if the measurements were placed on a firmer absolute basis, so considerable effort has been expended in determining the best value of λ to use for each exponential assembly. Since λ may be different for each fuel size and lattice spacing, horizontal traverses have been taken in each pile. This helps to reduce the uncertainty, but it still appears that errors in λ contribute about $10 \times 10^{-6} \text{ cm}^{-2}$ error in B^2 .

The maximum buckling for these large fuel elements is almost as high as for smaller elements, but with the maximum occurring at larger lattice spacings - about $10\frac{1}{2}$ inches for the 1.92-in. element and 12 in. for the 2.5-in. element.

The carbon-to-uranium ratio for equal wet and dry bucklings ("cross-over") decreases with rod size and also with water-to-uranium ratio. The measured crossover lattice spacings for the particular configurations reported here are 8.3 in. for the 1.92-in. elements and 9.5 in. for the 2.5-in. elements.

Further experiments are planned to measure B^2 for selected cases in larger exponentials and k_{∞} in the PCTR to develop a better absolute basis for the bucklings. Previous comparisons to PCTR measurements have indicated that the measured bucklings give high k_{∞} 's by 4 to 5×10^{-3} in k on the average. The bucklings used for this comparison were measured in large exponentials (approx. 8 ft) so λ variations were not a problem. The exponential k_{∞} 's were calculated from

$$k = (1 + L^2 B^2) e^{B^2 \tau}$$

It appears that the standard derivation of L^2 and τ should be modified to provide agreement between exponential and PCTR measurements.

¹D. E. Davenport, Geneva Paper pg. 599 (1955).

²E. D. Clayton, "Buckling Measurements with Enriched Uranium in Graphite Systems," (1957).

³D. J. Donahue, D. D. Lanning, R. A. Bennett, and R.E. Heineman, Nuc. Sci. and Eng., 4, 297 (1958).

Measured Bucklings in Graphite - Uranium Lattices

Lattice Spacing (inches)	Buckling (10 ⁻⁶ /cm ⁺²)	Side-to-side λ (in.)	Volume Ratios		
			Al/U	H ₂ O/U	C/U
Fuel Element: 1.66-in. x 1.1-in. tube inside a 2.5-in. x 2-in. tube					
14 9/16	- 127	2.25	0.487	1.089	68.48
14 9/16	+ 55	1.79	0.487	0	68.48
Fuel Element: 2.5 in. Solid					
8 3/8	- 60	1.55	0.185	0.403	12.67
8 3/8	- 199	1.30	0.185	0	12.67
10 3/8	+ 49	1.41	0.185	0.403	20.31
10 3/8	+ 88	1.59	0.185	0	20.31
12 3/8	+ 28	1.95	0.185	0.403	29.58
12 3/8	+ 108	1.68	0.185	0	29.58
14 9/16	- 29	1.29	0.185	0.403	41.58
14 9/16	+ 98	1.39	0.185	0	41.58
Fuel Element: 1.92 in. Solid					
6 3/16	- 186	1.34	0.264	0.288	11.64
6 3/16	- 400	1.20	0.264	0	11.64
8 3/8	+ 65	1.55	0.264	0.288	22.65
8 3/8	+ 67	1.55	0.264	0	22.65
10 3/8	+ 70	1.56	0.264	0.288	35.60
10 3/8	+ 125	1.50	0.264	0	35.60
12 3/8	+ 34	2.11	0.264	0.288	51.31
12 3/8	+ 102	2.25	0.264	0	51.31

Initial Experiments with the Organic-Moderated Reactor Critical Assembly G. B. Zwetzig, L. S. Beller, H. C. Field, L. A. Mountford, and R. J. Tuttle, (AI). Initial operating experience has been obtained on an organic-moderated reactor critical assembly using slightly enriched uranium fuel at elevated temperatures. The special techniques required for operation and experimentation at high temperatures have been explored, and a start has been made toward evaluation of two proposed core and poison-rod designs.

The system is basically similar to many existing liquid systems. The core vessel accommodates cores up to 5 ft. in diameter by 5 ft. tall and a wide variety of grid structures. A moderator dump system and separate storage is employed. Heating and insulation allow operation up to a temperature of 600°F. Poison rods are cable actuated and can be used in any lattice position.

The main points of difference from other liquid systems are the moderator, its handling system, and the high operating temperatures. The moderator is a commercial mixture of the three terphenyl isomers, with a melting point of 292°F and a boiling point of about 700°F. The critical assembly is operated above 325°F to preclude freezing in the core, while the higher operating temperatures approach those of proposed power reactor designs. The characteristics of the moderator and the operating temperatures involved require experimental techniques and apparatus not usually encountered.

Instrument sensitivity is one such area requiring special consideration. As with other hydrogenous systems, organic moderated reactors have low leakage fluxes, while the operating temperatures require that ion chambers and counters be placed outside the core vessel. Sensitivity is achieved in the startup ranges by a logarithmic count rate and period meter. (Ion current instruments are satisfactory for normal operation). Paraffin moderator-reflectors around detectors also increase sensitivity since roughly two-thirds of the leakage flux is episcadmium. Even so, subcritical multiplication statistics tended to be poor during the initial critical loading. In subsequent multiplication measurements, counters were placed in insulated and forced air cooled thimbles in the radial reflector.

Also, because of the high temperatures and a moderator that is solid at room temperatures, foil activation measurements require special techniques. Indium cannot be used at all, and cadmium, which melts at 608°F, can be used only at the lower temperatures. A manganese-nickel alloy containing a few percent copper is used for most measurements. Gross measurements are made with wire segments 0.030 in. in diameter by $\frac{1}{2}$ in. long inserted in cadmium or aluminum tubes. The tubes are 5 to 6 ft. long and must be sealed carefully to prevent entrance of the organic so that the segments can be recovered. Foils are beta-scintillation counted with automatic decay correction and automatic sample changing equipment.

Since the moderator is a dielectric, conductance probes cannot be used to determine upper reflector level. Instead, a capacitance probe and an FM discriminator system is used. The level can be determined to ± 0.005 in.

Each fuel element for the initial loading consisted of 12 uranium plates of 1.9% nominal enrichment, each measuring 0.1 x 5 x 54 in., covered with aluminum to mock up cladding and cooling fins. Each element contained approximately 100 kg of uranium. Control and safety rods were identical in structure. The poison was boron carbide powder packed in the annulus between two steel cylinders. There were two safety and four control rods operating in vacant lattice positions without thimbles. Thirty-nine fuel elements on a 6-in. square lattice spacing were required for criticality. Control rods were worth \$1.45 each, and safety rods, about \$1.80.

With this loading, a rough estimate of the isothermal temperature coefficient was made. It was -0.21 ± 0.07 μ /F° in the temperature range around 335°F. The temperature coefficient became increasingly negative as fuel was added.

A cylindrical fuel element with an internal concentric control rod was evaluated in this plate-type lattice by danger coefficient methods. The element consisted of two concentric cylinders containing about 80 kg of uranium of the same enrichment as before alloyed with $3\frac{1}{2}\%$ molybdenum and canned in aluminum. The control rod was of the same construction as before, but smaller in diameter. The element and the rod were worth only slightly less than their plate-type counterparts.

Exponential and Reactivity Experiments

Buckling Measurements on Natural Uranium-Heavy Water Lattices at Temperatures up to 215°C, *E. J. Hennelly, T. F. Parkinson, and W. B. Rogers (DuP-SR)*. Exponential buckling measurements on lattices of natural uranium tubes in D₂O were made in the Pressurized Subcritical Experiment (PSE)¹ at the Savannah River Laboratory. The metal fuel tubes were 1.957 in. OD, 1.387 in. ID, and were clad with 0.030 in. of X8001 aluminum alloy. Material bucklings were determined as a function of (a) lattice pitch, (b) moderator purity, and (c) moderator temperature up to 215°C. Figure 1 shows the effects of lattice spacing, expressed in terms of the moderator-to-fuel volume ratio, and of moderator purity. Figure 2 shows variation of the material buckling with temperature for four typical lattices. Variations in the radial buckling with temperature were small. As shown in Figure 1, the measurements of the material bucklings as a function of lattice spacing were in very good agreement with theoretical estimates based on the SRL buckling recipe.^{2,3} The theoretical estimates of the buckling as a function of temperature were normalized to the measurements at 20°C. Table I gives two-group lattice constants calculated for a typical pair of lattices in 99.7% D₂O at 20°C and at 240°C. L^2 and f were determined by P-3 calculations.

Fig. 1. Natural Uranium Metal Tubes in D₂O: Material Buckling vs Moderator-to-Fuel Ratio

Fig. 2. Material Buckling vs Temperature

TABLE I
Two-Group Lattice Constants

Lattice Pitch, (in.)	$V_{\text{mod}} / V_{\text{fuel}}$	Temp. (°C)	f	p^a	$\eta\epsilon$	k_{∞}	τ (cm ²)	L^2 (cm ²)	B^2 (m ⁻²)
7	27.1	20	0.980	0.895	1.363	1.195 ₁	126.1	125.1	7.42
7	27.1	240	0.983	0.868	1.363	1.164 ₂	184.6	190.5	4.20
6	19.8	20	0.983	0.862	1.363	1.155 ₃	128.3	87.8	6.94
6	19.8	240	0.985	0.828	1.363	1.111 ₃	188.0	135.4	3.37

^aThe change in p with temperature was calculated according to the formula

$$\ln p/p_0 = \frac{\sigma_s \rho}{\sigma_{s0} \rho_0} \frac{RI_0}{RI}$$

where σ_s is the scattering cross section, ρ the moderator density, and RI the resonance integral expressed as $RI = RI_0 (1 + 1.43 \times 10^{-4} \Delta T \text{ } ^\circ\text{C})$.

¹E. J. Hennelly, T. F. Parkinson, and W. B. Rogers, Jr., "Pressurized Facility for Subcritical Lattice Measurements at High Temperatures," Trans. Amer. Nucl. Soc. 2-1, 64 (1959).

²N. P. Baumann, "Process Development Pile Measurements of Lattice Parameters of Natural Uranium in Heavy Water," DP-407 (July 1959).

³G. Dessauer, et al., "Physics of Natural Uranium Lattices in Heavy Water," Proc. 2nd Int. Conf. on Peaceful Uses of Atomic Energy, Vol. 12, United Nations, Geneva (1958), p. 320.

Critical Experiments I

Shape-Perturbations in Critical Experiments, W. A. Reardon and R. C. Lloyd (GE-HAPO). When making heterogeneous critical mass measurements, several perturbations of unknown magnitude are usually present, such as the effect of an irregular outer boundary. Critical approach type measurements were made with circular, elliptic, and rectangular cylinders to try to evaluate some of the effects.

The objectives were specifically:

- to determine the effect of the irregular outer boundary on the measured critical mass,
- to obtain information on the effect of geometry on the extrapolation distance,
- to examine the feasibility of more accurately determining the effective radius of the extrapolated critical cylinder by introducing a shape deformation which is then successively decreased and put to zero, and
- to compare the theoretical buckling formulations with experiments.¹

The uranium rods used in these experiments were 23.5 in. long, 0.175 in. in diameter, and enriched to 3.063 percent in U-235. They were encased in 0.025-in. wall Lucite tubes and were arranged in a 0.5-inch triangular lattice; the resulting H₂O/U (total) volume ratio was 8.0. The assemblies were both fully reflected and moderated with H₂O.

The critical masses were calculated using the relation between the buckling and the semi-axis of ellipses given as tables by Gast and Bournia¹, and values of material buckling and reflector savings given by Lloyd, et al.² The results are summarized in Table I.

TABLE I
Measured and Calculated Critical Masses for 3.063% U-235,
0.4445-cm Diameter Rods, 1.27-cm Triangular Lattice,
Elliptic Cylinders, $\lambda = 6.4$ cm.

Eccentricity	Critical Semi-Major and Semi-Minor Axis		No. of 59.69 cm Rods in Critical Cylinder	M _c (meas.) (kg U)	M _c (calc.) (kg U)	B ² (cm ⁻²)
	a(cm)	b(cm)				
0	15.88	15.88	567.3 ± 1.0	99.3 ± 0.2		0.013530
0.212 (0.224) ^a	16.07	15.71	568.2 ± 1.5	99.3 ± 0.3		0.013520
	16.2	15.6			99.47	
0.465 (0.480)	16.92	14.98	570.3 ± 2.5	99.8 ± 0.5		0.013520
					100.56	
0.647 (0.624) (0.74)	18.34	13.99	572.2 ± 1.5	101.1 ± 0.3		0.013550
	19.2	13.6			102.78	
	21.9	12.6			108.59	

^aParentheses denote calculated values.

Rectangular Cylinders ($\lambda = 6.3$ cm)

Critical Dimensions		No. of 59.67-cm Rods in Critical Cylinder	M _c (kg U)	B ² (cm ⁻²)
a(cm)	b(cm)			
28.09	29.22	587.7 ± 0.6	102.9 ± 0.1	0.013440
25.30	33.00	597.7 ± 1.5	104.6 ± 0.3	0.013510

The bucklings shown in the table were evaluated from the assumed reflector saving, λ , the calculated values of the semi-axis of the critical assemblies, and the tables of Reference 1.

From the list of critical masses it can be concluded that appreciable deformations in shape do not introduce large errors into the critical mass determination.

Because of an uncertainty in λ , of the order of 0.2 cm, one concludes that unless the shape deformations are more pronounced than those introduced (i.e., long thin slabs, etc.), the value of λ is not a strong function of shape for these relatively small assemblies.

The method used to calculate the critical mass of the ellipses appears not to be adequate for the larger eccentricities.

¹P. F. Gast and A. Bournia, "Finding the Buckling of an Elliptic Cylinder," *Nucleonics* 14, 109 (April, 1956).

²R. C. Lloyd, E. D. Clayton, R. B. Smith, and V. I. Neeley, "Criticality Measurements of Heterogeneous 3.1% Enriched Uranium and Water Systems," *Nuclear Physics Research Quarterly Report*, HW-63576 (Jan. 20, 1960).

High Temperature Graphite Critical Experiments*, E. Goldberg, R. Finke, and H. L. Reynolds (UCRL-L). The high temperature critical assembly facility at Nevada Test Site¹ allows the performance of experiments at temperatures ranging from room temperature to 1200°F. The present paper will discuss a series of critical experiments involving a graphite moderator with U-235 as fuel, at various temperatures up to 1190°F.

The facility is essentially an oven, 20 ft x 16 ft x 12 ft high, connected by ducting to an external heater. Circulating nitrogen conveys heat to the oven and raises the oven contents to temperatures up to 1200°F. A low-mass table within the oven is designed to accommodate critical assemblies of various dimensions, but not exceeding 10 ft x 10 ft x 8 ft high. The oven walls are of low areal density.

The assemblies were constructed by stacking graphite blocks, 12 in. x 12 in. in cross-section and either 1.0 or 2.0 in. thick. The average core moderator density was 1.46 g/cc. Pulsed neutron measurements established the moderator poison level to be equivalent to 5 ppm of boron.

The fuel foils were 0.002-in. thick or alloy (uranium enriched to 93.2% in U-235), hermetically sealed in 0.002-in. thick type 347 stainless steel. Each foil contained, on the average, 123 g of or alloy and 105 g of steel. The overall dimensions were 23 5/8 in. x 9 1/4 in. x 0.006 in.

Neutronic control was achieved with a vane system composed of four vanes. These were inserted from above in the vertical median planes of the system. The vanes were accommodated in the assembly in gaps 1/2 in. x 10 in. running the full height of the core.

The experiments involved a bare near-thermal reactor. The temperature effect was determined in two ways. In one, the dimensions were maintained constant while the fuel loading was varied to achieve criticality at various temperatures whereas in the other, the molar ratio, (C/U-235) was maintained constant while the buckling was allowed to change.

The assembly dimensions at room temperature for the first set of experiments were 72 in. x 72 in. x 61.5 in. high. The latter value should be 62.5 in. since fission traverses indicated a one-inch perturbation from the low-mass table. It was not possible to distribute the fuel elements uniformly throughout the system; the procedure adopted was to fully insert half of the foils into their three foot long slots while inserting the other half 24 in. The effective loading was then determined by weighting each foil by ϕ^2 according to its position within the bare assembly. An IBM-709 code was developed for this task. Spacing between fuel layers was approximately five inches.

The results of the first series are given in Table I, where the second column is essentially the raw data. The third column has been corrected for assembly thermal expansion, the low-mass-table effect, and has been reduced to a uniformly-loaded system. The latter correction was slight (~3% effect on fuel loading) and was determined by multigroup calculations. The data of the third column correspond to a buckling, B^2 , of $(9.35 \pm 0.03) \times 10^{-4} \text{ cm}^{-2}$ with the moderator density of 1.46 g/cc. The extrapolation length was 0.84 in. The C/U-235 molar ratio for the room temperature critical assembly was 8950.

TABLE I
Results of High Temperature Experiments

T (°K)	M _c (kg of U-235)	M _c (corr.) (kg of U-235)
914 ± 5	23.1	22.1
675 ± 3	21.0	20.3
511 ± 2	19.3	18.8
422 ± 2	18.2	17.8
292 ± 2	17.2	16.9

The tabulated data are not corrected for fuel self-shielding and flux depression. Experiments performed at Livermore indicate a disadvantage factor of 0.79 ± 0.07 for the room temperature experiment.

The second approach involved the determination of critical buckling for a given C/U-235 molar ratio at various temperatures. This experiment was more carefully performed, with special attention given to constant spacing of foils. The spacing was five inches. The bucklings were altered slightly according to multigroup calculations to correspond to the case of uniform loading. The molar ratio throughout the assembly, uncorrected for self-shielding and flux depression, was C/U-235 = 8700. When adjusted to a common moderator density of 1.46 g/cc, the critical bucklings were: $(9.58 \pm 0.03) \times 10^{-4} \text{ cm}^{-2}$ at $283 \pm 2^\circ\text{K}$ and $(8.41 \pm 0.03) \times 10^{-4} \text{ cm}^{-2}$ at $744 \pm 5^\circ\text{K}$. The cold critical assembly was loaded with 1.5% less fuel than the previously described system of the same dimensions.

¹ H. L. Reynolds and C. E. Walter, "HOT BOX-A High Temperature Critical Facility," UCRL-5483 (1959).

Reactivity Measurements on Unperturbed, Slightly Enriched Uranium Dioxide Lattices, A. L. MacKinney and R. M. Ball (B & W). Uniform lattice critical experiments with 3 and 4% enriched uranium dioxide, clad with stainless steel, have been performed in ordinary water. These experiments were done with fuel pins, 0.500 in. OD with an active fuel length of 66.5 in. The fuel pins were fabricated by swaging the UO_2 powders to 87% theoretical density, in stainless steel tubing. The final wall thickness was 0.0282 in.

Table I lists the pitch spacings which were used and the volume fractions of materials present in the core. Figure 1 displays the minimum critical radius obtained for the fully reflected cores as a function of the ratio of water to equivalent uranium.

Measurements of the differential reactivity as a function of water height for each lattice yielded the common straight lines of $(\text{differential reactivity})^{-1/3}$ as a function of water height. Numerical integration of the differential reactivity curve yields the excess reactivity as a function of core radius, shown in Figures 2 and 3.

Fig. 1. Minimum Core Radius as a Function of Water to Uranium Ratio

Fig. 2. Excess Reactivity as a Function of Core Radius (3% Cores)

Fig. 3. Excess Reactivity as a Function of Core Radius (4% Cores).

TABLE I
Volume Fractions in Unperturbed Lattices

Square Pitch		Volume Fractions		
in.	cm	Water	Stainless Steel	UO ₂ (9.45 g/cc)
0.612	1.555	0.4758	0.1109	0.4133
0.663	1.684	0.5533	0.0945	0.3522
0.865	2.198	0.7378	0.0555	0.2067
0.975	2.381	0.7934	0.0437	0.1629

Neutron Age and Nonleakage Probability for a 2% U-235 Enriched UF₄-Paraffin Mixture With a H:U-235 Ratio of 404, J. T. Mihalcz (ORNL). The neutron age and nonleakage probability for a mixture of 2% U-235 enriched UF₄ and paraffin with a H:U-235 atomic ratio of 404 have been determined from critical experiments at the Oak Ridge National Laboratory Critical Facility. These experiments are a continuation of a program reported previously.¹ The material is a homogeneous mixture which contains 85 w/o UF₄ and 15 w/o paraffin and has a density of 3.45 g/cc.

The analysis of the experiments was based on the critical equation

$$k_{\infty} P(B) = 1, \quad (1)$$

the static relation

$$k_{\text{eff}} = (1 - \beta) k_{\infty} K_p(B) + \beta \frac{k_{\infty}}{\epsilon} K_d(B), \quad (2)$$

and the kinetic relation

$$\Delta k = \frac{\ell}{T} + \frac{k_{\infty}}{\epsilon} K_d(B) \sum_i \frac{\beta_i}{1 + \lambda_i T}, \quad (3)$$

where

$P(B)$ = nonleakage probability for all neutrons from fission,

$K_p(B)$ = nonleakage probability for prompt neutrons from fission, and

$K_d(B)$ = nonleakage probability for delayed neutrons.²

For $T \gg 100$ sec the term ℓ/T in equation (3) can be ignored.

For an incremental change in the buckling, a differential form of equation (2) is obtained. Combining this differential form with equation (3), keeping only terms to first order in ΔB^2 , yields

$$h + 2\delta \cong A \left(\frac{\Delta k}{\Delta h} \right)^{-1/3}, \quad (4)$$

where ΔB^2 has been written in terms of the height, h , and the extrapolation distance, δ . The inverse cube root of the measured $\Delta k/\Delta h$ values as a function of height was least-squares fitted to the equation of a straight line to determine the constant, A , which depends on the model assumed for the nonleakage probability. For a two-group model, A involves the buckling, fast-fission factor, delayed-neutron fraction, delayed-neutron age to thermal, thermal-diffusion area, and prompt-neutron age to thermal. The prompt-neutron age is obtained once the other quantities in the expression, A , are known.

The fast-fission factor was determined from the ratio of the activities of a depleted and an enriched uranium foil, the resulting value being 1.0214 ± 0.0021 . The buckling was obtained from the critical size of the assemblies and the extrapolation distance. The latter, which was determined by least-squares fitting the flux distribution, was found to be 2.64 ± 0.3 cm. The average buckling was 6.642×10^{-3} cm⁻². The thermal-diffusion area, calculated from thermal cross sections, was 1.87 cm². The delayed-neutron age to thermal was calculated by J. G. Sullivan using a Monte Carlo moments-method calculation. The value was 15.6 cm².

The resulting value for the prompt-neutron age to thermal was determined to be 28.2 ± 1.0 cm² and that for the nonleakage probability, $P(B)$, to be 0.8362 ± 0.0048 . The reciprocal of $P(B)$, from equation (1), is k_{∞} and is equal to 1.196. The results of the critical experiments are given in Table I.

TABLE I
Properties of Unreflected 2% U-235-Enriched UF₄-Paraffin Assemblies
with a H:U-235 Ratio of 404

Base Dimensions (cm)	Critical Height (cm)	Critical Mass (kg of U-235)	B^2 (cm ⁻²)	$\Delta k/\Delta h$ (cm ⁻¹)
61.2 x 46.0	144.5	18.0	6.429×10^{-3}	1.51×10^{-4}
51.1 x 56.1	110.5	14.0	6.461×10^{-3}	3.23×10^{-4}
51.1 x 61.2	88.3	12.2	6.468×10^{-3}	6.37×10^{-4}
61.2 x 61.2	64.8	10.7	6.471×10^{-3}	1.55×10^{-3}
61.2 x 71.6	56.7	11.0	6.482×10^{-3}	2.30×10^{-3}
Average			6.462×10^{-3}	

¹J. T. Mihalczo, "Determination of Infinite Medium Multiplication Factor for 2% Uranium-235 Enriched UF₄ in Paraffin," Trans. Amer. Nucl. Soc. 2-2, 122 (1959).

²G. R. Keepin, T. F. Wimett, and R. K. Zeigler, "Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium, and Thorium," Phys. Rev. 107, 1044 (1957).

Experimental Lattice Parameters for Heavy Water-Natural Uranium Lattices, T. J. Hurley, Jr. and G. F. O'Neill (DuP-SR). The Process Development Pile (PDP) at the Savannah River Laboratory is being used to obtain an extensive series of precise data for D₂O-moderated lattices of natural uranium metal. The large size (16 ft diam., 15 ft high) of the PDP tank practically eliminates edge effects even at very wide lattice pitches and therefore provides data of unusually high accuracy for detailed tests of various calculational methods.

The measurements were made in 99.75 mol % D₂O with clusters of natural uranium metal rods. The rods were one inch in diameter and were used in one, three, seven, and nineteen rod clusters at lattice spacings ranging from five to twenty-one inches. Approximately 20 lattices were studied in unreflected criticals. The lattice bucklings ranged from 200 to 700 μ B and were determined to about $\pm 2\mu$ B. A few of these lattices were also studied with a D₂O reflector and at a moderator isotopic purity of 99.3%. A comparison of calculated¹ and experimental bucklings for seven rod fuel clusters is given in Figure 1.

In addition to the buckling measurements, foil irradiations were used to determine the lattice parameters η , f , ϵ , p , and L^2 . Particular attention was paid to the resonance flux distributions as obtained from In, Au, Co, W, and U-238 foil irradiations. Calculations of the effectiveness of the internal surface of the fuel in absorbing resonance neutrons agreed, within $\pm 5\%$, with activation experiments using U-238 foils. However, in some cases the observed distributions of the resonance flux through the fuel assemblies showed an unexpected peaking inside the assemblies.

Fig. 1. Lattice Buckling for Seven Rod Fuel Clusters of Natural Uranium.

¹N. P. Baumann, "Process Development Pile Measurements of Lattice Parameters of Natural Uranium in Heavy Water," DP-407 (1959).

An Accurate Reactivity Measurement Technique for Critical Facilities, L. S. Beller (AI). A rapid period-measurement and data-reduction system, capable of routine relative accuracies of the order of 0.1 to 0.2% in reactivity, has been developed for the Organic Moderated Reactor Critical Experiment.^{1,2} Not only small changes in reactivity associated with danger coefficient experiments, but the large changes involved in control rod evaluations can be measured with this accuracy.

The basic instrument is a neutron-sensitive pulse-counting system with automatic readout. Differences from similar systems used elsewhere^{3,4} appear to lie in data-reduction technique and in optimization of experimental conditions.

A computer program does the data reduction. The program first makes a correction for paired-pulse resolving time, then computes the period and its standard deviation by a weighted-least-squares method, computes the corresponding reactivity and its standard deviation, and finally tests the results for goodness of fit. This procedure is designed to show whether or not the observed errors are randomly distributed.

This test can be used to eliminate the two most important sources of systematic error in the experiment. When the reactivity is changed, transient effects dominate for some time while the instantaneous period approaches an asymptotic value. Since the data are assumed to fit a true exponential, the effect of the transient shows up as a systematic error, easily detected by the goodness-of-fit tests. An experimental plot, similar to the calculated curves of Toppel,⁵ can then be constructed showing the necessary wait time for an arbitrary acceptable error as a function of the period for the given reactor and experimental conditions.

The second most important source of systematic error is finite paired-pulse resolution. Again, it is assumed that the data represent a true exponential. The observed systematic errors, after the wait time errors have been eliminated, then arise from resolution losses. The calculation is performed several times while varying the parameter resolving time until a minimum error is obtained.

The standard deviation of the answer does not depend strongly on the statistics of the individual counting periods. In practice, however, the best results are obtained when the individual counting periods are sufficiently long or are at a high enough count rate to have "good statistics," of the order of 1% or better, even at the expense of fewer counting periods if necessary.

During control-rod calibrations, reactivity increments are determined from differences in positive reactivity between two period measurements, rather than between one positive reactivity measurement and the critical condition (zero reactivity).

With this method, control rod calibrations have a relative uncertainty of about 0.3% in total integral worth and less than 0.06 cents/in. on differential curves. (The uncertainty in digital reading of control rod position contributes slightly more than half of this uncertainty.) This accuracy allows quantitative measurement of small changes in control rod calibrations. In one instance, it was possible to measure a temperature coefficient of -0.02% of control rod worth/°F (a total change of about -1.5 cents) in the presence of a change in interaction with other rods that was positive in sign and had a value of roughly 30 cents.

Similarly, it is possible to measure interaction changes as a function of position of an adjoining control rod.

The isothermal temperature coefficient of the OMR-CA was expected to be a quadratic function of temperature. When reactivities were calculated with this method, the quadratic behavior was found and the constants measured to good accuracy. The second-order term was an order of magnitude smaller than the linear term; when the same data were reduced by other methods, this coefficient could not be determined.

¹G. B. Zwetzig, "Organic Moderated Reactor Critical Experiment Hazards Summary," NAA-SR-3220 (Dec. 15, 1958).

²G. B. Zwetzig et al., "Initial Experiments with the Organic Moderated Reactor Critical Assembly," Trans. Amer. Nucl. Soc. 2-2, 85 (1959).

³A. L. MacKinney and R. M. Ball, "Inexpensive, Automatic System for Recording Reactor Data," Nucleonics 17, 128 (May 1959).

⁴H. A. Morewitz, Atomics International, Canoga Park, Calif., private communication.

⁵B. J. Toppel, "Sources of Error in Reactivity Determinations by Means of Asymptotic Period Measurements," Nucl. Sci. and Eng. 5, 88 (1959).

Critical Experiments II

Flexible Critical Experiments in the Army Reactor Branch Stationary Medium Power Reactor Series, W. J. McCool, R. A. Robinson, E. W. Schrader, and S. W. Weiss (ALCO). Flexible critical experiments were performed in continuation of the Army Reactor Branch Development of the stationary medium (SM-) power reactor series. The first reactor of this series, the SM-1 located at Ft. Belvoir, Va. and formerly known as the APPR-1, has a thermal rating of 10 megawatts and a core life expectancy of approximately 15 megawatt years. The intent in the SM-2 design is to produce a reactor with a thermal rating of approximately 28 megawatts and a life expectancy of 25 megawatt years with the same number and overall fuel element dimension as the SM-1. The SM-1 and SM-2 stationary fuel elements consist of 18 parallel fuel plates brazed to inert stainless steel side plates on two sides to form an active fuel cell with the approximate dimension of 3 in. x 3 in. x 22 in. The individual fuel plates are composed of 26 w/o uranium as the oxide held in a stainless steel matrix and clad with 0.005 in. thick stainless steel. The fuel matrices for the SM-1 and SM-2 are 0.020 and 0.030 in. thick to render a total fuel plate thickness of 0.030 and 0.040 in. respectively, including cladding.

The SM-1 and 2 control rods are of the absorber-fuel element follower type. The absorber is a hollow stainless steel-boron carbide box while the control rod fuel element is composed of 16 parallel plates similar in composition to the stationary elements. These two components are contained in a stainless steel basket which is driven into the core for control purposes.

Both the SM-1 and SM-2 core arrays consist of 38 stationary fuel elements plus 7 control rod assemblies arranged in a 7x7 fuel element lattice with the four corner elements removed. With the thicker SM-2 fuel plates and a consequent increase in the metal to water ratio, and the higher SM-2 fuel loading and thermal rating a flexible critical experiment was deemed essential to sound design and safe operation.

Parametric studies were performed in which core excess reactivity was related to uranium-235 and boron-10 loading and metal to water ratio. The merits of two different control arrays were studied and critical stuck rod configurations established for start-up and mid-life cores. Power distribution in the core was measured. A presentation of typical SM-2 start-up core data follows:

Core Composition	
Uranium-235	36.4 kg
Boron-10 (as burnable poison)	60.9 g
Metal to Water Ratio	0.337
Reactivity Worth	
Uranium-235 (average worth)	+0.154 cents/g
Boron-10 (average worth as burnable poison)	-42.5 cents/g
Control Rods (from cold clean critical to fully withdrawn position)	-1520 cents
Cold to hot change (103 to 510° F at 2000 psi)	-890 cents
Flow divider for two pass core	-96 cents
8 in. steel-water laminated reflector	+85 cents
Critical Position, Seven Rod Bank Withdrawal	
Cold clean, with reflector, no flow divider	7.14 in.

An SM-1 core mockup was assembled from SM-2 fuel plates and an estimate of the SM-1 core I boron loading obtained from substitution measurements utilizing a fuel assembly of known composition and the known SM-1 cold clean critical bank position. The estimated boron loss of 22% from the specified SM-1 Core I loading was in good agreement with chemical analysis.

An Enriched UO_2 - ZrH_2 Critical Assembly, M. V. Davis and A. W. Thiele (AI). A critical assembly has been built to obtain basic nuclear parameters necessary for the design of compact reactors for the SNAP Program. The assembly consists of two hemispheres, separated on a horizontal plane. A machine was constructed that provided rigid support to the upper hemisphere and lifted the lower hemisphere at rates that would result in an approximately constant reactivity insertion rate. A portion of the reflector on the lower hemisphere was movable to provide fine adjustments in the reactivity.

The fuel moderator material was prepared by blending 8 w/o UO_2 (93.2% U-235 in U) in ZrH_2 powder and compressing to form small blocks. Identical sets of blocks with and without UO_2 were prepared. Initially, the assembly was loaded with blocks of ZrH_2 only and was fueled by substituting UO_2 - ZrH_2 blocks maintaining a spherical loading. The core is reflected by a beryllium and a graphite layer around the core.

The results of the four critical configurations are given in Table I.

TABLE I
Pertinent Quantities for Four Different Critical Assemblies

Assembly	Outer Radius of Region (cm)				Mass of U-235 for Criticality (g)
	Region 1	Region 2	Region 3	Region 4	
	UO_2 - ZrH_2	ZrH_2	Be	Graphite	
1	12.28	13.87	17.15	25.4	2070
2	12.51	13.87	17.15	23.5	2222
3	13.06	13.87	17.15	21.6	2540
4	13.81	13.87	17.15	19.7	3000

The isothermal temperature coefficient of assembly 4 was measured by externally heating and cooling the assembly while adjusting the reactivity to maintain constant neutron flux. The computed best fit to the data is as follows:

$$\frac{dk}{kdt} = -(2.47 \pm 0.03) 10^{-5} / ^\circ\text{C} \quad (1)$$

The prompt neutron lifetime of the fourth assembly was measured by the Rossi α method and the noise analysis method. The results were in good agreement, giving $\ell = 14$ microseconds for a β of 0.0085.

Because of the high leakage of neutrons from the system (~40% of the neutrons are lost), the reactor is sensitive to the addition of reflecting material. A measure of this sensitivity was determined by bringing samples of reflecting material from a distance of approximately 1.5 meters up to the surface of the reactor. The measured change in reactivity bears an R^{-4} dependence on position (R the distance from the reactor center to the sample), implying angular isotropy in the neutron leakage from the reactor and in the back scattering of the neutrons from the sample of reflecting material.

The reactivity associated with different materials external to the core can be correlated by a formula that relates the reactivity to a linear product of the macroscopic scattering cross section and the energy change of the neutrons on reflection.

The effects of neutron scattering and neutron absorption within the reactor core was investigated by reactivity measurements. The samples were positioned in a sample changer so constructed that measurements could be conducted at the center of the reactor or at specific positions out through the reactor core. The spatial effects of reactivity for materials with varying ratios of scattering to absorption cross section were measured. The reactivity data are combined with U-235 foil traverses to determine the extrapolated radius of the reactor and to experimentally measure the quantity used in core calculations as the adjoint flux.

The information obtained in these and other experiments had led to the design and construction of the SNAP Experimental Reactor which has been operated at 50 kw and 1200°F outlet temperature.¹

¹ John A. McCone, Speech before American Rocket Society Meeting, Washington, D. C. (Nov. 19, 1959)

Critical Experiments in Slab Geometries, R. A. Hartfield, H. F. Finn, R. J. Neuhold, R. H. Vollmer (A-C) and G. H. Neils (NSPC). The Allis-Chalmers Critical Facility is presently being used to establish the first core design parameters of the Pathfinder Nuclear Power Plant being built for the Northern States Power Co. Pathfinder is a boiling water reactor with a centrally located integral nuclear superheater, designed to deliver steam at 540 psig and 825°F and rated at 66 MWe. The first series of experiments were performed on boiler lattices only, utilizing slab geometries that were built up on the core periphery. These slab experiments have led into the full scale mockups and extensive critical experiments required by the integral superheater and axial variation of fuel loading in Pathfinder.

The Boiler fuel rods used in the slab experiments are UO_2 , 0.388 in. OD and 17 in. long completely jacketed in a 0.040 in. cladding of X8001 aluminum (nominal composition 1% Ni, 1/2% Fe, remainder Al). Clips are used to obtain lattice spacings within bundles of fuel rods 5 in. square and 17 in. long. There is a 1 in. water gap between bundles so the total effective fuel length is about 6 feet. Four such bundles make up one fuel element. Lattices with water to uranium volume ratios of 4.4 and 6.0 were assembled into a fuel element box which is 5.2 in. square. The average fuel enrichment for a full core loading can be varied from 1.3% to 1.9% U-235, but only two enrichments, 1.8% and 1.59% were investigated during initial slab experiments.

The clean critical size, reflector savings, and temperature coefficients were measured for two 1.8% enriched cores with water to uranium volume ratios of 4.4 and 6.0 (excluding control rod water channels and axial water gaps). In addition the clean critical size was obtained for the 1.59% enriched core. The clean critical size and reflector savings were also measured for the 1.8% cores containing 18% and 14% voids (ratio of void to water plus voids), respectively. The voids were obtained by inserting low density (9 lb/ft³) foamed polyurethane void simulators between fuel rows to obtain a uniform displacement of water. These simulators are coated to reduce water absorption to a negligible value. The measurements are summarized in Table I.

TABLE I
Summary of Results on Slab Experiments

Enrichment (% U-235)	W/U	Core Loading (kg U-235)	Size (in.)	Excess (%Δk/k)	Void Content (%)	Temp. Coeff. (Δk/°F at 70°F) ($\times 10^3$)	Temp. Coeff. (Δk/°F at 160°F) ($\times 10^3$)
1.8	4.4	18.4	18 x 27.5 x 72	0.15	0	-0.6 ± 0.2	-4.4 ± 0.2
1.8	6.0	25.7	21.5 x 34.2 x 72	0.06	0	+0.1 ± 0.2	-0.6 ± 0.2
1.8	6.0	33.5	21.5 x 44 x 72	0.02	14	_____	_____
1.8	4.4	33.0	21.5 x 34.8 x 72	0.25	18	_____	_____
1.59	4.4	37.4	21.5 x 44 x 72	0.16	0	_____	_____

All the cores contained three 0.25 in. thick cruciform control rods. Two 2% boron in stainless steel safety rods were used along with one stainless steel regulating rod. The control rod water channels are 0.5 in. wide with a span of 10 in. The regulating rod was near the full out position during the measurements.

Experimental Nuclear Properties: Shielding

Resonance Absorption of Neutrons in Thick Sheets of Molybdenum, E. Goldberg and F. A. Kloverstrom (UCRL-L). Several experiments have been performed to determine the effective resonance absorption integral for molybdenum as a function of the thickness of molybdenum sheets. The results have been compared to theoretical predictions to appraise the contribution from unresolved resonances. The latter is significant when one is involved with structural components in a reactor that are quite thick such that lower energy resonances are strongly self-shielded whereas the resonances in the kev range and beyond are not. The resonance integrals were determined by placing molybdenum sheets in a critical assembly and finding the critical dimensions. A multi-group diffusion code was then employed to determine the cross sections which gave the correct critical dimensions.

The SPADE facility of LRL-Livermore was employed.¹ The core of the critical assembly was BeO moderated ($\rho = 2.86$ g/cc), and the fuel was enriched (i.e., 93.2%) U-235. The molar ratio was BeO:U-235 = 247:1. The foil thickness and spacing was eight mils per inch. The cross section was 24 in. x 24 in. for both the core and reflectors. The reflectors were of graphite ($\rho = 1.70$ g/cc), twelve inches thick, and were in contact with the top and bottom surfaces of the core. Control rods were not used. The lower portion of the assembly rested on a movable low mass pedestal while the upper rested on a steel diaphragm. Suitable corrections were made for the presence of the diaphragm in determining the critical height of the core.

The sheets of molybdenum of high purity, 24 in. x 24 in. in extent, were placed either at the mid-plane of the core or at the core-reflector interfaces. The results are given in

TABLE I

The Effective Resonance Absorption Integral for Molybdenum as a Function of Thickness

Material	Thickness (in.)	Critical Height, ^a H_c (in.)	$I(\text{Effective})$ (10^{-24} cm ²)
None	---	14.35 ± 0.2	---
Molybdenum	0.120 ^b	16.35 ± 0.2	7.75 ± 0.75
Molybdenum	0.080 ^b	15.8 ± 0.2	8.75 ± 1.25
Molybdenum	0.040 ^b	15.2 ± 0.2	10.8 ± 2.6
Molybdenum	0.040 ^c	15.0 ± 0.2	17.0 ± 5.2
Gold	0.0099 ^b	17.4 ± 0.2	---

^aCritical height of core

^bSheet at core mid-plane

^cSheet at core-reflector interface

Table I. Also included is the finding from an experiment involving a gold sheet, which was done for reasons of calibration.

Inasmuch as the assemblies were one-dimensional in nature, the IBM 709 multigroup diffusion code, ZOOM, was employed to determine the effective resonance absorption integrals for molybdenum. Conventional self-shielding factors were used.² The spacing of resolved resonances^{3,4} are such that it appears justified to assume that flux depression in the moderator does not influence resonance absorption. The resultant resonance integrals are given in Table I.

A similar approach using the results of the gold experiment, and the published resonance integral of 1558 barns⁵ yielded excellent agreement of experiment with the standard value, to within the experimental error. It was necessary here to treat the $1/v$ absorption carefully, taking self-shielding into account. This latter issue was seen to be unimportant for the molybdenum calculations.

Recent reactivity measurements from Harwell⁶ are found to be in satisfactory agreement with the present data. The estimated resonance absorption integral for zero-thickness sheets is 22.7 ± 2.5 barns based on the combined data. Of this, approximately four barns is due to unresolved resonances above 700 ev. The latter estimate was found by comparing the data to predictions based on known resonance parameters.³

¹F. A. Kloverstrom et al., "Critical Measurements on Near-Homogeneous BeO - Moderated Oralloy - Fueled Systems," UCRL-5369, Pt. 1 (1959).

²G. M. Roe, "The Absorption of Neutrons in Doppler Broadened Resonances," KAPL-1241 (1954).

³D. J. Hughes and R. B. Schwartz, "Neutron Cross Sections," BNL-325, 2nd Edition (1958).

⁴J. A. Harvey et al., "Spacings and Neutron Widths of Nuclear Energy Levels," Phys. Rev. 99, 10 (1955).

⁵"Reactor Physics Constants," ANL-5800, p. 83 (1958).

⁶R. B. Tattersall et al., "Pile Oscillator Measurements of Resonance Absorption Integrals," AERE-R 2887 (1959).

Criticality Hazards

A Review of Criticality Accidents, W. R. Stratton (LASL). On at least sixteen occasions a power excursion has resulted from an accidental increase in the reactivity of a fissile system to a value exceeding prompt critical. On two other occasions a prompt power excursion was planned but the total energy release was significantly larger than expected. The systems in which these excursions occurred were of various types: seven were water moderated solutions of uranium or plutonium, five were reflected plutonium metal or bare enriched uranium metal, four were inhomogeneous systems in water, and two must be classified as miscellaneous. The effective neutron lifetimes in these systems ranged between 10^{-4} second and 7×10^{-8} second. The quenching mechanisms responsible for terminating these excursions include thermal expansion, boiling of water, bubble formation from fission fragments, and rise of neutron temperature, and some few were terminated by safety devices or mechanical motion of some part of the system.

The power history of a system whose reactivity has been forced over prompt critical can be divided conveniently into two regions of time. The first of these is characterized by a rapid exponential rise and fall in power, commonly called the "spike." Following the spike the system usually adjusts itself to a relatively steady power which is sustained by delayed neutrons from precursors created during the spike and by whatever prompt multiplication exists in the system at this time. This portion of the power excursion, often referred to as the plateau, slowly decreases with time until the system is scrammed or in some manner becomes subcritical. In some accidental excursions the major portion of the fissions must have occurred in the region of the plateau.

The simplest case analytically is that of the so-called energy model in which the excess reactivity is removed at a rate proportional to the fission energy generated. For this model and for a step increase in reactivity sufficiently large that delayed neutrons may be ignored, the energy release in the spike is given by

$$E = \frac{2\Delta k}{b} \quad (1)$$

and the maximum power is given by

$$\left(\frac{dE}{dt}\right)_{\max} \approx \frac{2}{3.5} \frac{(\Delta k)^2}{b\lambda} \quad (2)$$

where Δk is the excess reactivity over prompt critical, λ is the neutron lifetime, and b is the quenching coefficient expressed in appropriate units. This simple model describes qualitatively the characteristics of most excursions and accurately several accidental excursions, and one can see in equation (2) the enormous difference in behavior as one goes from thermal systems to fast neutron systems. Nonlinear quenching mechanisms not so simply related to fission energy release such as fission bubble formation and boiling of water require more detailed examination and usually digital computing aids are required for extensive studies.

The range of yields observed in these eighteen accidental excursions is from 3×10^{15} fissions to 4×10^{19} fissions leading to a ratio of 1.3×10^4 . The ratio of spike yields is larger, about 2.2×10^5 . The distribution in numbers of accidents as a function of energy range peaks between 10^{15} and 10^{17} fissions with the end points noted above, but no real upper or lower limit in yield exists.

Interaction in Reflected Arrays, H. K. Clark (DuP-SR). A simple technique for rapidly computing the interaction between fissile units or between a fissile unit and a reflector has been previously described,¹ and is now extended to the general case of reflected arrays such as are often encountered in nuclear safety problems. The technique is illustrated by calculating the critical spacing of 20-kg enriched uranium spheres in a $2 \times 2 \times 2$ array enclosed by a concentric hydrogenous reflector as a function of the effective radius of the reflector.

The fraction, ρ , of the neutrons emitted through the surface of one unit that enters the surface of a neighboring unit is approximated by

$$\rho = \frac{\iint \frac{\cos \psi \cos \psi' dA dA'}{\pi R^2}}{\int dA} \quad (1)$$

where dA and dA' are elements of surface on the two units, R is the separation between these elements, and ψ and ψ' are the angles R makes with normals to the elements.

Because this integral is difficult to evaluate for curved surfaces, interacting spheres and circular cylinders are approximated by cubes and square cylinders having the same volume and the same center-to-center separation. In a group of interacting units the effective albedo of the surroundings is expressed in terms of the fractions ρ . The albedos are equated to the ratios β of the currents entering the surfaces to the currents emitted from the surfaces. For a fissile sphere

$$\beta = \frac{\frac{\pi - B\bar{S}}{1 + (\pi - B\bar{S}) \cot BS} - \frac{\pi - BS_0}{1 + (\pi - BS_0) \cot BS_0}}{\frac{\pi - BS}{1 + (\pi - BS) \cot BS} + \frac{\pi - BS_0}{1 + (\pi - BS_0) \cot BS_0}} \quad (2)$$

where $\bar{S}$ is the effective reflector saving of the surroundings, S_0 is the bare extrapolation distance for an isolated unit, and B^2 is the buckling.

For a reflected array of identical units β is found as the appropriate root of the equation

$$\begin{vmatrix} \rho_{11} + \frac{\rho_{1r} \rho_{r1} \beta_r}{1 - \beta_r \rho_{rr}} - \beta & \dots & \rho_{1n} + \frac{\rho_{1r} \rho_{rn} \beta_r}{1 - \beta_r \rho_{rr}} \\ \dots & \dots & \dots \\ \rho_{n1} + \frac{\rho_{nr} \rho_{r1} \beta_r}{1 - \beta_r \rho_{rr}} & \dots & \rho_{nn} + \frac{\rho_{nr} \rho_{rn} \beta_r}{1 - \beta_r \rho_{rr}} \end{vmatrix} = 0 \quad (3)$$

where r denotes reflector, β_r is the albedo of the reflector, and ρ_{ij} is the average fraction of the neutrons from units in symmetrically equivalent positions j that reaches a unit in position i multiplied by the number of units in position j .

For a $2 \times 2 \times 2$ cubic array the equation reduces to

$$\beta = \rho_{11} + \frac{\rho_{1r} \rho_{r1} \beta_r}{1 - \beta_r \rho_{rr}} \quad (4)$$

since all positions are equivalent. Solution of this equation for an array of eight 20-kg oralloy (uranium enriched to 93.5% U-235) spheres gives the results shown in Table I when the reflector is H_2O .

TABLE I
Calculated Critical Spacings of $2 \times 2 \times 2$ Array of 20-kg Oralloy Spheres as a Function of H_2O Reflector Dimensions

Dimensions of Reflecting Cube, ft	Critical Pitch, in.
1.38 x 1.38 x 1.38	8.3
2 x 2 x 2	6.4
$\infty \times \infty \times \infty$	5.5

¹H. K. Clark, "Interaction of Subcritical Components," Trans. Amer. Nucl. Soc. 1-1, 97 (1958).

Neutron Multiplication Measurements with Plutonium-Aluminum Alloy Rods in Light Water, V. I. Neeley, R. C. Lloyd, and E. D. Clayton (GE-HAPO). A series of near critical and exponential experiments have been performed with natural water moderated assemblies of Pu-Al alloy rods in order to establish bases for nuclear safety specifications for handling and storage of this kind of fissile material.

The fuel elements were Pu-Al alloy rods of 5 w/o Pu content. These rods were 24 in. long and 0.506 in. in diameter. They were clad in 0.030-in. Zircaloy-2 with 0.020 and 0.125-in. thick end-caps corresponding to a Zr/Pu atomic ratio of 31.92 and an Al/Pu atomic ratio of 168.20. Each rod contained about 11 g of Pu. The plutonium contained 5 w/o Pu-240.

The experiments were carried out in a 4-ft diam. by 5-ft deep tank of light water. The lattices were hexagonal in shape and placed so as to provide effectively infinite water reflection on all sides. The lattice plates were constructed from lucite. Safety circuits, neutron sources, and BF_3 counters for monitoring neutron multiplication were adapted from previous experiments of this type on 3.1% enriched uranium-water systems.¹

Experiments were conducted using critical approach, "backoff," and exponential measurement techniques. An external source was used during the critical approach and removed during the "backoff" measurements since the neutron emission from (α, n) and spontaneous fission reactions at high multiplication provided sufficient flux for the measurements. The critical approach was carried to 96% of the critical mass as determined by inverse multiplication plots. The critical number of rods, N_c , was obtained by a least squares fit to the $\sqrt{N/\phi}$ -vs- N plot in the region 85-96% of the critical mass for these multiplication measurements. Enough fuel was then removed to make the exponential measurements. A larger source (2.0 g Ra-Be) was used for this purpose.

Fig. 1. Values of Critical Mass and of Buckling Measurements for Pu-Al Alloy Rods in Light Water

Figure 1 shows the experimental results of the multiplication and buckling measurements in addition to the critical mass for spherical geometry.

The extrapolation length was determined by equating the buckling expressions, for the critical size, as determined from the multiplication measurements and the exponential measurements. This expression was then used for determining the extrapolation lengths. The extrapolation lengths for these systems were in the range 7-8 cm.

¹R. C. Lloyd, R. B. Smith, and E. D. Clayton, "Neutron Multiplication Measurements of Heterogeneous 3.1% Enriched Uranium-Water Systems," *Trans. Amer. Nucl. Soc.* 2-1, 62 (1959).

Nuclear Safety of UO_2 - ThO_2 - H_2O Systems, D. B. Wehmeyer and K. E. Roach (B&W).

All known clean critical assemblies with H_2O moderator have been correlated with a modified two-group theoretical model.¹ Agreement with experimental critical sizes is good over a wide range of H/U-235, for homogeneous and heterogeneous systems, for high and low enrichment, for metal and oxide fuels, for U-238 and Th-232 fertile materials, and for aluminum and stainless steel structural materials. A representative set of experiments and comparison calculations are given in Table I.

This theoretical model has been applied to a number of experimental critical systems fueled with mixtures of thorium oxide and highly enriched uranium oxide. This comparison is shown in Table II. The experiments were performed at the Babcock & Wilcox Critical Experiment Laboratory as part of the Consolidated Edison Thorium Reactor Experiments² and the Thorium-Uranium Physics Experiments³ programs. Excellent agreement between calculated and experimental critical sizes justify this model for the prediction of mass and geometry limits for production lines, and it has been applied to the process line for manufacturing the Consolidated Edison fuel elements. Five different fuel mixtures were considered for homogeneous material, fuel pellets, and clad fuel pins. Safety factors based on operating conditions and on estimated uncertainties in the calculations were applied to the calculated minimum values of the critical parameters in order to determine safe values. The calculated minimum values of the critical parameters, and the safe values, are presented in Table III.

TABLE I
Representative Comparisons of Calculation and Experiment^a

Experiment	Reference	Fuel	Enrichment	Density ^b	Clad	H/U-235	Critical Measurement	Measured	Calculated
1	3	Homogeneous	> 90%	-	-	524	Sphere Radius	18.0 cm	16.8
2	3	Homogeneous	> 90%	-	-	53	∞ Cylinder Radius	7.6 cm	8.8
3	3	Homogeneous	4.9%	-	-	530	B ³	0.0127 cm ⁻³	0.0123
4	4	Homogeneous	2%	-	-	294	B ³	0.0067 cm ⁻³	0.0073
5	4	Homogeneous	2%	-	-	195	B ³	0.0052 cm ⁻³	0.0054
6	5	UO ₂ Pins	4%	9.45	SS	345	5 1/2 ft Cylinder Radius	19.25 cm	19.9
7	6	UO ₂ Pins	4%	9.45	SS	90	5 1/2 ft Cylinder Radius	21.73 cm	21.7
8	5	UO ₂ Pins	3%	9.45	SS	373	5 1/2 ft Cylinder Radius	24.2 cm	24.0
9	5	UO ₂ Pins	3%	9.45	SS	120	5 1/2 ft Cylinder Radius	26.7 cm	26.6
10	6	UO ₂ Pins	2.7%	10.2	SS	151	4 ft Cylinder Radius	26.8 cm	24.5
11	6	UO ₂ Pins	1.3%	7.53	Al	418	4 ft Cylinder Radius	39.7 cm	37.4
12	6	U Metal Pins	1.3%	18.9	Al	319	4 ft Cylinder Radius	25.5 cm	22.3
13	6	U Metal Pins	1.15%	18.9	Al	242	4 ft Cylinder Radius	29.6 cm	25.0

^a All experiments water reflected.^b Density of oxide or metal fuel in g/cc.

TABLE II
Critical Experiment Data^a for UO₂-ThO₂ Fuel Pins in H₂O^a

Experiment	Pin Radius (cm)	Fuel Radius (cm)	Clad Thickness (cm)	Clad Material	Fuel Density (g/cc)	N ₂₃₅ /N _{Th}	Volume Pins Volume H ₂ O	H/U-235	Fuel Height (cm)	Critical Cylinder Radius (cm)	Experimental	Calculated
14	0.396	0.330	0.048	SS	8.36	0.0388	0.890	152	121.9	30.15	30.10	
15	0.396	0.330	0.048	SS	8.36	0.0388	1.116	121	121.9	43.61	43.93	
16	0.396	0.330	0.048	SS	8.36	0.0667	0.152	531	121.9	28.0	28.88	
17	0.396	0.330	0.048	SS	8.36	0.0667	0.358	225	121.9	19.2	19.08	
18	0.396	0.330	0.048	SS	8.36	0.0667	0.656	123	121.9	20.1	19.85	
19	0.396	0.330	0.048	SS	8.36	0.0667	0.890	91	121.9	21.9	21.67	
20	0.396	0.330	0.048	SS	8.36	0.0667	1.116	72	121.9	24.3	23.85	
21	0.3924	0.2972	0.048	Al	8.45	0.04	0.300	524	152.4	27.3	27.03	
22	0.3924	0.2972	0.048	Al	8.45	0.04	0.4797	328	152.4	23.4	23.44	

^a All experiments water reflected

TABLE III
Calculated Minimum Critical Parameters for UO₂-ThO₂-H₂O

N ₂₃₅ /N _{Th}	Form	Mass U-235 (g)		∞Cylinder Radius (cm)		∞Slab Thickness (cm)		Sphere Volume (L)		Mass U-235/Unit Length in an ∞Cylinder (g/cm)	
		Minimum	Safe	Minimum	Safe	Minimum	Safe	Minimum	Safe	Minimum	Safe
		Critical	Value	Critical	Value	Critical	Value	Critical	Value	Critical	Value
0.031	Homogeneous	10,526	4,160	26.23	22.71	28.77	22.13	208.93	127.73	105.89	41.85
	Pellets ^a	8,831	3,345	22.32	18.45	23.81	17.11	132.26	67.83	101.42	36.75
	Fuel Pins (SS Clad)	78,671	32,644	49.38	43.70	58.95	46.79	1,257.28	838.19	468.12	191.85
0.045	Homogeneous	4,609	1,822	18.32	15.86	18.44	14.18	76.50	47.22	58.98	23.31
	Pellets	4,979	1,886	16.38	13.54	15.84	11.48	56.78	29.12	70.28	25.46
	Fuel Pins (SS Clad)	12,161	5,046	24.27	21.48	26.16	20.76	166.92	111.28	131.56	53.92
0.062	Homogeneous	3,043	1,203	15.26	13.21	14.44	11.11	46.26	28.56	43.81	17.32
	Pellets	3,932	1,489	13.85	11.45	12.55	9.09	36.24	18.58	60.55	21.94
	Fuel Pins (SS Clad)	6,848	2,841	18.18	16.09	18.20	14.44	75.32	50.21	90.54	37.11
0.085	Homogeneous	2,240	885	13.35	11.56	11.96	9.20	31.97	19.73	35.41	14.00
	Pellets	3,438	1,302	12.22	10.10	10.45	7.57	26.03	13.35	55.69	20.18
	Fuel Pins (SS Clad)	5,001	2,075	14.79	13.09	13.77	10.93	43.24	28.83	75.00	30.74
0.109	Homogeneous	1,885	745	12.41	10.74	10.78	8.39	26.06	16.09	31.35	12.39
	Pellets	3,265	1,237	11.42	9.44	9.42	6.83	21.85	11.21	55.16	19.99
	Fuel Pins (SS Clad)	4,423	1,835	13.23	11.71	11.74	9.32	32.20	21.47	69.08	28.31

¹ M. C. Edlund et al., "Physics of Water-Moderated Thorium Reactors," Proc. 2nd Int. Conf. on Peaceful Uses of Atomic Energy, Vol. 13, United Nations, Geneva (1958) p. 482.² N. L. Snidow, The Babcock and Wilcox Co., Lynchburg, Va., private communication.³ H. F. Henry et al., "Criticality Data and Nuclear Safety Guide Applicable to the Oak Ridge Gaseous Diffusion Plant," K-1019, 5th Rev. (May 1959).⁴ E. P. Blizard et al., "Neutron Physics Division Annual Progress Report," ORNL-2609 (Oct. 1958).⁵ A. L. MacKinney and R. M. Ball, "Reactivity Measurements on Unperturbed, Slightly Enriched Uranium Dioxide Lattices," Trans. Amer. Nucl. Soc. 3-1, (1960).⁶ H. Kouts et al., "Physics of Slightly Enriched, Normal Water Lattices," Proc. 2nd Int. Conf. on Peaceful Uses of Atomic Energy, Vol. 12, United Nations, Geneva (1958) p. 446.

Nuclear Safety in Handling Long Fuel Rods, N. Ketzlach (AI). Maximum safe cylinder diameters, slab thicknesses, and minimum critical masses in spherical geometries for water-moderated and water-reflected fuels of less than 5% U-235 enrichment have been evaluated earlier.¹ Since these critical masses and critical geometries are relatively small, other criteria must be used to determine the safe handling limits for large quantities of reactor fuels. This is particularly true for long fuel rods, one of which may contain a quantity of U-235 greater than the minimum critical mass based on spherical geometry.

The criticality calculations for the earlier study were made using the four-factor formulation and an approximation to age theory.² The results presented in this study for fuel enrichments of 3% U-235 and lower were based on these earlier calculations. However, the criticality calculations for 5% U-235-enriched fuels were made using two-group theory. It was decided that the latter method gave more accurate results at the higher enrichments, particularly for fuel rods of larger diameters.³

It has been determined that much larger masses of fuel may be stored safely than would be predicted by the presently used methods of calculation. The critical mass per unit length of an infinite cylinder goes through a minimum when plotted against fuel concentration (water-moderated systems).⁴ Therefore, if the long fuel rods can be stored in clusters which can be thought of as being inside infinitely long cylinders, nuclear safety can be determined by calculating the minimum critical mass per unit length of cylinder for the fuel rod size under consideration.

The critical masses per unit length of fuel have been calculated for several enrichments, fuel rod sizes, and volume ratios of moderator to fuel. The minimum critical masses per unit length of fuel at optimum moderation are plotted in Figure 1 as a function of fuel rod diameter for fuels of several U-235 enrichments. It is interesting to note that for the larger diameter fuel rods the minimum critical mass per unit length (kg U-235/ft) is smaller for lower enrichments than for higher ones. However, the minimum critical mass per unit length at optimum rod size and moderation is always smaller for the higher enrichments.

This phenomenon has also been observed over a much wider range of enrichments for fuel rods of $\frac{1}{2}$ -in. diameter. The calculated critical masses for fuels (1 to 5% U-235) of this rod size are compared with experimentally determined critical masses of this diameter at an enrichment of 93.5% U-235.⁵ The minimum critical mass at the higher enrichment is about 6 kg contained U-235, compared with about 2, 2.5, and 5 kg at 5.0, 3.0, and 1.6% U-235 enrichment, respectively.

Fig. 1. Calculated Minimum Critical Mass/Unit Length vs. Fuel Rod Diameter

- ¹N. Ketzlach, "Nuclear Safety in Processing Less than 5.0 Per Cent U-235 Enriched Reactor Fuels," HW-58049 (Dec. 11, 1958).
- ²N. Ketzlach, "Nuclear Safety of Iron-Encased Fuel Elements," Nucl. Sci. and Eng. 6, 187 (1959).
- ³N. Ketzlach, "Criticality of UC Fuel Elements," NAA-SR-MEMO-4746 (Dec. 14, 1959).
- ⁴N. Ketzlach, "Bases for Establishing Nuclear Safety Criteria," Preprint V-35, Nuclear Engineering and Science Conference (1959).
- ⁵G. A. Graves and H. C. Paxton, "Critical Masses of Oralloid Assemblies," Nucleonics 15, 90 (June 1957).

Nuclear Safety in Enriched Uranium Processing,* H. T. Williams and J. W. McWhirter (CV-FW). The growth of the enriched, unirradiated uranium processing industry has emphasized the importance of a realistic appraisal of the measures required to insure safety in handling enriched uranium and of the need for a clear appreciation of the potential hazards involved.

This paper presents the results of a detailed study of enriched uranium processing and fabrication operations in commercial plants. The various steps in handling enriched uranium were considered in detail to identify the potential sources of nuclear accidents and to determine the probable circumstances of their occurrence. From this information, estimates were made of the probable range of accident magnitudes and the possible consequences of such accidents.

The relative importance of the various possible accident sources can be best evaluated in terms of the total fission yield potentially available in each case. In evaluating these yields an attempt was made to consider the effects of each of the major excursion parameters. Combining this information with the estimated relative probabilities of various accident sources, it was possible to identify the more significant types of excursions that might occur. It appears that the majority of nuclear excursions will probably occur in aqueous assemblies and produce fission yields lying within a relatively narrow range. The range of magnitudes for these "average accidents" is estimated to be from 4×10^{16} to 4×10^{17} fissions if the excursion terminates after the initial power burst. However, the excursion may reach 10^{19} fissions if prolonged power operation occurs.

Fission yields from the most reactive of the aqueous and metallic assemblies considered to have a reasonable probability of occurrence are taken as representative of the upper end of the accident spectrum. For "maximum" aqueous accidents, single burst and prolonged yields up to 1×10^{18} and 8×10^{18} fissions, respectively, are considered possible. Accidents involving enriched uranium "buttons" were calculated to be potentially capable of producing the highest yields, i.e., from 10^{20} to 10^{21} fissions in a single burst. Fortunately, such an extreme accident, which might produce an explosion equivalent to a ton or so of high explosives, appears to be far more unlikely than the "average accident."

Following the identification of probable accident sources and their magnitudes, the extent of human injury that might be anticipated as a result of the direct radiation from the assembly and from the release and subsequent dispersal of the fission products was determined. The range of significant, total effective exposures, as a function of accident magnitude, shows that the 1-AED hazard range of single burst "average accidents" is less than 100 meters, while the lethal range is probably less than 10 meters. Prolonged aqueous accidents were found to be potentially lethal to distances not exceeding 50 meters even under adverse weather conditions. However, they could result in 1-AED exposures out to about 200 meters, primarily due to the thyroid dose from inhaled iodine. This assumes that only the volatile fission products are released from aqueous assemblies.

In the event of a "maximum accident" producing a total of 10^{20} fissions in a single burst, it is likely that most of the persons within a radius of 100 meters of the source would be killed by the explosive blast, flying debris, or by direct radiation. Lethal doses from exposure to the radioactive cloud may be received downwind from the source as far as 300 meters. One-AED doses from the direct radiation and cloud exposure could be received at distances of 300 and 1800 meters, respectively.

It should be emphasized that these results apply only to the case of accidents involving unirradiated, enriched uranium and should not be taken as a measure of the potential hazard associated with the processing of irradiated uranium.

Critical Experiments

Critical Experiments Involving Gaps in Simple Configurations, E. Goldberg, F. A. Kloverstrom and D. E. Kraft (UCRL-L). A series of critical assembly experiments have been performed to determine the neutronic effect of gaps in simple configurations. In one series, the gap is located in the median plane of a bare rectangular parallelepiped. In the other, a gap separates a core from its side reflector. Procurement of data which could be compared to predictions regarding gaps, and development of satisfactory computational techniques were the main reasons for the experiments. In addition the data may provide values for reflector savings in several cases.

The first series involved BeO as a moderator and enriched uranium, i.e. 93.2% U-235 as fuel. The fuel loading was BeO:U-235 = 493:1 and the foil thickness and spacing was four mils per inch. The BeO density was 2.86 g/cc. The facility and experimental details are discussed elsewhere.^{1,2} The results are given in Table I. No control vane assembly was present.

TABLE I
Critical Heights of Assemblies Involving Gaps

Exp't. No.	Base Dimensions (inches) ^a	Core Weight H _c ^a (inches)	Gap Thickness T _v ^a (inches)	Gap Contents
1	24 x 24	22.2 ± 0.2	0.00	--
2	24 x 24	24.53 ± 0.2	0.47	Hexcell ^b
3	24 x 24	27.56 ± 0.2	0.94	Hexcell ^b

^a The overall height is H_c + T_v.

^b Hexcell is aluminum, $\rho = 0.053$ g/cc.

The findings of Table I establish an empirical recipe suggested by perturbation theory that the effective density change of the assembly is the product of $\phi_{\text{gap}}^2/\phi_{\text{assembly}}^2$ and the apparent density change, $V_{\text{gap}}/V_{\text{assembly}}$.

A second series of experiments was performed using graphite in the core as moderator, and a graphite reflector. The findings of the previous BeO experiments are used to correct the experiments listed in Table II for vane slot voids.³ The correction to critical height ranged from 0.2 in. to 0.3 in. in the second series. The core is of square cross-section (i.e. L_c = W_c). A gap, T_v in width, separates the core from the side reflector of thickness, T_r, which encompasses the four vertical surfaces of the core. The core height, H_c, is equal to the side reflector height. The vertical reflector faces are parallel to the neighboring core faces, and the overall cross section is (L_c + 2T_v + 2T_r) x (W_c + 2T_v + 2T_r). Experiments 97, 99 and 103 illustrate the effect on critical height of various core side reflector gaps where the reflector is 6.0 in. thick. It is found from the data and buckling arguments that the gaps cost about \$3.00/inch of gap.

¹E. Goldberg and F. A. Kloverstrom, "Resonance Absorption of Neutrons in Thick Sheets of Molybdenum," Trans. Amer. Nucl. Soc. 3-1, 213 (1960).

²F. A. Kloverstrom et al., "Critical Measurements on Near - Homogeneous BeO - Moderated, Oralloid - Fueled Systems," UCRL-5369. Pt. 1 Revised (1959).

³H. L. Reynolds, "Critical Measurements and Calculations for Enriched-Uranium Graphite-Moderated Systems," UCRL-5175 (1958), pp. 2-7.

TABLE II
Critical Assemblies Containing Core-Side Reflector Gaps

Exp.	L _c , W _c (in.)	H _c (corr.) ^a (in.)	T _r (in.)	T _v (in.)	ρ_c (g/cc)	ρ_r (g/cc)	C/U-235 ^c	$\Delta K_{\text{eff}}/\Delta H_c$ ^d (\$/inch)
92	48.5	35.57 ± 0.2	--	--	1.633	--	296	
93	42.5	31.92 ± 0.2	6.0	--	1.632	1.70	295	\$2.44 ± 10%/inch
94	42.5	28.97 ± 0.2	12.0	--	1.636	1.70	295	
97	42.5	33.13 ± 0.2	6.0	1.0	1.632	1.70	295	
99	42.5	34.25 ± 0.2	6.0	2.0	1.632	1.70	295	
103	42.5	35.30 ± 0.2	6.0	3.0	1.632	1.70	295	
119	36.5	35.73 ± 0.2	6.0	--	1.635	1.70	298	\$1.96 ± 5%/inch
122	36.5	40.27 ± 0.2	6.0	2.0	1.635	1.70	298	

^a Corrected for vane slot voids.

^b Density of aluminum honeycomb in gap: 0.018 g/cc.

^c Uncorrected for self-shielding. Foil spacing: 4 mils/1/4 inch.

^d These values were found by performing a series of inverse multiplication measurements at two different assembly heights with a control rod calibrated in dollars.

High Temperature Beryllium-Oxide Critical Experiments, R. Finke, G. Goldberg and H. Reynolds (UCRL-L) and G. Grammens (UCRL). The effect of temperature on the critical dimensions of a beryllium-oxide moderated assembly has been determined at the HOT BOX high temperature critical facility¹ at the Nevada Test Site. The temperature range between room temperature and 950°F was studied.

The oven and supporting equipment, and the oralloy fuel foils were the same as for the earlier graphite experiments.²

The 6 in. x 6 in. x 1 in. beryllium oxide blocks were stacked on spacers five inches above the 2½ in. thick graphite base-seal layer which covered the supporting low-mass table. The density of the BeO within the dimensions of a block was 2.86 gm/cm³. Pulsed neutron measurements established the thermal absorption cross section in the BeO to be 7 ± 4 mb per molecule. Each layer was made up of four blocks vertically and four blocks laterally to give nominally 24 in. x 24 in. for the cross-

section dimensions of the BeO assembly. The longitudinal dimension was adjusted to approach criticality by changing the number of layers.

The assembly was divided into four quadrants in the horizontal plane by a one inch thick graphite cruciform with a ½ in. x 10 in. section slot in each arm to accommodate the vertically inserted control elements.

The BeO blocks were maintained in position by an inner graphite box as shown in Figure 1. The density of the graphite in the box walls and in the cruciform was $1.74 \pm .02$ gm/cm³. The base seal layer graphite had a density of 1.67 gm/cm³.

A pair of the standard 9 in. wide fuel foils were suspended vertically in the center of each 12 in. wide lateral slot between layers in all four quadrants. Unfueled BeO regions within the quadrants were produced by this arrangement.

The 2-mil fuel elements had an average weight of oralloy (93.2% U-235) of 123 gm with a variation of about 10%. The cosine-squared flux-weighted average oralloy mass of the actual foils used was 128.9 gm. Associated with each fuel element is 105.5 gm of stainless steel Type 347.

Two lengths of assembly were studied: 31 layers for the room temperature (90°F) assembly and 33 layers for the elevated temperature (950°F) assembly. A space of 0.043 in. was left between layers for the fuel elements and for differential thermal expansion.

For ease of interpretation, the results are stated in the following manner. The assemblies are reduced to a simplified geometry of 4 homogeneous BeO quadrants divided by a graphite cruciform, all isolated from any reflective surroundings. Experimentally determined equivalences allow conversion of the graphite supporting box, the graphite base layer, and the low-mass table to BeO. The dimensions are then modified. The effective density of the BeO in each quadrant was 2.72 gm/cm³. The effective width of the assemblies was 26.08 in. including the 1 in. thick half density graphite cruciform. The effective critical length of the 90°F homogenized assembly was 33.79 in. and the effective length of the assembly critical at $955 \pm 15^\circ\text{F}$ was 35.91 in. The latter value is corrected to the same width, height, and density as the 90°F assembly. The BeO/U-235 molar ratio averaged over a quadrant was about 560:1 uncorrected for self-shielding.

The experimentally measured overall average temperature coefficient of reactivity between 90°F and 950°F was $-\$0.48 \pm 0.05/100^\circ\text{F}$ and the calculated thermal expansion contribution was $-\$0.09 \pm 0.02/100^\circ\text{F}$, giving $-\$0.39 \pm 0.05/100^\circ\text{F}$ for the nuclear effect. The local temperature coefficient in the vicinity of 900°F was $-\$0.48 \pm 0.03/100^\circ\text{F}$ and the nuclear component was $-\$0.37 \pm 0.04/100^\circ\text{F}$.

Fig. 1. View From Above of Bare BeO Critical Assembly Depth: 24.0

¹H. L. Reynolds and C. E. Walter, "Hot Box, A High Temperature Critical Facility," UCRL-5483 (1959).

²E. Goldberg, R. Finke, and H. L. Reynolds, "High Temperature Graphite Critical Experiments," Trans. Amer. Nucl. Soc. 3-1, 71 (1960).

Exponential Experiments and Reactivity Measurements

Pulsed Neutron Measurements of Subcritical Lattices, E. E. Carroll, Jr., N. Hartmann, and D. Klein (W-BAPD). The thermal neutron time decay constants of a series of reflected, highly enriched, light water moderated, essentially one-dimensional, subcritical lattices have been measured using the pulsed neutron technique. These measurements were compared with values obtained through four-group diffusion theory calculations. Two-group diffusion theory calculations of decay constants for higher harmonic spatial modes were made to examine the possibility of higher harmonic contamination in fundamental mode measurements. A few measurements of harmonic decay constants were made as a further check.

The cores pulsed consisted of sheets of uranium-zirconium fuel 29.5 x 48 x 0.1428 in. thick, separated by 0.350 in. of water. The 15-sheet thick core (~7 in.) was clean critical ($K_{eff} = 1.0039$; $K_{eff} = 1.0018$ with pulsing equipment inserted in the reflector). Measurements were made on assemblies from 14-sheet thick to 7 sheets thick.

A Kaman pulsed neutron source was used, located in the reflector adjacent to the core, with the target positioned at the center of one face. Scintillators of $B^{10} + ZnS(Ag)$ detected the leakage flux, and the pulse output was observed with a 256-channel analyzer.³ Some measurements were made with miniature BF_3 counters inserted in the core proper. Detectors were positioned to achieve maximum suppression of higher mode contamination. Neutron bursts were about 10 μ sec long, and only data taken more than about 1.3 msec from the burst time were considered.

P-1 fast group constants were obtained using the MUFT-5 fast neutron spectrum code.¹ Thermal constants were calculated over a Wigner-Wilkins flux spectrum using the KATE-1 code.² Thermal flux-weighting corrections for core heterogeneities were made using a one-energy group, one-dimensional, double P-3 approximation to the transport equation (FLIP code).³ The one-dimensional few-group, diffusion theory code WANDA⁴ provided values of K_{eff} for each lattice.

The time dependent diffusion equations may be reduced to time independent equations when exponential flux behavior is assumed. A $(-\omega/v)$ correction to the absorption cross-sections results. If a prompt critical eigenvalue is known and few-group parameters appropriate to the subcritical core are used, WANDA will solve for $(-\omega/v)$.

TABLE I
Comparison of Calculated and Measured Decay Constants

Core Size (Sheets thick)	Calc ω (sec ⁻¹)	Meas ω (sec ⁻¹)	ω_{1st} Meas (sec ⁻¹)	ω_{1st} Meas (sec ⁻¹)	K_{eff} (Calc)
15	85.06	--	--	--	1.0072
14	444.00	444.0 $\pm$.8	1000 $\pm$ 10%	1280 $\pm$ 20%	0.9807
13	809.75	805.6 $\pm$ 1.9	1380 $\pm$ 10%	1970 $\pm$ 20%	0.9514
12	1188.9	1178.8 $\pm$ 1.9	1920 $\pm$ 10%	2630 $\pm$ 20%	0.9198
11	1573.0	1559.7 $\pm$ 3.1	uncertainties	uncertainties	0.8852
10	1957.9	1937.0 $\pm$ 16	estimated	estimated	0.8472
9	2336.1	2297.3 $\pm$ 6.3	--	--	0.8056
8	2700.3	2630.5 $\pm$ 2.0	--	--	0.7597
7	3044.5	2928.6 $\pm$ 2.0	--	--	0.7093

^a ω_{1st} and ω_{2nd} indicates the second and third harmonics, respectively, in the longest core dimension.

This ω/v must be converted to a negative number density of a $1/v$ absorber and inserted into MUFT-5 and KATE-1 to correct for the spectral softening in the time dependent case. Further iterations provide converged values of ω . The first measured decay constant was used to normalize the procedure and provide a prompt-critical eigenvalue for use in calculations for the smaller cores.

Table I compares the predicted and measured decay constants. Calculated values of K_{eff} are shown for reference. The uncertainty indicated for the decay constants is only the standard deviation due to counting statistics based on the least squares analyses of the various runs. Measured and calculated values agree within 4%. The differences increase steadily with decreasing core size.

¹ H. Bohl, Jr., "MUFT-5—A Fast Neutron Spectrum Program for the Philco-2000," WAPD-TM-218 (June 1960).

² H. J. Amster and J. B. Callaghan, "KATE-1—A Program for Calculating Wigner-Wilkins and Maxwellian Averaged Thermal Constants on the PHILCO-2000," WAPD-TM-232 (October 1960).

³ B. L. Anderson, et. al., "FLIP—An IBM-704 Code To Solve the PL and Double PL Equations in Slab Geometry," WAPD-TM-134 (March 1959).

⁴ O. J. Marlowe, "WANDA—A One-Dimensional Few-Group Diffusion Equation Code for the IBM-704," WAPD-TM-28, Add. 2 (July 1959).

Exponential Experiments in Uranium-Diphenyl Lattices, R. W. Campbell, T. L. Guzzle, and R. K. Paschall (AI). Material buckling and thermal utilization measurements have been made for a series of diphenyl-moderated, uranium metal exponential assemblies in order to provide experimental data for simple lattices which can be treated most readily by analytical means. Material bucklings have been reported earlier,¹ and these measurements indicated that a positive buckling is possible for a square array of 1-in. diam natural uranium fuel rods spaced to give a moderator to fuel ratio of approximately 3.5:1. Considerable doubt is cast on the validity of these data due to the presence of considerable epithermal leakage into the exponential assembly. To eliminate this uncertainty, measurements have now been repeated for one lattice spacing of natural uranium fuel elements. Thermal utilizations were determined for several additional square lattice spacings with fuel enrichments of 0.496 (depleted), 0.721 (natural), and 0.912 (enriched) a/o U-235.

Several improvements in the experimental assembly have been made:

- 1) The diameter of the assembly was increased from 27 to 60 in. The height was 48 in. The 848 natural uranium fuel elements were clad with 0.060-in. aluminum and aligned in a square array to give a moderator to fuel ratio of 6.46:1. The entire assembly was at a temperature of $180^\circ \pm 2^\circ F$.
- 2) Fuel rods across an entire radius were removable as two units thus enabling horizontal flux measurements to be made with the full fuel loading.
- 3) Arrangements were made so that a layer of Cd could be put between the thermal column and the loaded as-

TABLE I
Thermal Utilization in Uranium-Diphenyl Lattices

Fuel	Cell Size (in.)	Thermal Utilization	
		(measured)	(theory -S ₄)
0.721 a/o U-235 18.88g/cc	1.437	0.927 ± 0.004	0.925
	1.578	0.897 ± 0.002	0.901
	1.816	0.842 ± 0.008	0.854
	2.016	0.803 ± 0.007	0.812
0.496 a/o U-235 18.80g/cc	1.437	0.910 ± 0.006	0.913
	1.816	0.818 ± 0.004	0.786
0.912 a/o U-235 18.83g/cc	1.437	0.929 ± 0.006	0.932
	1.816	0.856 ± 0.004	0.866

sembly so as to enhance any effects resulting from epithermal leakage into the sides and top of the assembly.

The values for the horizontal and vertical components of the buckling were found to be $\mu = 3.073 \pm 0.002 \text{ m}^{-1}$ and $\nu = 3.694 \pm 0.011 \text{ m}^{-1}$, respectively. The buckling as determined from the expression $B^2 = \mu^2 + \nu^2$ was found to be $-4.203 \pm 0.015 \text{ m}^{-2}$. This value is in disagreement with the value reported earlier of $+2.2 \pm 0.2 \text{ m}^{-2}$. Thorough examination of the experimental techniques used in both measurements strongly indicates that the results of the latter measurements should replace those done earlier.

Intracell thermal flux distributions were measured for varying fuel enrichment and lattice spacing in 13-cell clusters buffered by the lattice in which the buckling was made.

The values of thermal utilization for the 8 lattices as determined from these measurements are shown in Table I. The theoretical values were obtained from the S₄ approximation to the transport equations. Cross sections used were averaged over a Wigner-Wilkins spectrum.

¹W. W. Brown, "Exponential Experiments with Organic Moderated Uranium Lattices," *Proc. Second Geneva Conf.* (United Nations, New York), 12 (1958), p. 514.

Buckling Measurements of D₂O Moderated Clusters of Natural Uranium Oxide Rods, N. P. Baumann and E. J. Hennelly (DuP-SR). Buckling measurements have been made on lattices of Swedish R-3/Adam fuel assemblies in the Savannah River critical facility, the PDP,¹ and the high temperature exponential facility, the PSE.² The R-3/Adam fuel was in the form of sintered pellets of natural uranium oxide 1.70 cm in diameter with a density of 10.32 g/cc averaged over the active length of the rod. The pellets were contained in Zircaloy-2 tubes 1.72 cm ID, 1.86 cm OD, and 75 cm long. Each cluster consisted of 19 such rods in a hexagonal array with a center-to-center spacing of 2.11 cm, except that the six rods on the flats of the outer hexagonal ring were displaced radially outward an additional 0.29 cm.

For the exponential measurements, two lengths of rods were used to make assemblies 150 cm in length. For the 24-cm pitch square lattice, 24 assemblies were arranged on a 6 x 6 array with three assemblies removed from each corner. For 27- and 30-cm pitches, 21 assemblies

Fig. 1. Bucklings of R-3/ADAM Lattices

were arranged in a 5 x 5 array with one element removed from each corner. Vertical relaxation lengths were measured by activating foils made on a high melting alloy of 20 w/o indium in lead and spaced at 5-cm intervals. Two problems arose from the construction of the assemblies. First, because of the short length of the fuel, a topreflector of D₂O resulted which varied from 0 to 20 cm as the temperature increased. Second, since approximately 4.5 cm of fuel was missing at the vertical junctions of the fuel rods, a local thermal flux peaking occurred in the vicinity of the missing fuel. Experiments at room temperature showed no systematic dependence of the measured buckling on reflector thickness. Two-energy group diffusion calculations along with experimental checks indicated that the missing fuel would have little effect (less than 0.1 m⁻²) on the measured buckling as long as foils on either side of the missing fuel were equally weighted in the analysis. The results of the buckling measurements extrapolated to 20° and 220°C are given in Figure 1. The radial buckling value of $8.45 \pm 0.05 \text{ m}^{-2}$ used for the exponential facility at room temperature was based on a recent intercomparison in this Laboratory between critical and exponential measurements using metal uranium lattices.

The critical facility measurements with the R-3/Adam fuel were made by substitution of seven or twelve test assemblies into a reference lattice with nearly identical nuclear properties. A further calculated correction was included to adjust the measured bucklings to those that would have been obtained had the fuel rods been continuous. Figure 1 shows the measured bucklings for both facilities and compares them with calculated values.

²G. Dessauer, "Physics of Natural Uranium Lattices in Heavy Water," *Proc. Second Geneva Conf.* (United Nations, New York), 12 (1958), p. 320.

²E. J. Hennelly, et al., "Pressurized Facility for Subcritical Lattice Measurements at High Temperatures," TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 2, 1 (1959), p. 64.

Precision Critical Substitution Measurements of Bucklings in D₂O Moderated Lattices, W. E. Graves (DuP-SR). Recent refinements in critical reactor substitution techniques have permitted the measurement of the bucklings of D₂O moderated lattices with high precision using only a few "unknown" fuel assemblies. The general applicability of these techniques depends on the availability of a full pile load of fuel assembly components which can be combined into a wide variety of fuel assemblies. It is then possible to construct reference fuel assemblies which closely match the expected nuclear parameters of the unknown fuel assemblies. It must also be possible to poison the reference assemblies fairly uniformly with, for instance, copper. An example of this type of substitution measurement is described below.

As part of a Swedish-U.S. exchange program on D₂O moderated power reactors, critical substitution measurements were made in the Process Development Pile (PDP) of the bucklings of D₂O moderated lattices utilizing fuel assemblies intended for use in the Swedish R-3/Adam reactor. Exponential measurements on these fuel assemblies have been reported in the previous paper.¹ Each of these fuel assemblies was a cluster of nineteen UO₂ rods. Complete descriptions of the assemblies have been given.²

The combination driver and reference fuel assemblies chosen to match the Swedish assemblies were clusters of 31 smaller UO₂ rods.³ Copper poison was added to the 31 rod fuel assemblies in three steps. At each stage of poisoning, lattice bucklings were measured at two triangular pitches, 9.33" and 12.12", by means of critical one-region pile loadings.

Critical substitution measurements were made at both the 9.33" and 12.12" pitches by using the unpoisoned reference lattice as an outer, or driver, lattice surrounding a central test region. The test region was calibrated by loading it with the unpoisoned and the three poisoned lattices of the same pitch and measuring the critical moderator height in each case. The unknown lattice was then loaded into the test region and a critical moderator height again obtained. At each pitch, two different sized test regions were used, the first including 7 fuel assemblies and the second 12. By means of an IBM-650 code which matches the two-group fluxes and currents at the test region boundary, the reference lattice data were used to fix the effective location of the boundary and also the bucklings in the test region. These parameters were then used in connection with the measured moderator heights in the unknown cases to compute the buckling of the unknown lattice by two-group theory. One group theory was also used for comparison.

The agreement between bucklings obtained for the R-3/Adam lattices using the two different sized test re-

gions at a single pitch was quite good, the discrepancy being less than 0.02 m⁻² at both pitches. The difference between one-group and two-group results was only 0.01 m⁻², indicating a good match between outer region and test region.

The Swedish fuel assemblies had a vertical gap of 4.5 cm in the UO₂ near the vertical center of the reactor where two separate fuel sections were attached together. A correction was calculated by perturbation theory and applied to the measured bucklings to obtain buckling corresponding to the condition of no fuel gap. Bucklings results, corrected to 99.75 mol % D₂O and 20°C, are given in Table I.

¹N. P. Baumann and E. J. Hennelly, "Buckling Measurements of D₂O Moderated Clusters of Natural Uranium Oxide Rods," TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 4, 1 (1961), p. 48.

²Heavy Water Moderated Power Reactors - Monthly Progress Reports. E. I. du Pont de Nemours and Co., Wilmington, Del. AEC Research and Development Reports DP-565, (November 1960) and DP-585 (January 1961).

Lattice Parameter Measurements for a Concentric Tube Fuel Element, D. E. Wood, K. R. Birney, and E. Z. Block (GE-HAPO). Lattice parameters have been measured for a concentric tube, natural uranium fueled lattice, moderated by graphite. The experiments were made to test calculational models for lattices with the complex fuel geometry involved. For the 10-1/2-in. lattice, k_{∞} , f , p , and ϵ were measured with water and air in the coolant channels. For the 8-3/8-in. lattice, k_{∞} and f were measured with water coolant only.

The concentric tube fuel assembly contained a 1.66 x 1.1-in. uranium tube inside a 2.5 x 2.0-in. tube. Aluminum cladding was used on the fuel, and an aluminum outer tube defined the outer coolant channel. The Physical Constants Test Reactor was used as a reactivity measurement facility and neutron source for the measurements. A nine-cell array of the test lattice was placed in the PCTR core with driving fuel around the core. Reactivity measurements established the amount of thermal absorber required to poison the central cell to unit multiplication. Copper foils were placed in all cell components and irradiated. The gamma decay of the 12.8-hr activity of the foils was counted. The excess resonance absorption in the foils was subtracted using cadmium ratio measurements. The resulting "1/v" relative reaction rates were used to infer the neutron absorption rates in all cell components.

The measurements of resonance escape probability and fast fission effect involve measurements of resonance capture rate and fission rate in integral foils of uranium placed in the fuel. The experimental results are CR²⁵, the cadmium ratio of neutron capture in U-238, and δ , the ratio of fast to slow fission.

The results of the measurements are shown in Table I for the three lattices. The values of f include 1/v absorption in the fuel, and the values of ϵ include fast absorption above 0.5 Mev. Thus, the value of p^{25} includes absorption in excess of 1/v absorption and has fast absorption removed.

The values of the effective resonance integral derived from the p measurement (neglecting resonance flux depression) are 9.61 ± 0.35 barns for the dry case and 11.15

TABLE I
Buckling Results

Triangular Lattice Pitch (in.)	R-3/Adam Fuel Assemblies		Unpoisoned 31 Rod Clusters
	B_m^2, m^{-2} , Fuel Gap Present	B_m^2, m^{-2} , No Fuel Gap	B_m^2, m^{-2}
9.33	5.32 ± 0.07	5.16 ± 0.10	5.29 ± 0.05
12.12	5.11 ± 0.08	5.13 ± 0.11	4.98 ± 0.06

TABLE I
Lattice Parameters for a Concentric Tube Fuel Element

Parameter	Lattice and Coolant		
	8-3/8-in., Water	10-1/2-in., Air	10-1/2-in., Water
k_{∞}	0.978 ± 0.003	$1.040 \pm .003$	0.984 ± 0.002
f	0.858 ± 0.005	$0.923 \pm .003$	0.831 ± 0.005
CR ² (inner tube)		2.503 ± 0.039	2.728 ± 0.012
CR ² (outer tube)		2.060 ± 0.033	2.575 ± 0.018
p^2		0.815 ± 0.006	0.867 ± 0.006
δ (inner tube)		0.1206	0.1266
δ (outer tube)		0.0910	0.0721
ϵ		1.061 ± 0.007	1.056 ± 0.007
η (from k_{∞}/fpe)		1.304 ± 0.012	1.295 ± 0.012

± 0.54 barns for the wet case. The equation $R.I. = 8 + 27.5 S/M^1$ yields 9.54 and 10.36 barns for the dry and wet cases, respectively. The value of $\bar{V}$ (the average effective fraction of the internal surfaces) derived from the difference in the resonance integrals is 0.53. The calculated value of $\bar{V}$ is 0.28 assuming that fast neutrons are scattered into a resonance in the internal water and do not make a second collision and that resonance neutrons already in the water are scattered out of resonance in a single collision.² A multiple scattering treatment would rise the calculated value of $\bar{V}$.

The value of f agrees with P_3 calculations with an adjusted neutron temperature.³ The value of η agrees with previous lattice derivations, but it is lower than the value from basic cross sections.

¹E. Hellstrand, *J. App. Phys.*, 28 (1957), p. 1493.

²E. Nilson, Unpublished Derivation, 1960.

³D. E. Wood, TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 3, 2 (1960), p. 492.

k_{∞} Measurements in the Pawling Lattice Test Rig, G. Foster, W. L. Brooks, M. Fleishman, and R. D. Schamberger (NDA). Measurements of the infinite multiplication factor of D_2O moderated natural uranium lattices are now being made in the Pawling Research Reactor (PRR) as modified to incorporate the Pawling Lattice Test Rig (PLATR). A totally new core has been installed in the PRR which has been described previously.¹ These core modifications allow k_{∞} measurements of the type obtained in the PCTR at Hanford² for D_2O lattices over a range of lattice pitches from 7.0 to 14 in. The measurements are part of the AEC Heavy Water Power Reactor Program.

The PLATR core consists of a cylindrical outer region fueled with fully enriched uranium on both ends and

the sides and a central region 4 ft high and 4.5 ft in diameter which constitutes the experimental region. The experimental region contains natural uranium fuel, part of which is a seven-cluster sample (4 ft long) of the lattice under test. The rest, the outer buffer fuel, is used to adjust the neutron spectrum in the central test section. The spectrum, as indicated by gold-cadmium ratio measurements, is adjusted to that found in an infinite lattice of the test clusters when poisoned so that k is unity.

The middle 27 in. of the central test cell, called the test section, is removable and may be replaced by a void can (helium filled). The fueled test section is then poisoned with thin copper strips until its removal and replacement by the void can causes no change in the reactivity of the system. The amount of copper poison together with the thermal flux ratios in the cell is used to calculate k_{∞} for the test lattice.

Measurements have been completed on a 190 rod cluster of 1/2-in. UO_2 rods on an 8-in. pitch with aluminum pressure and calandria tubes for comparison with a similar lattice measured in the PCTR. The k_{∞} results for 99.7% D_2O purity are 1.059 ± 0.005 for the PLATR measurement and 1.052 ± 0.003 for the PCTR measurement. In addition, a measurement was made in PLATR at a purity of 98.3% D_2O . The result of this measurement was 1.039 ± 0.005 . A simple purity correction was made to the 98.3% result assuming no change in the flux distribution. The corrected result, 1.055 ± 0.007 , was in good agreement with the measured value at the higher purity.

In order to obtain correct results from the experiments it is necessary to duplicate, over the test region, the neutron environment which would be present in an infinite lattice of the type under consideration. Studies were made of the magnitude of the error in k_{∞} resulting from failure to create these conditions exactly. The effect of an axial variation, of the thermal neutron flux over the test can, was less than 0.003 in k_{∞} for values of flux at the ends of the test can 5% less than that at the center. The effect of failure to match the neutron spectrum to its proper value, as indicated by gold-cadmium ratios on the test can and on the inner buffer, was less than 0.003 in k_{∞} for a 10% mismatch.

Some of the factors affecting the accuracy of the measurements were D_2O purity degradation, poisoning effects of aluminum structure, and D_2O height sensitivity.

¹G. G. Foster and R. D. Schamberger, TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 2, 2 (1959), p. 108.

²D. J. Donahue, D. D. Lanning, R. A. Bennet, and R. E. Heineman, *J. Nuc. Sci. and Eng.* 4 (1958), pp. 297-321.

Critical Experiments I

Experimental and Calculated Activations in a One-Dimensional Slab Core with Zirconium-H₂O Reflector, J. Korsmeyer, C. Bolmgren, and S. Milani (W-BAPD). The one-dimensional slab core was a highly enriched, light-water-moderated assembly consisting essentially of a fuel region with an adjacent Zr-H₂O reflector on one side and the remainder of the core surrounded by water. The core was designed to contain heterogeneities in one dimension of the core while maintaining homogeneity in the remaining two dimensions. Experimental and calculated activation shapes have been obtained in the Zr-H₂O region along the dimension containing the heterogeneities.

The fuel and Zr-H₂O regions consisted of 0.070 x 47.6 x 48-in. sheets of highly enriched U-Zr alloy and Zircaloy-2, respectively, separated by 0.070-in. water channels to give a metal-to-water ratio of unity. The fuel and Zr-H₂O regions were 4 and 20 in. wide, respectively. A total U-235 loading of 15.8 kg gave an essentially clean critical core.

Foil activation measurements were made with bare Mn foils and with Cd-covered In, Au, and U-238 foils along a horizontal line 18.1 in. above the bottom of the core passing through the center of the 48-in. core length. The Mn, In, Au, and U-238 foils were 0.375 in. in diameter and had effective thicknesses of 29, 3.4, 250, and 210 mg/cm², respectively. The Cd-covered data were obtained with the foils surrounded by a 0.020-in. thickness of Cd. The foils were imbedded in the core plates such that the integrity of the plate was maintained except for the volume occupied by the foil and the Cd when used. The Cd-covered data were obtained with a minimum foil spacing of 1.4 in. The error in the In and Mn activations varies from about 1.5% near the fuel region to about 3% at the opposite side of the

Zr-H₂O region. The error in the Au data varies from 1.5% to 4% while the error in the U-238 data varies from 2% to 5%. The error in the foil position is estimated to be ± 0.04 in.

Epi-cadmium activation shapes were calculated using few-group diffusion, multigroup P-1, and several group P-3 approximations to the transport equation to obtain the fluxes. Machine code calculations of multigroup average detector cross sections obtained from resonance parameters were also utilized. Bare foil activation shapes were calculated using P-1 and double P-1, one thermal group fluxes and multigroup double P-1 thermal fluxes. The epithermal activation contributions for bare foils were calculated in the approximations stated above.

P-3 calculations are shown to be necessary for agreements between experiment and calculation in the large Zr-H₂O reflector. Figure 1 indicates the experimental activation shape of Cd-covered depleted U-238 foils, several P-1 calculated shapes, and a P-3 calculation. Similar results were obtained for the other detectors used.

Analysis of the slopes of the multigroup fluxes on a semi-log plot indicates a variance of <1% in the range of Cd-covered In and Au activation.

The percentage contribution to the activations of Cd-covered In and Au between cadmium cutoff (0.425 ev) and the lower limit (0.625 ev) considered in fast machine codes is shown to be $\sim 1\%$ in the Zr-H₂O reflector region. The epithermal contribution to the total activation of bare Mn is shown to vary spatially and to be sizable (up to 24%) across the regions considered and hence quite important in determining the activation shape.

Technique for Analyzing Partial Water Height Experiments, W. B. Doub and J. E. Mott (W-BAPD). Previously, partial water height experiments¹ have been used to determine excess reactivity. Although useful as a core parameter, the experimentally inferred excess reactivity did not lend itself to precise comparison with theory. A related parameter, the buckling coefficient of reactivity at critically, $\frac{\Delta\rho}{\Delta B^2}$, can easily be inferred from a properly designed partial water height experiment. A comparison of this parameter between experiment and theory was carried out. The value of the comparison for the cases presented here lay mainly in the checking of the fast neutron cross sections.

Two experimental comparisons with theory are presented for two highly enriched cores with distributed burnable poison and in which the structural material is zirconium. The results for one of the cores is shown in Figure 1. The theoretical curves labeled Tape O were derived using no zirconium inelastic scattering while those labeled Tape 5 included zirconium inelastic scattering in the fast neutron cross sections. The theoretical curves are also separated with respect to the thermal constants, i.e., whether the Maxwellian or Wigner-Wilkins flux approximation is used. The theoretical calculations were carried out in the equivalent bare core, single buckling, few-group, P-1 approximation. The fast few-group constants were obtained from the MUFT code² and the thermal constants from the SOFOGATE code.³ The curves in the figure were obtained by computing a curve of reactiv-

Fig. 1. Cd-Covered U-238 Foil Activation

ity vs geometrical buckling and taking the derivative. It was found in the course of the calculations that if only one buckling (the material buckling) was used in MUFT, the buckling coefficient was quite sensitive to the number of few groups used. In addition these few-group calculations disagreed with the results obtained with P1MG,⁴ a multi-group code. However, this dependency and disagreement disappeared when the geometrical bucklings at which the calculations were made were used in MUFT. This latter observation is expected simply because the structure of the P1MG and MUFT equations and the few-group recipe in MUFT guarantee that it be so identically.

The experimental points represent the same experimental data but have different ordinates depending on whether Tape 0 or Tape 5 fast neutron cross sections are used for computing the effective delayed neutron fraction to be used in correcting the reactivity inferred from the period measurements. The experimental points are plotted at the critical experimental buckling. The vertical flags represent the maximum error in inferring the slope of the partial water height data as plotted on a critical water height vs (differential water worth)^{-1/2} graph. The horizontal flags represent the approximate error in inferring the total critical experimental buckling.

Fig. 1. Reactivity Buckling Coefficient vs Buckling Core M9

As can be seen in Figure 1, the reactivity buckling coefficient is quite sensitive to the presence or absence of zirconium inelastic scattering and is much less sensitive to the assumption of Maxwellian or Wigner-Wilkins thermal constants, respectively. The best agreement with experiment is obtained using fast neutron cross sections which include zirconium inelastic scattering and using Wigner-Wilkins thermal constants.

¹ PWR-FA Staff, "Slab Experiments-IV," PWR-Ph-151 (December 1956).

² H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan, "MUFT-4, Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (July 1957).

³ H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged Over a Wigner-Wilkins Flux Spectrum—Description of the SOFOCAT Code," WAPD-TM-39 (January 1957).

⁴ H. Bohl, Jr., E. M. Gelbard, G. R. Culpepper, and R. F. Buerger, "P1MG—A One-Dimensional Multigroup P-1 Code for the IBM-704," WAPD-TM-135 (July 1959).

Reactivity and Activation Measurements and Calculations in Highly Enriched U-Zr-H₂O Critical Assemblies Between 65 and 460°F. L. O. Herwig, W. F. Vogelsang, J. J. Kepes, E. L. Humez, W. A. Vogele, and T. M. Ryan (W-BAPD). Reactivity, temperature coefficient of reactivity, and neutron activation measurements have been made on highly enriched U-Zr-H₂O slab critical assemblies between 65° and 460°F and are compared with calculated results from several reactor design models.

Two cores, each four module boxes wide but of different length, were assembled using zirconium module boxes and fuel plates as described below. The fuel plates are 9 w/o enriched U-Zr alloy, 0.060 x 1.386 x 30 in. long. They are clad with 0.015 in. of Zircaloy-2 and are located by Zircaloy-2 clips welded to two opposite sides of a module box. The outside dimensions of the square module box are 1.636 in. with an effective wall thickness of 0.053 in. The average metal-to-water ratio is 1.2 with an average fuel loading of 168 g/liter. The cold and hot critical sizes were about 11 and 15 bundles long, respectively. At room temperature this core has a positive temperature coefficient which decreases to zero at about 150°F and increases negatively thereafter. For a fixed core size eleven bundles long, this temperature coefficient characteristic gives rise to a clean critical condition (control rods out) at two temperatures, differing by about 200°F.

Results from three reactor design models are compared with measured results at three temperatures, 65°, 232°, and 460°. The development of one of these models [Model (3)] has been motivated to account for effects caused by large spatial spectrum changes in regions near lumped absorbers and other large heterogeneities encountered in critical assemblies. In the present uniformly loaded critical assemblies, the core-reflector interface provides a region in which the thermal neutron spectrum changes rapidly.

Model (1) is based on the few-group, single thermal group diffusion theory approximation for the neutron transport equation as solved by the WANDA-5 program.¹ The three group fast constants were calculated regionwise using homogenized number densities for the core materials and the MUFT-5 program.² The thermal constants were calculated regionwise using the KATE program,³ a Wigner-Wilkins hardened neutron energy spectrum, and flux-weighted number densities for the lumped core materials. Flux-weighting factors were calculated by means of the FLIP program⁴ using the Double P-3 approximation to the transport equations.

Model (2) is an earlier version of the Model (1) calculation. It is identical with Model (1) except for the thermal constants which are calculated using a Maxwellian thermal neutron energy spectrum rather than the Wigner-Wilkins spectrum.

Model (3), the Mixed Number Density design model,⁵ utilizes a mixed system of thermal constants together with a technique for calculating number densities rather than flux. Calculations for this model are accomplished using the WANDA-5 program. This model is employed to account empirically for the variation in thermal spectrum with position in critical assemblies having large heterogeneities.

Table I shows the reactivities of each of the nuclear models as a function of temperature and compares them to the experimental results.

TABLE I
Comparison of Measured and Calculated Reactivities

Temp (°F)	Experiment K_{eff}^a	K_{eff} and Calculational Bias ^{b, c}			Temp Coef ^c	
		Model (1)	Model (2)	Model (3)	Model (1)	Meas
65	1.0065	1.0063(-2)	1.0099(+34)	1.0061(-4)	+0.89	+0.97
232	1.0041	1.0048(+7)	1.0060(+59)	1.0100(+19)		
459	1.0023	1.0032(+9)	1.0089(+66)	1.0048(+25)	-3.1	-3.1

^a $\bar{B}/B - 1.27$

^b $K_{eff}(\text{calc}) - K_{eff}(\text{expt})$

^c in units of $10^{-4} \Delta K$

Table II compares the measured and calculated ratios of the thermal neutron activation at a core position approximately 0.5 cm from the core-reflector interface to the thermal activation at the center of the reactor core.

TABLE II
Thermal Neutron Activation Ratios

	65° F	460° F
Experiment	2.42	2.76
Mixed Number Density	2.44	2.95
Maxwellian	2.47	2.98
Wigner-Wilkins	2.07	2.50

The comparisons of the measured and calculated reactivities and activation ratios shows in the temperature range from 65° to 460° F that (a) the calculational bias for each model is relatively constant with temperature, Model (1) having somewhat better consistency; (b) Models (1) and (3) have a lower calculational bias than Model (2); and (c) models (2) and (3) are superior to Model (1) in predicting the thermal neutron activation near the core interface. In the application of a single nuclear model to calculate both reactivity and thermal neutron activation, the Mixed Number Density design model, Model (3), is superior to the other two for these cores.

¹ O. J. Marlowe and M. C. Suggs, "WANDA-5—A One-Dimensional Neutron Diffusion Equation Program for the Philco-2000 Computer," WAPD-TM-241 (to be issued).

² H. Bohl, Jr., "MUFT-5—A Fast Neutron Spectrum Program for the Philco-2000," WAPD-TM-218 (June 1960).

³ H. J. Amster and J. B. Callaghan, "KATE-1—A Program for Calculating Wigner-Wilkins and Maxwellian Averaged Thermal Constants on the Philco-2000," WAPD-TM-232 (to be issued).

⁴ B. L. Anderson, et al., "FLIP—An IBM-704 Code to Solve the P_L and Double P_L Equations in Slab Geometry," WAPD-TM-134 (1959).

⁵ R. J. Breen, "A One Group Model for Thermal Activation Calculations," (to be published in Nuc. Sci. and Engr.).

Analytical and Experimental Results for a Compact, Zirconium Hydride Moderated Critical Assembly, L. I. Moss, C. M. Podeweltz, R. L. Randall, J. L. Shapiro, and R. E. Wimmer (AI). The SNAP Critical Assembly-2 (SCA-2) is a solid, homogeneous reactor used as a research and development tool in the design of compact power reactors. The core is built up of 7-in. diam $\frac{1}{2}$ -in. thick plates of

TABLE I
Experimental vs Diffusion Theory K_{eff} 's

No. of fuel plates	Radial Reflector	Axial Reflector		K_{eff}	
		Lower	Upper	Measured	Calculated
		1 in. Ni +			
10	2.5 in. Be ^a	1.5 in. Be	2.32 in. Ni	1.0006	0.9960
9	2.5 in. Be	2.5 in. Be	2.35 in. Be	1.0025	0.9801
10	2.5 in. Be	2.5 in. Be	0.8 in. Be	1.0011	0.9852
10 ^b	2.75 in. Be	1.75 in. Be	1.6 in. Be	1.0031	0.9896

^a Radial reflector segments here did not fit together as tightly as in the other assemblies.

^b Ten 0.15-in. thick Be plates alternated with fuel in this core.

U-ZrH_{1.73}. Each plate contains an average of 280 g of fully enriched uranium and 6.1×10^{23} hydrogen atoms/cc. Various reflector materials including beryllium, nickel, and plastic have been used. Since this reactor has a clean, homogeneous core and simple cylindrical geometry it is well suited for comparing theory and experiment.

Fig. 1. Radial Power Density (U-235 Foil)

Table I compares the experimental and calculated K_{eff} 's for four critical configurations. The calculated values were obtained using the AIM-6, one-dimensional diffusion code¹ with a 16-group cross-section library obtained from Los Alamos.² The method of computation required agreement between K_{eff} 's computed in slab and cylindrical geometry. Equivalence in K_{eff} between reflected cores and bare cores (with zero flux boundary conditions) provided energy independent transverse buckling terms, $\left(\frac{2.405}{R_0}\right)^2$ and $\left(\frac{\pi}{H_0}\right)^2$ for each geometry.

Uranium foil activation was measured as a function of position throughout the core. Figures 1 and 2 show radial distributions for beryllium and nickel reflectors and an axial distribution for a nickel reflector. The corresponding diffusion theory distributions are represented by the dashed line curves.

Foil activation off-axis and away from the central plane showed that the fluxes are separable to within experimental error throughout the core. Only in the outer corner of the reflector did the activation increase (by about 40%) over the product of the separated fluxes.

Fig. 2. Axial Power Density (U-235 Foil)

The parameter of $\frac{\beta_{eff}}{\ell}$ was measured on an axially and radially nickel reflected assembly using a noise analysis technique.³ The measured value of $\frac{\beta_{eff}}{\ell}$, $1410 (\pm 10\%)$, compares with a calculated value of 1755. The value of ℓ was calculated as $4.5 \mu\text{sec}$ using the homogeneous $\frac{1}{V}$ absorber perturbation method of A. F. Henry.⁴ The value of β_{eff} was computed a number of ways, involving deletion of delayed neutrons and alteration of the multigroup fission neutron spectrum. In each instance the value of 0.0079 was obtained.

Isothermal temperature coefficients were measured from room temperature to 700°F . The average coefficient from 68°F to 300°F was $-0.2\text{c}/^\circ\text{F}$ ($\pm 10\%$) and from 300°F to 700°F was approximately $-0.3\text{c}/^\circ\text{F}$ ($\pm 10\%$). The calculated values over the same range were $-0.16\text{c}/^\circ\text{F}$ and $-0.18\text{c}/^\circ\text{F}$. The part of the coefficient due to the spectrum shift was calculated to be negligible.

¹ H. P. Flatt and D. C. Baller, "AIM-5, A Multigroup, One-Dimensional Diffusion Equation Code," NAA-SR-4694 (March 1960).

² G. E. Hansen and W. H. Roach, "Sixteen Group Cross Sections for the S_N Code," Internal Los Alamos Report N-2-753.

³ C. W. Griffin, "Measurement of the SRE and KEWB Prompt Neutron Lifetime Using Random Noise and Reactor Oscillation Techniques," NAA-SR-3765 (October 1959).

⁴ A. F. Henry, "Computation of Parameters Appearing in the Reactor Kinetics Equations," WAPD-142 (December 1955).

Critical Arrays of Neutron Interacting Units, L. W. Gilley and J. T. Thomas (ORNL).

A series of critical experiments using as many as 100 subcritical vessels of U-235-enriched uranyl nitrate solution has been performed with the units assembled in unreflected and unmoderated arrays. The problem of neutron interaction between units in storage arrays is a complex one for which an adequate theory does not exist. It is intended that the data presented here will support analyses of such interaction problems in general and will provide bases for safe storage and transportation of these particular vessels which are widely used in industry.

Each unit consisted of an aqueous solution of $\text{UO}_2(\text{NO}_3)_2$, enriched to 92.6% in U-235, having a concentration of 410 g U/liter and a specific gravity of 1.55 contained in a $5\frac{1}{2}$ -in. OD seamless polyethylene bottle approximately 48 in. long and of ~12-liter capacity. The volume-averaged ID was 4.67 in. The variables studied included the volume content of the bottles and their spacing and geometric pattern within an array. The bottles were arranged with their axes vertical and their bases in either a square or a

UNITS: POLYETHYLENE BOTTLES: $5\frac{1}{2}$ -in. O.D., ~12-liter CAPACITY.
SOLUTION: $\text{UO}_2(\text{NO}_3)_2$ AT 410 g of U per liter AND SPECIFIC GRAVITY 1.55, ENRICHED TO 92.6 wt. % IN U^{235} .

SQUARE PATTERNS

● SOLUTION HEIGHT = 22½ in.; 5.90 liters/unit, SINGLE TIER.
▲ SOLUTION HEIGHT = 33½ in.; 9.30 liters/unit, SINGLE TIER.
■ SOLUTION HEIGHT = 44½ in.; 12.76 liters/unit, SINGLE TIER.
○ SOLUTION HEIGHT = 44½ in.; 12.76 liters/unit, DOUBLE TIER.

TRIANGULAR PATTERN

△ SOLUTION HEIGHT = 44½ in.; 12.76 liters/unit, SINGLE TIER.

VERTICAL SPACING BETWEEN SOLUTION IN TWO TIERS WAS 5.6 in.

Fig. 1. Unreflected and Unmoderated Critical Arrays of Units Containing Aqueous Solution of Uranyl Nitrate

triangular pattern. Most experiments were performed with a single tier of bottles—that is, all bases were in a horizontal plane, although in one series two such tiers were arranged, one above the other, with the solution separated vertically by 5.6 in. The outer boundary of an array was either square or hexagonal depending upon whether a square or triangular pattern was used.

The number of units in a critical array as a function of their surface-to-surface separation is shown in Figure 1 for each of three different solution volumes. The results from the experiments with the double tier are plotted with ordinates equal to the number of units in one of the tiers (i.e., one-half of the total number at a particular spacing). In this manner a double tier is shown to have characteristics similar to a single tier of units but with extended solution height.

The following characteristics of these critical arrays are apparent from the data:

- 1) As the volume of solution per unit increases, the separation of a fixed number of units increases.
- 2) A triangular pattern of units requires a larger separation than a square pattern of the same number of a particular unit, i.e., a triangular pattern is more reactive.
- 3) For finite units arranged in a single tier there is a limiting separation which describes an array composed of an infinite number of units.

These results are directly applicable to storage and transportation of this particular unit. The surface-to-surface separation necessary to preclude criticality can be specified provided safety factors suitable to guard against reflection, moderation, and the chance commingling of arrays are chosen. Application of these data to other solutions or containers must depend on future theories or existing empirical formulations.

Critical Measurements for the KIWI-A Nuclear Propulsion Test Reactor,* J. C. Hoogterp and J. D. Orndoff (LASL). This presentation covers measurements in reactor mockups which contributed to the design of Kiwi-A—the first nuclear rocket propulsion test reactor. Initial surveys of critical parameters and fission power distributions, where flexibility was more important than geometric detail, were conducted in the "Honeycomb" machine. Results of these surveys permitted design of a propulsion test reactor save for tailoring of fuel-element loadings and controls. The "Zepo" machine, with fixed geometry and more realistic controls, was used to finalize these characteristics. A few zero-power measurements on the actual Kiwi-A reactor confirmed the data from Zepo.

Honeycomb is a split matrix of 3-in. square aluminum tubes forming a 6-ft cube when assembled. Numerous critical configurations consistent with acceptable reactor power production were obtained in this machine with an annular fuel core geometry of U-235 foil and graphite surrounding a moderating island. Beryllium and D_2O were examined as island materials and Be and graphite as external reflectors. Annular fuel thicknesses ranged from 2 to 6.5 in. with critical masses from 7 to 24 kg U-235. Cost determined the choice of graphite over Be for the reflector; simplicity of internal controls and of cooling led to choosing D_2O for the island.

The Zepo (Zero Power) machine with its graphite reflector, retractable annular core, and D_2O island had a

*Sponsor: H. C. Paxton

Fig. 1. Fission Distribution in Zepo and Kiwi-A

range of loadings from 17 kg U-235 with negligible excess reactivity, to 40 kg with \$10 excess reactivity. In the final Zepo configuration a portion of the fine-structure foil-graphite fuel was replaced by impregnated graphite to more nearly simulate Kiwi-A. Radial fuel-loading distributions which gave approximately "flat" fission density (from 40 to 230 mg U-235/cc) were adjusted for elevated temperature by means of detailed computations. Figure 1 inter-compares fission density and flux distributions in the final Zepo mockup with those observed in the Kiwi-A reactor. A 20% peak in the power distribution at the outside of the Kiwi-A core during Nevada Test Site operation is attributed to a lower reflector temperature than was assumed in calculations that referred to operating conditions.

Special Features of the Advanced Epithermal Thorium Reactor Critical Assembly, H. A. Morewitz, S. G. Carpenter, D. T. Eggen, and C. Myhre (AI). A split-table, honeycomb-type, critical assembly machine of modern, low-friction design has been built for use with U-233 assemblies in the intermediate and fast neutron spectra regions. Stability and reproducibility have been emphasized in the design so that accurate measurements may be obtained on small samples in the nuclear environments of interest. Special features include a multiregion, pseudospherical assembly; personnel shielding; safety fuses; and a high-precision oscillator system.

A multiregion pseudospherical assembly (Figure 1) has been designed which uses only 25 kg of U-233. A spherical U-233 test region is surrounded by a U-235 buffer region of about the same nuclear characteristics. A thorium

Fig. 1. Epithermal Critical Assembly - Multi-region Pseudosphere

shell decouples inner regions from thermal neutrons originating in the outer driver and reflector regions. The thorium shell also provides a gamma shield (to the rear) against radioactive daughters of U-232 (in U-233 fuel).

Two 6-in. thick iron shields provide gamma shielding at the loading faces. When the assembly separates, these shields may be driven between the halves, reducing a 25-r/hr field to less than 2 mr/hr, thus permitting manual loading.

Specially designed internal fuses back up fast-acting safety rods. Flat springs interspersed between driver- and reflector-region plates are compressed by split-bolts incorporating U-235 solder-fuse links. These fuse links separate at 120°C and permit the springs to expand the driver-reflector regions by approximately 50%, thus introducing \$6 reactivity shutdown. This action requires 15 msec and limits a 2% reactivity insertion excursion to 10 Mw-sec.

An improved oscillator system, having a sensitivity an order of magnitude better than other techniques, permits relative reactivity measurements in the nuclear environment of the test region. A mechanical oscillator alternately inserts one sample while withdrawing another from the core. A stainless steel rod, with two pockets for samples, fills the passage through the core and prevents neutron streaming.

A special instrument system is used with the oscillator. A sensitive, uncompensated ion chamber drives a vibrating-reed electrometer which in turn feeds a voltage to a pulse converter. Output of the converter feeds a transistorized, 200-channel time analyzer. All analyzer channels are spread across the oscillation cycle. After completion of the first cycle, subsequent cycles are stored in the corresponding equiphase channels. Over a number of cycles the coherent signals add constructively, while the noise fills all channels uniformly. Limiting reactivities as small as $10^{-8} \Delta k/k$ may be obtained.

Over-all instrument linearity and stability are better than 0.02%, and the instrument's dynamic range is greater than 4 decades. Its excellent statistics and lack of counting losses have made it the instrument of choice for reactor period and rod drop measurements.

The Japan Atomic Energy Research Institute Semihomogeneous Critical Assembly, K. Inoue, W. Y. Kato, K. Sumita, M. Iizumi, K. Nishimura, Y. Kaneko, S. Kobayashi, I. Kobayashi, S. Matsura, T. Kurosawa, S. Yasukawa, and T. W. Park (JAERI). The Japan Atomic Energy Research Institute (JAERI) has constructed a critical facility, the Semihomogeneous Critical Experiment (SHE), for studies on graphite moderated enriched uranium oxide thermal reactors which have as much as 20% to 30% of their fissions occurring in the resonance region. The SHE facility is operated at or about room temperature and will be used to obtain nuclear parameters necessary in the design of a high temperature graphite moderated power reactor as well as for generalized reactor physics studies.

The SHE facility is a two-half type machine whose tables, one movable and the other stationary, are 2.5 meters in width and 2.0 meters in length. On each of the tables are stacked approximately 1400 graphite tubes and rods 120 cm long and 6.5 cm OD. The tubes and rods are stacked in such a manner that the vertical cross section at the interface of the two halves shows a hexagonal array of hollow tubes in the center surrounded by solid rods. The distance across the flats of the hexagon is 2.4 meters. There are 397 hollow tubes in the central region which form the core of the reactor where the composition may be varied; the solid graphite rods around the core form the reflector.

The fuel for the facility is composed of homogenous mixtures of 20% enriched uranium (UO_2) and graphite in the form of cold pressed discs of 4.5-cm diam and 1-cm thickness with the graphite to uranium oxide weight ratio being 10. Discs of thorium oxide and graphite mixtures as well as of pure graphite are available to obtain varied core compositions. The fuel discs were made in the above fashion in order to simulate as closely as possible conceptual fuel elements of the high temperature power reactor and still maintain a high degree of flexibility over the composition of the core. These fuel and graphite discs are inserted into 5.5-cm OD graphite sheaths to form fuel rods for the core tubes. The half length of the core is variable to 60 cm using these sheaths. The graphite discs and structural graphite have a density of 1.7, while the fuel discs have a density of 2.1. Variation of the composition of the core is obtained by changing the combination or pattern of the discs in the sheaths. The lowest carbon to U-235 ratio obtainable in the core is 2226.

There are one control and three safety rods in each half of the facility which are hollow cylinders of either cadmium or boral construction. The control rod drive mechanisms have been designed so that it is possible to set and read position to ± 0.1 mm. The movable table's position has been found to be reproducible to about ± 0.01 mm in the closed position corresponding to a reactivity change of about ± 0.07 cents. The initial results indicate that reactivity measurements can be made with high accuracy due to the good reproducibility of the table and control rod positions.

The initial loading was for the purpose of understanding the characteristics of the facility with a simple composition and geometry and was accomplished in January 1961.

The first loading was an approximate right cylinder with a 100-cm length and 88-cm equivalent diameter. The sequence within each fuel rod was a fuel disk, 1-cm graphite disc, and a 0.5-cm graphite disc. This pattern was repeated for the first 50 cm of the sheath. Each rod contained 10.3 g of U-235. The remaining 10 cm was filled with graphite discs. This resulted in a core whose carbon to U-235 ratio was 5378 and whose carbon to U-238 ratio was 1349. The radial reflector thickness was about 80 cm, and the axial reflector thickness was 70 cm. The critical mass for this configuration was determined to be about 325 fuel rods or 3.4 kg U-235. With 333 fuel rods in the core the average reactivity worth of the cadmium control rods was found to be about 80¢ each. Axial and radial

neutron flux distributions throughout the reactor were made using Cu, In, Au, and Dy foils. Fission rate measurements were made using miniature U-235 fission counters. The reactivity effect of the gap produced by the separation of the tables was found to be about 70¢/cm near the closed position.

The initial measurements indicate that this facility is well suited for the study of clean core graphite systems since no structural materials other than graphite are used in the core. Most systems studied will be well reflected because of the thick radial and axial reflectors. In addition, the heterogeneity correction should be small because of the graphite uranium oxide mixture fuel discs.

Critical Experiments II

Analytical Techniques for Interpreting Epithermal Critical Experiments, A. V. Campise (AI). Interest in sodium-cooled reactors to operate in an intermediate neutron energy range between 100 ev and 1 Mev has led to design and construction of a multi-region, spherical critical assembly of a fast-slow type.¹ This critical assembly was constructed as part of the Advanced Epithermal Thorium Reactor program being conducted by Atomics International and sponsored by Southwest Atomics Energy Associates. The experiments to be performed with this critical assembly, called the ECE, required theoretical methods for proper interpretation of experimental data for reactor design. For this purpose a data processing system, called AETRA,² was developed.

AETRA incorporates multi-group, multi-region, modified diffusion theory equations (flux and adjoint) of AIM-6³ and perturbation theory equations of PERT.⁴ The programmed system (Figure 1) is capable of analyzing fissions and capture rates, reactivity coefficients, neutron flux, and importance spectra. Included in this data processing system are adjustment routines for introducing in-

cremental changes in the neutron group cross sections. Changes are made as indicated by differences between theoretical and measured values.

Experimental results of the ZPR-III-5, -9a, -11, and -12 cores⁵ have been studied using this data processing system. These cores were chosen from 11 different assemblies since central reactivity coefficient measurements have been published. Results of this study are listed on Table I. It is noted that good agreement was obtained in the critical radius, fission ratio, and reactivity coefficient comparisons. The differences were attributed to the inaccuracies in the neutron cross sections.

These results indicate that the theoretical methods incorporated into this data processing system are sufficiently accurate. This program will allow rapid evaluation of the ECE experiments which are currently in progress.

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²J. S. Temple, "(AETRA) Data Processing Program in Fortran for the 709 or 7090," Unpublished Atomics International Report (1960).

³H. P. Flatt, "(AIM-6) A Multi-Group, One Dimensional Diffusion Theory Code," Unpublished Atomics International Report (1960).

⁴H. P. Flatt, "(PERT) A Perturbation Theory Program," Unpublished Atomics International Report (1960).

⁵W. B. Loewenstein and D. Okrent, "The Physics of Fast Power Reactors; A Status Report," *Proc. Second Geneva Conf.* (United Nations, New York) 12 (1958), p. 16.

Fig. 1. Critical Experiments Flow Diagram

TABLE I
Comparison Between Measurements and Calculations on ZPR-III Cores

ZPR-III	N(238) N(235)	L/D	Critical Radius (cm)		Fission Ratios, F^{238}/F^{235}		Reactivity Coefficients U^{235} β /Mol	
			Meas	Calc	Meas	Calc	Meas	Calc
5	1.136	0.936	24.40	23.43	0.074	0.076	21.2	21.9
9A	3	Sphere	25.2	25.88	--	0.056	--	23.19
12	3.5	0.87	28.8	28.8	0.044	0.051	20.9	20.71
11	7	0.88	31.8	32.77	0.036	0.040	19.1	16.79

Reactivity and Spatial Distributions in Multi-Region Lattices: Comparison Between Analysis and Experiment, W. J. Eich and W. P. Kovacik (WAPD). Critical experiments involving single- and multi-region cores of low enrichment UO_2 fuel rods moderated by light water are being continued by the Westinghouse Atomic Power Department. This summary deals with that aspect of the program involving the correlation of analytical results with experimental measurements of criticality and spatial power and flux distributions.

Stainless steel clad fuel rods, enriched to 1.6%, 2.7% and 3.7% were utilized in lattice loadings having 2.5:1 and 4.5:1 W/U ratios. The cores assembled included cylindrical and rectilinear cores, both single enrichment and multi-region.

The control rods, which had fuel rod followers of the appropriate enrichment, were fully withdrawn from the cores during operation. Reactor control was accomplished, where necessary, by limiting moderator height. Among the experimental methods used were fuel rod activations and foil (gold and U-238) scans in the moderator. The fuel rod scans provide radial power distributions and axial bucklings. The latter are required as input, on a region-wise basis, where necessary, to the various digital computer analyses in which the spatial mesh describes the critical configurations in a radial plane.

The fission activity and Np-239 activity in U-238 foils are a measure of the very fast flux and resonance flux shapes, respectively. The activity of gold-covered gold is a measure of the thermal flux shape, and the difference between the bare and gold-covered gold activities is a measure of the resonance flux-shape around 4.9 ev.

TABLE I

Eigenvalues for Multi-Region Cores Calculated on the Basis of Few-Group Diffusion Theory

Core Description and Designation

- A: W/U = 2.5/1; Single region cylinder; 2997 - 2.7% fuel rods
 B: W/U = 2.5/1; Three region cylinder; loading from the center: 1313-1.6%, 1604-2.7%, 1728-3.7%
 C: W/U = 2.5/1; Three region cylinder; loading from the center: 3481-1.6%, 3412-2.7%, 3648-3.7%
 D: W/U = 4.5/1; Single region cylinder; 1264-2.7%
 E: W/U = 4.5/1; Two region cylinder; loading from the center: 2385-1.6%, 2276-2.7%

Eigenvalue Tabulation

Core	Conventional Thermal		MND
	2 Group	4 Group	4 Group
A	0.9906	0.9909	0.9933
B	0.9919	0.9910	0.9922
C	0.9933	0.9933	0.9953
D	0.9966	0.9960	0.9989
E	0.9946	0.9939	0.9956

Analytical methods relied on the MUFT-4¹ and SOFO-CATE² codes to provide input for few-group diffusion and transport codes, e.g., the WANDA³ code. The PIMG⁴ multi-group diffusion code has also been applied to core configurations amenable to one-dimensional analysis; the all-important self-shielding factors in both the PIMG and MUFT codes have been determined on the basis of a semi-empirical fit⁵ to similar critical experiments; the bucklings required to apply this fit as well as the MUFT-PIMG "resonance buckling" parameters were the ma-

terial bucklings appropriate to each enrichment in each lattice. Finally, the Mixed Number Density (MND) model⁶ was applied to the thermal group input of both few- and multi-group diffusion analyses.

The underlying feature of the analysis is a combination of a simple model⁷ of the equilibrium reactor cycled with a multi-group scheme as embodied in the MUFT-SOFO-

Fig. 1. Core B: Normalized Power Shapes vs Experiment

CATE system. The results for eigenvalues and spatial distributions are rather consistent over a wide range of critical configurations.

Table I provides a representative list of eigenvalues as calculated by few-group diffusion codes and a description of the cores to which they correspond. Note that the "average" eigenvalue is slightly larger for the larger W/U ratio. Figure 1 compares various methods of analysis with an experimental power distribution in a small, three-region core. It should be noted that any radical improvement in analytic power shape due to application of the MND model in a clean critical core is restricted to *thin* outer regions in a multi-region core. Nevertheless, a multi-group analysis employing the MND model has provided the best agreement with experiment; in fact, the only significant improvement that could be made would be a correction of the power depression near the core-reflector interface for transport effects. The analytic flux shapes are also in reasonable agreement with the corresponding foil distributions.

¹H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan, "MUFT-4—Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (1957).

²H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFO-CATE Code," WAPD-TM-39 (1957).

³O. Marlowe, et al., "WANDA—A One-Dimensional Few Group Diffusion Equation Code for the IBM-704," WAPD-TM-28 and Addenda.

⁴H. Bohl, et al., "PIMG—A One-Dimensional Multi-Group P₁ Code for the IBM-704," WAPD-TM-135.

⁵W. H. Arnold, Jr., "Critical Masses and Lattice Parameters of H_2O - HO_2 Critical Experiments: A Comparison of Theory and Experiment," YAEC-152.

⁶R. J. Breen, "A One Group Model for Thermal Activation Calculations," *J. Nuc. Sci. and Eng.*, 9 (1961), p. 91.

Analysis of Critical Experiments with Slightly Enriched UO_2 Clusters in Heavy Water, J. D. Cleary, B. Jennings, F. L. Langford (WAPD), and W. H. Arnold, Jr. (NUS). The Carolinas Virginia Tube Reactor Critical Assembly is described in detail in a companion paper.¹ Experiments with this facility are intended in part as a check on the analytical methods developed for the design of this power reactor.

Critical sizes of this assembly were computed through four-group diffusion theory using fast constants fitted to agree with the Fourier transform of the Boltzmann equation in a P-1 Greuling-Goertzel approximation. Thermal constants were obtained with the SLOP-1 code.² The results of calculations were found to agree with measurement to within 1% in k.

Two-dimensional diffusion theory calculations were performed to obtain gross power shapes in the core using group constants obtained as above. Comparison with measured power shapes, as presented in the second companion paper,³ shows reasonable agreement.

Power shapes within a fuel bundle were calculated for two arrangements of enrichment using P-3 approximation.⁴ They are presented in Figure 1, together with measured values, and show excellent agreement.

Parameters relating to resonance escape probability, fast effect, and conversion ratio, namely ρ^{25} , δ^{25} , δ^{23} , were

TABLE I
Calculated and Measured Cadmium Ratios of U-235 and U-238

Fuel Enrichment (w/o)	U-235		U-238	
	Calculated	Measured ^a	Calculated	Measured ^a
1.1	16.85	19.5 ± 0.5 21.7 ± 0.6	2.52	2.46 ± 0.02 2.49 ± 0.03
2.0	10.63	12.60 ± 0.13 17.14 ± 0.34	1.88	2.08 ± 0.06 1.96 ± 0.04

^aMeasured in center and outside fuel rod of a 19 fuel rod cluster.

obtained by hand calculation of the cadmium ratio of U-238 using resonance integral data from Hellstrand,⁵ corrected for Dancoff effects as described by French.⁶ The values of cadmium ratio of U-235 were calculated from MUFT.⁷ Measurements of ρ^{25} , δ^{25} , δ^{23} followed the standard cadmium ratio techniques⁸ which involved placing bare and cadmium covered U-235 and U-238 foils at 3-in. intervals inside one of the fuel rods and normalizing by fuel rod activity. Measured and calculated results are presented in Table I.

Calculated reactor neutron spectrum was compared with the experimental results through Au and Lu activations. A fair agreement was found between calculated Au-Cd ratio and the measured value of 6.94 adjacent to fuel clusters and 5.48 in the moderator, halfway between tubes. The neutron spectral index was calculated at different positions in the unit cell and the core using thermal spectra obtained with the SLOP-1 code. Measurements of this quantity were done with Lu foils.

Fig. 1. U-235 Fission Rate within a Cluster

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- ²H. Bohl, E. Gelbard, P. Buerger, and G. Culpepper, "SLOP-1, A Thermal Multigroup Program for the IBM-704," WAPD-TM-188.
- ³J. A. Roll, "Power and Flux Distributions in the Carolinas Virginia Tube Reactor Critical Experiments," TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 4, 1 (1961), p. 108.
- ⁴R. R. McCreedy and D. B. Vollenweider, "An IBM-704 Program for the Solution of the Neutron Transport Equation in Fifty Concentric Cylindrical Annuli by the Weill Method, Program I₂," APEX-468.
- ⁵E. Hellstrand, "Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries," *J. Appl. Phys.*, 28 (1957), pp. 1493-1502.
- ⁶R. J. French, "The Dancoff Correction to the Resonance Absorption in Close-Packed Cylindrical Fuel Rod Lattices," YAE-149 (1960).
- ⁷H. Bohl, Jr., E. M. Gelbard, and G. H. Ryan, "MUFT-4—Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (1957).
- ⁸D. Kein, A. Z. Kranz, G. G. Smith, W. Baer, and J. DeJuren, "Measurements of Thermal Utilization Resonance Escape Probability and Fast Effect in Water-Moderated, Slightly Enriched Uranium and Uranium Oxide Lattice," *J. Nuc. Sci. and Eng.*, 3 (1958), p. 403.

Natural Uranium—D₂O Bucklings over an Extended Range of Pitch and Fuel Assembly Size, T. J. Hurley, Jr., H. R. Fike, and G. F. O'Neill (DuP-SR). Buckling measurements on D₂O moderated natural uranium lattices have, in the past, generally been limited to high buckling regions and,

consequently, to fairly small lattice pitches. The few bucklings that have been available at the larger lattice pitches, as well as most of those at smaller lattice pitches, have been obtained using exponential or substitution techniques rather than critical lattice studies.

A series of critical measurements on full-scale lattices has been made over an extended range of lattice pitch and fuel assembly size in the 16-ft diam Process Development Pile (PDP) at the Savannah River Laboratory. The lattices measured were triangular with a pitch that ranged from 7.00 to 21.00 in. The fuel assemblies were natural uranium metal rods, 1 in. in diameter and clad with 0.032 in. of aluminum, which were used singly or in clusters of 3, 7, and 19 rods.

The results for the single rod measurements, corrected to 99.75 mol % D₂O, are shown in Figure 1. This plot illustrates the generally good agreement of these measurements with previous work. The use of full-scale criticals, however, allows the present results to be given with a considerably smaller error than that quoted in the earlier work.

The results for the clustered rod lattices are given in Table I. These experiments extend well beyond the range of previous work so that no comparisons with existing data are possible. However, the full-scale critical techniques employed in these measurements were identical with those used in the single rod lattices, and hence the accuracy should be similar.

TABLE I
Lattice Buckling for Natural Uranium Rod Clusters

Fuel Assembly		Lattice			Buckling (m ⁻²)
No. Rods Per Assembly	Mass of Assembly (kg/cm)	Lattice Pitch (in.)	Moderator-U Vol Ratio Per Lattice Cell	Moderator Purity (Mol % D ₂ O)	
1	0.0954	7.00	53.05	99.70	6.41
		8.08	71.13	99.70	5.21
		9.33	95.23	99.70	3.99
		12.12	161.53	99.69	2.18
3 ^a	0.2862	7.00	16.89	99.70	7.56
		9.33	30.95	99.71	7.06
		12.12	53.05	99.71	4.99
		14.00	71.13	99.70	3.72
7 ^a	0.6677	9.33	12.58	99.71	6.01
		12.12	22.05	99.76	6.13
		14.00	29.80	99.76	5.17
		18.52	53.05	99.75	2.86
		21.00	68.55	99.75	2.10
		14.00	29.80	99.30	4.60
7 ^b	0.6677	18.52	53.05	99.31	2.12
		18.52	53.05	99.73	3.22
19 ^a	1.8124	14.00	10.23	99.72	3.39
		18.52	18.79	99.73	2.99

^{a,b}Center-to-center rod spacing: ^a = 1.5 in.; ^b = 2.0 in.

Fig. 1. Intercomparison of Buckling Measurements of Natural Uranium Rods (Nominally 1-in Diam) in D₂O

Heterogeneous Uranium Metal-Graphite Critical Experiments, C. A. Guderjahn (AI).

A series of clean heterogeneous critical experiments was carried out to provide information for normalizing and improving sodium graphite reactor calculations. The critical assembly consisted of a large cylindrical pile of reactor grade graphite 8 ft high and 12 ft in diameter. Fuel elements were placed vertically in a triangular array. The elements consisted of thin-walled aluminum tubes containing 1.97% enriched uranium metal hollow slugs, each 1-3/4 OD and 1 in. ID. The assembly was axially unreflected and had effectively an infinite reflector thickness in the radial direction. Eight "Y" shaped cad-

TABLE I
Experimental and Calculated Critical Radii for 1.97%
Enriched Uranium Rods in Graphite

Pitch (in.)	Experimental		Calculated		% Error in k of Calculations
	Critical No. of Elements	Critical Radii (in.)	Critical No. of Elements	Critical Radii (in.)	
6.9	30.8	20.2	29.8	19.9	+ 0.4
8.0	22.5	20.0	23.1	20.2	- 0.3
10.6	18.4	23.9	20.2	24.9	- 1.4
12.0	18.8	27.4	22.2	29.7	- 2.5
13.9	23.2	35.1	29.1	39.2	- 2.8
16.0	34.4	49.4	57	63.4	- 4.3

mium-silver alloy control rods were used for controlling the assembly.

The fuel was loaded in successive concentric rings, thus maintaining the core as nearly circular as possible. Table I gives the room temperature critical number of fuel elements and critical radii for several triangular lattice pitches.

Criticality calculations were made, and the results are shown in Table I. The calculations were performed in the following manner.

- 1) Thermal constants were obtained by using an S_4 cylindrical cell transport code for the 7090 IBM computer to calculate the thermal flux distribution in a cell. The cross sections were Maxwellian averaged.
- 2) Fast group constants were obtained from the MUFT-4 code which performs the cross section averaging over the fast spectrum but assumes a homogeneous cell.
- 3) Critical radii were calculated using the constants from (1) and (2) above in a two-group, two-region (AIM-5) reactor code.

The calculations compare well with experiment for the smaller lattice spacings, but the agreement becomes poorer as the lattice pitch increases. The error in the calculations appears to arise from the assumption of a homogeneous cell in the process of obtaining the fast group constants. This assumption becomes more unrealistic as the lattice pitch increases.

Critical Experiments on Enriched Uranium Stainless Steel Water Moderated Lattices, L. M. Welshans and K. M. Johnson (MARTIN).

A series of 12 cold, clean critical cores has been studied under the Army Nuclear Power Program in conjunction with the Pressurized Water Reactor Code Development task. The purpose of the experimental program is to supply data on highly enriched uranium, water moderated systems using stainless steel as cladding material.

The results can be used for checking the validity of the physical models and codes developed for this type of reactor. In some cases the data are of value in determining the physical model used in analytical calculations.

Critical masses for rectangular arrays of flat plate cores were determined over the following range: U-235, 7.1 to 24.5 kg; bundle thickness, 32 to 44 mils (corresponding to metal-to-water ratios of 0.243 and 0.358, respectively). The burnable poison concentration varied from 0 to 260 g of natural boron.

Flux measurements were made with thorium, cadmium covered gold, and dysprosium foils to provide buckling data at various neutron energy levels. Fine flux structure was investigated by intracell and thermal activations with dysprosium foils.

Pulsed neutron techniques were used to measure neutron lifetimes on most of the cores.

Various codes, adapted to an IBM-709, were used extensively in reducing the buckling, temperature coefficient, and pulsed neutron data to least-squares fit values.

The temperature coefficient, percent subcadmium fissions and material worths were measured on each core to furnish additional data for analytical comparison.

Experimental comparisons of the various assemblies include critical mass as a function of boron content or metal-to-water ratio and buckling as a function of boron content.

The method of assembling fuel bundles with the pertinent dimensions is shown in Figure 1 and tabulated in Table I.

The experimental results are summarized in Table II.

Fig. 1. Typical Fuel Bundle

TABLE I
Physical Characteristics and Composition of a Fuel Element

Core No.	Bundle Thickness Z (in.)	Shim Steel Thickness X (in.)	Boron Steel Thickness Y (in.)	Total Stainless Steel Weight (g)	Boron Weight (g)	M/W Volume Ratio
454	0.0434	0.0070	0.0058	6264	10.40	0.371
404	0.0394	0.0050	0.0058	5661	10.40	0.330
354	0.0334	0.0020	0.0058	4886	10.40	0.273
453	0.0418	0.0070	0.0042	6054	7.57	0.354
403	0.0378	0.0050	0.0042	5451	7.57	0.314
353	0.0318	0.0020	0.0042	4676	7.57	0.259
452	0.0428	0.0080	0.0032	6139	5.06	0.364
402	0.0388	0.0060	0.0032	5602	5.06	0.324
352	0.0328	0.0030	0.0032	4761	5.06	0.286
450	0.0436	0.0100	0	6280	0	0.373
400	0.0396	0.0080	0	5726	0	0.332
350	0.0336	0.0050	0	4902	0	0.275

TABLE II
Experimental Results

Core No.	Critical Mass, 20°C		Average Fuel Worth (% ΔK/K per g U-235)	Average Boron Worth, % ΔK/K Per g B	Cd Covered Gold Buckling (m ⁻²)	Temperature Coefficient (% ΔK/K-C°) x 10 ⁵ at 35°C
	U-235 (kg)	Boron (g)				
454	24.508	260.67	0.00130	-0.0575	0.01000	-8.34
404	22.019	234.46	0.00181	-0.0670	0.01106	-8.67
354	19.940	212.35	0.00198	-0.0709	0.01168	-9.64
453	17.507	134.63	0.00130	-0.0955	0.01280	-9.52
403	15.834	121.98	0.00137	-0.0999	0.01307	-9.63
353	14.192	109.34	0.00211	-0.1128	0.01448	-11.70
452	14.388	73.97	0.00170	-0.1066	0.01407	-11.07
402	13.484	69.45	0.00177	-0.1447	0.01487	-12.38
352	11.732	60.20	0.00212	-0.1159	0.01601	-10.30
450	8.616	0	0.00230	---	0.02032	-10.90
400	8.234	0	0.00164	---	0.02017	-15.94
350	7.085	0	0.00336	---	0.02162	-12.50

Critical Experiments III

Critical Experiments for the Preliminary Design of the Argonne High Flux Reactor, Part A, J. W. L. DeVilliers, Q. L. Baird, J. Juliano, C. N. Kelber, R. Kiyose, and K. E. Plumlee (ANL). The maximum available neutron flux, the flux to power ratios, and various reactivity effects for the preliminary design of the Argonne High Flux Reactor (AHFR)¹ were determined for two core mockups in order to supplement the theoretical data.

The critical assembly consisted of a 100-liter, 50.8-cm high, annular fuel region enclosing an 8.2-liter, 12.7 x 12.7-cm light water Internal Thermal Column (I.T.C.). The external radial reflector consisted of beryllium blocks stacked on plexiglas spacers to yield approximately 90% Be, 10% H₂O, and plexiglas. The axial reflectors were effectively infinite light water. The critical masses for the two cores were 3.64 and 7.25 kg 93.5% enriched U-235, respectively, contained in 40 fuel clusters.

Fig. 1. AHFR Critical Assembly Core

Gold and U-235-Al foils were used to measure the axial and radial flux and fission rate distributions in each of the assemblies. The effect on these distributions of a 0.635-cm aluminum liner, placed along the periphery of the I.T.C., or the introduction of a central stainless steel thimble (3.81-cm diam, 0.056-cm wall thickness) was also determined. A fission rate map was obtained at 37 radial points over a quadrant of the core with U-235-Al foils.

The total power developed in the core during the calibration run was obtained by integrating the U-235-Al flux map using a five point quadrature formula² for each cluster. The results of these measurements are listed in Table I. A peak flux of 3×10^{15} n/cm²-sec⁻¹ at 100 Mw should be obtained throughout the useful life of these cores.

The beryllium radial reflector had a worth (vs H₂O) of $+15.5 \times 10^{-2} \rho$. The worth of the top reflector was $+0.4 \times 10^{-2} \rho$ for c-1; however, in c-2 no clear-cut value was obtained. The worth of a 10.13-cm diam beam tube, piercing the beryllium reflector up to the fuel face was $-0.3 \times 10^{-2} \rho$. The void coefficient in the I.T.C. was constant at $+0.68$ (c-1) and $+1.0$ (c-2) $\times 10^{-2} \rho$ per v/o void (percent of total core volume) to about 3% total void fraction, becoming negative at 5%. In the fuel region the void coefficient was -0.45 (c-1) and -0.47 (c-2) $\times 10^{-2} \rho$ per v/o void.

The "gross" average temperature coefficient for the range 20° to 60°C was about the same in each core, $-0.24 \times 10^{-3} \rho$ °C. The control rod used was found to decrease in worth by 0.5 - 1.0% of total value for each degree rise in temperature in the region covered.

Measurements were made of the worth of stainless steel (304) uniformly distributed in the fuel zone. These gave a value of $-0.38 \times 10^{-5} \rho/g$. The fuel worth in the fuel zone ranged from $+0.78$ to $+1.28 \times 10^{-4} \rho/g$ for c-1 and was roughly a factor of eight lower for c-2. The worth of cadmium in the fuel zone of c-1 ranged from -3.6 to $-1.8 \times 10^{-3} \rho/cm^2$, with the value of $-3.6 \times 10^{-3} \rho/cm^2$ being that at the interface of fuel and internal thermal column.

¹L. E. Link, et al., "Argonne High-Flux Research Reactor-AHFR Conceptual Design Study," ANL-5983.

²J. C. P. Miller, "Numerical Quadrature over a Rectangular Domain in Two or More Dimensions," *Mathematics of Computations*, 14 (1960), p. 13.

TABLE I

Power Distribution and Peak

2200 m/s flux per watt for the AHFR critical assemblies

Item	3.6408 kg core	7.251 kg core	Conditions
Average power density (taken at 1 watt total power)	0.01 watt/liter	0.01 watts/liter	All
Peak power density (at inner boundary of fuel)	0.0171 watts/liter 0.0164 watts/liter --	0.0226 watt/liter 0.0218 watt/liter 0.0217 watt/liter	no liner, no central thimble no liner, with central thimble with liner, no central thimble
Peak flux in I.T.C. at 1 watt total power	2.9×10^7 2.7×10^7	3.1×10^7 3.4×10^7 2.9×10^7	no liner, no central thimble Al liner in, no central thimble thimble in center, no Al liner

Critical Experiments for the Preliminary Design of the Argonne High Flux Reactor, Part B, J. O. Juliano, C. N. Kelber, and K. E. Plumlee (ANL). Theoretical neutron flux-distribution of a homogenized idealization of the Argonne High Flux Critical (AHFCR) core C-1 was made and compared with the experimental measurements.

The actual spectrum near the reflector-core interface is obviously a rapidly varying function of the space coordinates. As shown by C. Kelber and P. Kier¹ and P. Michael,² the influence of the "soft" neutrons entering the core from the reflector should be limited to a region of the order of the relaxation length of such neutrons in the core which in C-1 was of the order of 3.2 cm. Thus, it can be expected that the calculations will not give good results for the cadmium ratios of various absorbers within 3 cm of the interfaces.

The predicted flux distributions, together with the microscopic absorption cross sections (given in Table I) of the various indicators were used to estimate the cadmium ratios. In making the estimate, that part of the thermal group activity which stems from neutrons whose energies are above the cadmium cut-off was added to the fast group activity to compute the total epicadmium activity. The group energy ranges were: thermal, 0-0.625 ev; epithermal, 0.625 ev - 9.12 kev, 9.12 kev - 1.353 Mev, and 1.353 Mev - 10 Mev. The correction is made under the weak assumption that the upper end of the thermal spectrum and the lower end of the fast spectrum are both proportional to 1/E and join smoothly.

Since the cadmium cut-off was assumed to be 0.53 ev,³ the estimated cadmium ratio, CR, is given by

$$CR = [1 + (g \sigma_{th} \phi_{th} / \sigma_e \phi_e)] / \{1 + \{0.086[\sigma_e - \phi_e(\text{res})] / \sigma_e\}\},$$

where g is the foil flux-depression factor, σ_{th} and σ_e the thermal and epithermal group absorption cross sections respectively, and $\sigma_e(\text{res})$ the resonance capture cross section ignoring the $1/v$ capture.

The calculated CR values for C-1 and experimentally measured CR values for cores C-1 and C-2 including their internal thermal columns are listed in Table II.

The agreement is qualitative but is probably within the combined range of theoretical and experimental errors.

¹C. Kelber and P. Kier, "Few Group Analysis of D₂O-U-235 Assemblies," *J. Nuc. Sci. and Eng.*, 8, 1(1960), p.1.

²P. Michael, "Thermal Neutron Flux Distributions in Space and Energy," *J. Nuc. Sci. and Eng.*, 8, 5(1956), p. 426.

³N. B. Stewart and G. B. Gavin, "Activation Experiments," J. R. Stehn, ed., UC-81 (September 1958), Ch. 6.

TABLE I
Calculated Microscopic Cross Sections of Indicators Used
in Spectral Measurements

Indicator Foil	Neutron Energy Group			
	Thermal		Epithermal	
	C-1	ITC	C-1	ITC
	(barns)	(barns)	(barns)	(barns)
Uranium - 235	350.2 (fission)	474.7 (fission)	26.73 (fission)	35.8 (fission)
Manganese	8.45	11.03	1.1	1.5
Gold	63.26	83.08	151.6	202.9
Indium	124.0	162.8	122.9	164.5

TABLE II
Calculated and Experimental Cadmium Ratios for the AHFCR

Indicator Foil	C-1		ITC		C-2
	Calculated	Experimental	Calculated	Experimental	Experimental
Uranium - 235	19.0±1.7	16.1	82±4	72.	3.6
Manganese	11.6	9.6	45.6	55.	4.8
Dysprosium	--	64.	--	312.	17.
Gold	1.6	2.2	3.6	10.5	1.65
Indium	2.44	3.3	7.24	14.1	2.0

PM-1 Zero Power Test Experimental Studies, H. B. Rosenthal (MARTIN). The PM-1 Zero Power Test Program (PMZ-1) was conducted to provide experimental data to confirm and aid in the analytical design of the Army Nuclear Power Program PM-1 reactor core. The PMZ-1 mockup of the PM-1 is a cylindrical, heterogeneous, light-water-moderated, unpressurized core containing approximately 29 kg of U-235 as highly enriched uranium. The 0.5-in. OD tubular fuel element is a stainless steel clad, UO_2 -stainless steel cermet with 30 in. of active length and a 0.042-in. thick wall. In addition to the fuel tubes, the core includes 0.5-in. diam borated stainless steel rods for lumped burnable poison and six stainless steel clad Eu_2O_3 , Y-shaped control rods. Critical experiment and scram rods are cylindrical stainless steel clad B_4C rods which travel inside fuel tubes.

The program consisted of essentially three phases. Nuclear characteristics of the initial PM-1 configuration were determined in the first phase. Based on the data obtained, it was determined that the configuration had to be modified to satisfy design requirements. The second phase was pursued to establish what configuration would satisfy these requirements. Analysis of data from the second phase led to the reference design configuration which was studied in the third phase.

Preliminary to the studies of the initial configuration, a clean, unperturbed, minimum-size core was investigated. The minimum-size core contained 127 fuel tubes (5011 g U-235) to establish criticality. Some preliminary studies of the effects of lumped poison rods were also performed on this small core.

TABLE I

Total Core Reactivities and Critical Six Y-Rod Bank Positions

Core ^a	Reactivity ($\% \Delta k/k$)		Bank Position (in. above Core Bottom)	
	Initial Configuration	Reference Design Configuration	Initial Configuration	Reference Design Configuration
0	18.48 $\pm$ 0.7	19.50 $\pm$ 0.5	3.92	3.51
27	13.20 $\pm$ 0.5	11.93 $\pm$ 0.4	7.42	8.17
52	10.42 $\pm$ 0.5	--	8.87	--
84	8.79 $\pm$ 0.4	--	10.11	--
FE	28.83 $\pm$ 0.6	--	--	--

^aCore number represents boron content of lumped poison rods; i.e., Core 27 means 0.27 w/o boron; FE means fuel elements in place of lumped poison rods.

TABLE II

Reactivity Worths of Rod Banks ($\% \Delta k/k$)

Rod Bank	Initial Core 27	Initial Core 52	Initial Core FE	Reference Design Core 27
1	--	1.82	1.82	2.11
2	--	4.83	4.98	--
3	--	8.02	8.31	8.17
4	--	7.42	8.59	7.85
5	12.79	--	13.93	12.14
6	--	--	19.06	--

Studies on both the initial configuration and the reference design configuration consisted of total core reactivities and critical six rod bank positions for the various cores as summarized in Table I, rod bank worths as summarized in Table II, temperature coefficients, and power, flux and gamma distributions.

Studies for the selection of the reference design configuration consisted of determining the effects of lumped poison rods on Y rod worth and the effects of various fuel and lumped poison rod arrays on the shutdown margin for stuck rod conditions and total reactivity.

Initial Criticality and Experiments on N. S. Savannah Core I, R. M. Ball and A. L. MacKinney (B&W).

Core I, designed and manufactured by B&W for the N. S. Savannah passenger cargo ship, was installed and checked as a complete nuclear unit at the Company Critical Experiment Laboratory in February 1960. Figure 1 is a photograph of the core. The fuel consists of UO_2 pellets in stainless steel tubes. The outer 16 cans are filled with 4.6% U-235 and the inner 16 cans are filled with 4.2% U-235.

The active length of the core is 66 in. The fuel pin cladding is 35-mil thick 304 SS with an OD of 0.500 in. There are 164 pins per element, 32 elements in the entire core, and a total loading of 8050 kg of UO_2 , 312 kg of which is U-235. The can wall structure is also 304 SS; the inner zone (16 cans) is 94 mils thick, and the outer zone (16 cans) is 109 mils thick.

Initial criticality at full water height required withdrawal of all "E" control rods and 30% withdrawal of "D" control rods. Figure 2 shows a plan view of the core with proper designations.

Reactivity and hold down of various rod patterns, shown in Figure 3, were determined by the partial water height method.¹ "Excess reactivity" is defined as the amount between critical (zero reactivity) and the pattern shown. "Hold down" is the amount held down by the rod pattern. The control rods are fabricated from boron stainless steel with 1.7% of boron enriched in B-10. The followers are made from Zircaloy-2.

Measurements on the core with 4.6% U-235 in the central 16 cans and 4.2% in the outer 16 cans indicate a reactivity \$2.0 higher than the normal core. The core is subcritical with all rods inserted and just critical, clean cold with "E" rods withdrawn 85%.

The excess reactivity for the normal core is \$14.6. The water temperature effect on reactivity is negative in the range of measurement 16° to 51°C.

Additional work^{2,3} included power and flux distributions, instrument response to loading sequence, approach to criticality and kinetics work with an oscillator, a pulse neutron source, and reactor noise system.

¹A. L. MacKinney and R. M. Ball, TRANSACTIONS OF THE AMERICAN NUCLEAR SOCIETY, 2,1(1959), p. 169.

²R. M. Ball and A. L. MacKinney, "Nuclear Merchant Ship Reactor Zero Power Test—Core I," BAW-1202 (July 1960).

³A. L. MacKinney, et al., "Nuclear Merchant Ship Reactor Extended Zero Power Tests," BAW-1203 (January 1961).

Fig. 1. N. S. Savannah Core I

Fig. 2. Plan View of Savannah Core

<u>Rod Pattern of Fully Inserted Rods (Others are Followers)</u>	<u>Excess Reactivity in Dollars</u>	<u>Hold Down in Dollars</u>	<u>Rod Pattern of Fully Inserted Rods (Others are Followers)</u>	<u>Excess Reactivity in Dollars</u>	<u>Hold Down in Dollars</u>
2.25	12.35		6.30	8.30
1.50	13.10		9.60	5.00
2.50	12.10		12.70	1.90
0.50	14.10		13.50	1.10
			14.60	0

Fig. 3. Reactivity and Hold Down of Rod Patterns

Plutonium Recycle Reactor Critical Tests, J. R. Triplett, L. C. Schmid, R. E. Peterson, R. E. Dunn, and J. J. Regimbal (GE-HAPO). The PRTR¹ achieved criticality on November 21, 1960. Since that date a considerable amount of zero-power testing has been performed. The following results are among those of particular interest in the heavy water reactor field; a complete description of all tests is also available.²

TABLE I
Reactivity Effects of Coolant Removal

Description	Total Effect (mk)	Effect per Element (mk)
Loss of 99.7% D ₂ O:		
Total reactor	-16.4	-0.193
13 central U	+ 0.5	+0.04
12 intermediate Pu	- 5.8	-0.5
24 peripheral U	- 2.7	-0.1
Loss of 90% D ₂ O, 10% H ₂ O:		
Total reactor	- 9.5	-0.112
Loss of H ₂ O:		
19 central U	+23.4	+1.23
6 peripheral Pu:	+ 3.5	+0.58
Estimated total:	+81	+0.95

Normally, 30% to 45% of the 85 fuel elements in the reactor will be 1.8 w/o plutonium in aluminum, the remainder swaged natural uranium oxide. Three loadings were given particular attention in the critical tests. A "two-zone" loading contained 36 plutonium elements in the periphery and 49 uranium elements in the center. A "three-zone" loading contained 30 plutonium elements in a ring

at an intermediate radius. Sample flux traverses, showing effects of various perturbations, are shown in Figure 1. The third loading consisted of 11 plutonium elements only, charged into the center.

The primary control and shutdown system for the reactor is moderator level, which can be varied from 16 to 109 in., although the minimum level for criticality is restricted to 50 in. A secondary, slow-response set of 18 three-piece shim assemblies is also employed. These two systems were calibrated together and also independently by use of boric acid poison in the moderator. The shim poison elements used were gray rather than black to thermal neutrons, being made of Inconel X. The full system has a strength of ~135 mk, corresponding to a maximum of 4 to 7 mk in single assemblies (because of the non-uniform fuel loadings used, the interaction of these shim rods is positive under some conditions). A summary of results on coolant loss is presented in Table I. Since earlier measurements³ had indicated a definitely positive void coefficient for an infinite lattice of this type, these results are particularly significant from the reactor safety standpoint. They also indicate that while high D₂O purity is not required for the coolant system, H₂O would be an unsafe choice. The reflector worth in the cold reactor (100°F) was 40.0 mk.

¹"Reactor Engineering Development Operation, Plutonium Recycle Test Reactor, Final Safeguards Analysis," HW-61236 (1959).

²"Hanford Laboratories Operation, Plutonium Recycle Test Reactor Critical Tests," HW-61900-B (1960).

³J. R. Lilley, "Correlation of Data on Heavy-Water Moderated Cluster Lattices," HW-60275 (1959).

Fig. 1. Radial Flux Distribution:

Curve A is from typical two zone loading. Curve B shows the tilt of the distribution because of the replacement of a Pu-Al fuel element by a UO₂ element and also the distortion caused by two shims which pass completely through the reactor. Curve C is from a typical three zone loading. Curve D is from the same loading as Curve C but without a two foot reflector. Curve E is obtained from gold foils which were covered by 40-mil thick cadmium.

Critical Experiments I

Measured Nuclear Parameters for Natural Uranium Rod Lattices in D₂O, T. J. Hurley and G. F. O'Neil, (DuP-SR).

Full sets of nuclear lattice constants were measured for various systems of natural uranium rods in D₂O over a wide range of rod cluster sizes and spacings. The fuel rods were 1 in. in diam and clad with 0.032 in. of aluminum. Cluster sizes included 1, 3, 7, and 19 rod bundles. Lattice spacings ranged from 7.00 to 21.00 in. Buckling values have been previously reported for these lattices¹.

TABLE I. Experimental Lattice Parameters and the Inferred Bucklings Compared with the Measured Bucklings^a

No. Fuel Rods	Lattice Pitch (in.)	n	ϵ^b	f	p	L ³ (cm ³)	τ (cm ²)	B ₁ ² (m ⁻²)	B ₁ ^{2c} (m ⁻²)
1	7.00	1.312	1.039	0.969	0.956	237	132	6.59	6.48
	8.08	1.318	---	0.960	0.968	337	132	5.51	5.29
	9.33	1.315	---	0.948	0.972	456	131	4.19	4.08
3	12.12	1.318	---	0.914	0.979	783	130	2.41	2.30
	7.00	1.318	1.048	0.984	0.869	95.0	140	7.14	7.59
	9.33	1.306	---	0.976	0.916	181	133	6.65	7.11
7	12.12	1.315	---	0.961	0.948	331	131	5.19	5.05
	14.00	1.315	---	0.948	0.941	459	131	3.74	3.81
	9.33	1.308	1.053	0.985	0.843	86.0	141	6.19	6.02
19	12.12	1.311	---	0.978	0.889	164	136	6.28	6.11
	14.00	1.306	---	0.970	0.902	242	134	5.09	5.20
	18.52	1.317	---	0.945	0.917	496	132	3.05	2.92
7 ^e	21.00	---	---	0.931	0.916	648	132	2.28 ^d	2.10
	18.52	1.313	1.044	0.947	0.916	418	132	3.35	3.28
	14.00	1.311	1.077	0.983	0.788	106	146	3.35	3.41
19	18.52	---	---	0.970	---	236	139	---	3.02

^aAll numbers corrected to D₂O isotopic purity of 99.75 mol %.

^b ϵ values averaged for each fuel assembly.

^cB₁² = $\frac{-(L^2 + \tau) + \sqrt{(L^2 + \tau)^2 - 4(L^2\tau)(1 - \eta\epsilon p)}}{2L^2\tau}$

B₁² = Full lattice buckling measurements (Flux shape fitted to Cos and J₀)

^d η used 1.31 as inferred from other measurements.

^eIndicates 2.0-in. center to center internal rod spacing; all others 1.5-in. center to center spacing.

The lattice parameters were all determined from foil activation measurements made in the central cells of the full critical lattice loadings in the Process Development Pile (PDP). Thermal and epithermal flux distributions were determined from bare and cadmium covered irradiations of Cu, Mn, and U-235 foils placed in both the fuel and moderator.

The subcadmium activation distributions were used in conjunction with cross sections computed by Westcott² to calculate values of the thermal utilization f and the thermal migration area L² in the usual way. A base value of η was calculated from Westcott values, assuming the neutron flux spectrum in the moderator to be Maxwellian at 20°C. This value was then modified for flux hardening effects by comparing the ratios of the 1/v activations (Cu and Mn) and the U-235 activations at various locations. Values of the fast fission factor ϵ were obtained by comparing the fission product activities of natural and depleted uranium foils according to the technique described by Futch³. The neutron age τ was measured to indium resonance from isolated fuel assemblies in D₂O⁴. Corrections were calculated for the age to thermal energy and for lattice effects.

The measurement of the resonance escape probability p involved measurements of the 104 kev neptunium X-ray according to techniques described by Kranz⁵. The ratio of epicadmium to subcadmium U-238 absorptions was corrected for both leakage and epicadmium 1/v absorptions.

The results of these measurements are given in Table I. The over-all accuracy, excluding possible systematic errors, is of the order of $\pm 1\%$ in the microscopic parameters, and τ , while in L², is of the order of $\pm 2\%$. The numbers were also used in a four-factor, diffusion kernel to compute buckling values. The last two columns of the table compare the computed buckling values with the directly measured bucklings previously reported. This comparison indicates that the measured parameters can be used to predict the buckling with an accuracy of ± 3 to 4% in most cases.

¹T. J. Hurley Jr. et al., "Natural Uranium-D₂O Bucklings Over an Extended Range of Pitch and Fuel Assembly Size," *Trans. Am. Nuclear Soc.*, 4, 1 (1961), p. 103.

²C. H. Westcott, "Effective Cross Section Values for Well-Moderated Thermal Reactor Spectra," CRRP-960 (Nov. 1960).

³A. H. Futch Jr., "Fast Fission Effect in Lattices of Natural Uranium and Heavy Water," *Nuclear Sci. and Eng.*, 5, 1 (1959) p. 61.

⁴W. E. Graves, "Measured Slowing Down Distributions From a Line Source," *Trans. Am. Nuclear Soc.*, 3, 2 (1960) p. 486.

⁵A. Z. Kranz, "Measurements of Thermal Utilization, Resonance Escape Probability, and Fast Fission Factor of Water Moderated Slightly Enriched Uranium Lattices," WAPD-134 (Sept. 1955).

Comparison of Exponential and Critical Bucklings for Natural Uranium Rods in Heavy Water, E. C. Wingfield and E. J. Hennelly (DuP-SR).

A previous study indicated the possibility of differences between bucklings measured for heavy water-uranium lattices in critical assemblies and those measured in exponentials. The critical bucklings obtained from ZEEP¹ at Chalk River, were approximately 30 μB ($1 \mu\text{B} = 10^{-6} \text{ cm}^{-2}$) lower than the exponential bucklings obtained with the same lattice at North American Aviation² (NAA). Sufficient data have not been available to ascertain if similar differences can be generally expected between exponential and critical buckling measurements.

As part of an extensive exponential-critical intercomparison, buckling measurements in 42 lattices of 1 in. natural uranium metal rods in heavy water have been made in an exponential facility, the Subcritical Experiment (SE), at two moderator purities. The rods were clad with 0.032 in. of aluminum. The fuel assemblies were single rods or clusters of 3, 7, or 19 rods with a center-to-center spacing of 1.50 in. Lattice pitches were varied from 3.00 to 21.59 in. The material buckling B_m² was computed from B_m² = B_r² - B_z². The radial buckling B_r² was assumed to be constant at a previously measured value of 925 $\pm \mu\text{B}$. The axial buckling B_z² was measured with a gamma-compensated, neutron-sensitive ion chamber which was moved vertically through the lattice. Experimental values of the effect of moderator purity on buckling were used to correct all the measurements to a moderator purity of 99.75 mol % D₂O. The exponential buckling measurements of single rod lattices in the SE and ZEBRA³ and at NAA are in excellent agreement.

Fig. 1. Effect of Moderator to Fuel Ratio on Differences Between Exponential and Critical Bucklings for Natural U Rods in D₂O

Fig. 2. Comparison of NAA and ZEEP Bucklings of Natural U Rods

A systematic comparison was made between these exponential bucklings and highly precise bucklings that were measured in a large critical facility, the Process Development Pile⁴ (PDP). Fuel used in the SE was a portion of what had been used in the PDP. A systematic trend in the differences between the SE and PDP bucklings is shown in Figure 1. Possible causes for the differences between the exponential and critical bucklings include (1) changes in the nuclear parameters (such as resonance escape probability) due to differences in the flux profiles of the two facilities, (2) changes in the neutron diffusion properties for lattices with larger fuel assemblies, and (3) systematic changes in SE radial buckling for different lattices.

Agreement between the NAA-ZEEP measurements can be obtained if the appropriate "adjustments" from the single rod curve of Figure 1 are applied to the data. These data are shown graphically in Figure 2. This agreement is particularly gratifying because these adjustments are obtained from the extrapolated portion of the single rod curve at low moderator-to-fuel ratios.

¹D. W. Hone et al., "Natural-Uranium Heavy-Water Lattices, Experiment and Theory, P/212" *Proc. 2nd UN Conf. on Atomic Energy—Geneva (1958) 12*, 315, United Nations, New York.

²E. R. Cohen, "Exponential Experiments on D₂O-Uranium Lattices, P/605" *Proc. 1st UN Conf. on Atomic Energy—Geneva (1955) 5*, 268, United Nations New York.

³R. Persson et al., "Exponential Experiments on Heavy-Water-Natural Uranium Metal and Oxide Lattices," P/160, *Proc. 2nd UN Conf. on Atomic Energy—Geneva (1958) 12*, 364, United Nations, New York.

⁴T. J. Hurley et al., "Natural Uranium—D O Bucklings over an Extended Range of Pitch and Fuel Assembly Size," *Trans. Am. Nuclear Soc.*, 4, 1 (1961), p. 103.

Critical Experiments with Low Enriched Double Annular Superheater Fuel Elements in Light Water and Their Analysis, P. G. Klann, C. O. Dechand, E. Fein, and W. B. Wright (CE).

Criticality and power distribution measurements were made with clean uniform hexagonal arrays of these elements with the interior coolant annuli both voided and flooded for the purpose of checking analysis methods. Double annular fuel elements are of interest in the design of integral nuclear superheat reactors. The double annular geometry of the elements required extension of

present methods for calculating resonance escape probability and required application of existing method for calculating thermal disadvantage factor to more complex geometries. Lattice pitches of 1.875 and 2.092 inches were investigated, which correspond with the annuli voided, to water to oxide volume ratios of 1.085 and 1.718, respectively. The cores were hexagonal cylinders, moderated and reflected with light water. Criticality was obtained by adjusting water height. Power distributions were measured by activating the fuel elements and counting the fission product activity.

The double annular element consisted of two concentric UO₂ fuel annuli clad with Type 304 stainless steel, forming a central, and an intermediate steam coolant annulus. The elements were fabricated by vibratory compaction to a density of 9.31 g/cm³. The active fuel length was 42 in. and the U-235 enrichment was 2.55% (Table I).

TABLE I. Fuel Element Dimensions, Inches

	UO ₂ Annulus I.D.	Inside Clad O.D.	Outside Clad Thickness	Outside Clad Thickness
Inner Annulus	0.269	0.714	0.016	0.018
Outer Annulus	1.000	1.436	0.034	0.032

Experiments were also performed where the complexities of slab and Y-shaped water channels were introduced.

The analysis consisted of two group PDQ calculations using MUFT fast and Maxwellian thermal cross sections modified for hardening. Resonance escape probability for the U-238 resonances was calculated using the Adler-Hinman-Nordheim Method¹ by means of the QUERY Code.² The effective radius of the isolated double annular rod was taken as the radius of the simple isolated rod which has the same P₀ escape probability over a range of energies. The escape probabilities were computed by means of the cylindrical P₀ CEPTR Code³. The interaction of this isolated simple rod with the lattice was taken into account with the Bell method⁴. In Table II are compared the calculated and experimental eigenvalues. The analysis gave agreement with experimental eigenvalues to within 1%, while the calculated flux shape compared to within 15%, the largest disparity occurring in regions bordering slab and Y water channels.

¹F. T. Adler et al., "The Quantitative Evaluation of Resonance Integrals," P/1988 *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958, 16*, 155, United Nations, New York.

TABLE II. Calculated Eigenvalues for Hexagonal Cores Composed of Double Annular Fuel Elements

Description of Core	Water to UO ₂ Vol. Ratio ^a	Distance Across Hexagonal Core Flats (in.)	Water Height (in.)	Calculated Eigenvalue	Experimental Eigenvalue
169 flooded elements	1.718	27.17	28.11	1.0039	1.0000
169 voided elements	1.718		37.24	1.0008	1.0000
271 flooded elements	1.085	30.85	23.48	0.9989	1.0000
271 voided elements	1.085		35.11	0.9903	1.0000
264 flooded elements (Y channel)	1.085	30.85	26.26	1.0055	1.0000
264 voided elements (Y channel)	1.085		43.76	1.0084	1.0000
267 flooded elements (slab channel)	1.085	30.85	25.56	1.0045	1.0000
267 voided elements (slab channel)	1.085		39.93	0.9973	1.0000

^aIn the voided condition

²D. Drawbaugh and G. Sadowsky, "An IBM-704 Program for Calculating Effective Absorption Cross Sections by the Narrow Resonance or Narrow Resonance Infinite Mass Approximation," Combustion Engineering report CEND-94.

³P. B. Daitch et al., "CEPTR—An IBM-704 Code to Solve the P₁ Approximation to the One-Velocity Transport Equation in Cylindrical Geometry," Combustion Engineering report MPC-20 (Sept. 1959).

⁴G. I. Bell, "A Simple Treatment for Effective Resonance Absorption Cross Sections in Dense Lattices," Letter to Editor, *Nuclear Sci. and Eng.* 2, (1959) p. 138.

Description and Analysis of Some "Over-Moderated" Critical Experiments from 68-550°F, G. H. Miley, R. D. Gillespie, J. W. Pollard, and J. A. Bistline (KAPL).

Recent over-moderated light water-Zircaloy-U-235 critical experiments performed in the Pressurized Test Reactor at KAPL afforded an unusual check of analytical techniques, since in some cases two clean critical (rod-free) temperatures were obtained for a single core.¹

Experimental reactivity results are summarized in Figure 1 as are the core specifications. All of the cores were rectangular parallelepipeds, divided into four equal quadrants by two mutually perpendicular "cross water gaps" (rod channels) and were characterized by extreme heterogeneity, fuel plate spacing being about 1 in.

Calculations used a two-dimensional model based on the KARE² diffusion theory code and WOXX³ three-group cross sections. "Homogenization" (self-shielding) factors were determined from fine structure calculations using the Tranvar⁴ thermal transport code. Several other cross section schemes and transport codes¹ in common usage were also investigated, but no significant differences were noted.

Calculated reactivities for Core 208 are included in Figure 1 and are typical of the results for all cores of the series. In general, close agreement at the hot clean critical temperature was obtained, but cold K_{eff}'s were calculated too high by as much as 1%. The error for low temperatures appeared to correlate with the metal to water ratio, the agreement improving with higher ratios. Absolute magnitudes of positive temperature coefficients

were calculated too small by 0.2-0.3¢/F° in the range 80°-100°F, while absolute values of the hot clean critical coefficients were calculated too large by 0.3-0.8¢/F°.

Previous semi-empirical self-shielding temperature correlations, based mainly on data from cores with higher metal to water ratios and with closer plate spacing, could not be used, and a modified correlation based on the Tranvar fine structure calculations has been developed.¹

In general, calculated plate power fractions agreed exceptionally well with both hot and cold measurements. The largest deviations were noted for plates adjacent to the outer reflector. Good agreement was observed for those fuel plates next to the cross water gaps, and this agreement was independent of the gap thickness. The peaks, calculated without spectrum corrections, were consistently higher (1 to 10%) than measured. Because of the low B of these cores, water gap spectrum corrections are expected to be small, so it is likely that errors due to other assumptions in the calculations are more important and cancel out this effect.

¹G. H. Miley and R. D. Gillespie, "Description and Analysis of Some Over-Moderated Critical Experiments From 68 to 550°F, KAPL-M-GHM-6 (Aug. 1961).

²J. A. Archibald Jr., "KARE Input," KAPL-M-JA-6 (Apr. 1960).

³J. A. Archibald Jr., "Equations for the WOXX Cross-Section Routine of the KARE System," KAPL-M-JA-8 (May 1960).

⁴N. C. Francis et al., "Variational Solutions of the Transport Equation," p/627 *Proc. 2nd UN Conf. on Atomic Energy Geneva (1958)* 16, 517 United Nations, New York.

Description and Analysis of Some Semi-Distributed Boron Cores, G. H. Miley, B. Miller, and J. H. Holmes (KAPL).

Recent critical experiments in the Pressurized Test Reactor at KAPL are uniquely characterized by "semi-distributed" poison (boron-steel plates adjacent to every second or third fuel plate).¹ The very complex power distribution perturbations thus produced have been analyzed with good accuracy by means of two different techniques that are modifications of conventional analytic methods. Reactivity and temperature coefficient results based on these techniques also show fair agreement with experiment, but some discrepancies are noted.

Fig. 1. Excess Reactivity vs Temperature

TABLE I. Reactivity Comparisons

Core	Temp (°F)	K _{eff}		Temp Coeff (¢/F°)	
		Cal	Exp	Cal	Exp
255	522	1.002	1.000		
	500 - 522			-4.3	-3.8
225B	417	1.005 ^a	1.000		
	400 - 417	1.004		-1.54	-1.34

^aBased on the thin region model. Other results for the unit cell model.

Assemblies were rectangular and measured 24 x 24 x 30 in. All were light water moderated and reflected, with enriched uranium-Zircaloy construction and a metal to water ratio of 0.5. Fuel plates were uniformly distributed with $\sim 1/3$ in. spacing and with a total loading of 15.48 kg U-235 and 40.6 g B-10. Core 225 (Table I) was close-packed, but Core 225B was divided into four equal quadrants by the introduction of 1 in., mutually perpendicular water gaps.

Much of the analysis was based on a two-dimensional diffusion theory model using the KARE² code and three-group cross sections. This model was "homogenized within unit cells"; i. e., each fuel plate and fuel-boron plate combination, along with the associated Zircaloy-water structure, was homogenized and treated as a separate region, the final model being a composite of 180 such regions.

For comparison, a second two-dimensional model was constructed in which all fuel and boron plates were represented by thin region theory⁴ superimposed on a homogenized Zircaloy-water structure region. Thin region constants for each plate were averaged over the spectrum determined for the corresponding unit cell used in the first model. Cross sections for the explicit Zircaloy-water regions between plates were averaged over the spectrum used for the associated plates.

Detailed plate-by-plate power fractions calculated with these two models were compared with hot clean critical activation measurements analyzed by gamma scanning the fuel plates. The average calculated to experimental power fraction ratio for all plates associated with boron was 0.99+ for Model 1, which is considered excellent agreement in light of the random distribution and importance of the boron. Model 2 was somewhat less accurate, giving an average ratio of 0.97. Some discrepancies were noted for both models for plates immediately adjacent to the reflector and water gap. The peak power for core 225B, which occurred at the cross gap intersection, was calculated in both models somewhat higher (>10%) than measured.

Some reactivity comparisons are noted in Table I. k_{eff} calculated for the hot clean critical condition using either model agreed closely with experiment. Errors up to 0.5% were noted in the temperature coefficient calculation. Errors of this order of magnitude have been noted in other studies.

¹G. H. Miley and B. Miller, "Description and Analysis of Some Relatively Large, Low $\beta\gamma$ Critical Experiments With Semi-Distributed Boron Strips From 68-550°F, KAPL-M-GHM-7 (Aug. 1961).

²J. A. Archibald Jr., "KARE Input," KAPL-M-JA-6 (Apr. 1960).

³J. A. Archibald Jr., "Equations for the WOXX Cross-Section Routine of the KARE System," KAPL-M-JA-8 (May 1960).

⁴E. L. Wachspress, "Thin-Region in Diffusion Theory Calculations," *Nuclear Sci. and Eng.*, 3, (1958), pp. 186-200.

⁵G. D. Hickman et al., "Water-Moderated Cores With Boron Steel Septa at Elevated Temperatures," *Nuclear Sci. and Eng.*, 8, (1960), pp. 381-392.

Analysis of Highly Enriched Water Moderated Critical Assemblies—Part 1; The Computational Model and Analysis of Slab Experiments at Room Temperature, R. J. Breen, G. F. Bogar, D. R. Connors, H. J. Litke, D. S. Rampolla, and J. A. Smith (W-BAPD).

The calculational model described here has been developed as a compromise between accuracy and simplicity for calculating the criticality, power distributions, and temperature coefficients in highly enriched, water moderated critical assemblies with zirconium structure. The applications of this model are illustrated in the analysis of various experiments described in this and the following paper.

Four-group diffusion theory is used; fast cross sections (Groups 1, 2, 3) are obtained with the MUFT code¹ using the P-1 approximation. The MUFT library used includes zirconium with a resonance integral of 1.1 barns and inelastic scattering in Group 1. Thermal cross-sections (Group 4) are based on the Mixed Number Density Model² (averaging the diffusion coefficient over a Maxwellian spectrum and the absorption and fission cross sections over a SOFOCAT³ spectrum) with continuity of neutron density enforced at boundaries.

Thin absorbing membranes are treated in Group 3 with P-1 blackness theory where transmission of the first and second moments of the flux is calculated with transport theory and fictitious constants are found which yield the appropriate absorption rate in diffusion theory. In Group 4, fictitious constants are found which in diffusion theory match the absorption rate calculated with a double P-1 transport approximation using 25 thermal neutron energy groups. Two-dimensional calculations in a horizontal plane are synthesized axially to three dimensions.

The above methods have been applied to the analysis of several water reflected critical slab assemblies (Table I) which varied widely in size, metal to water volume ratio, and fuel density. Each assembly consisted of a uniform array of 1.1-in. square zirconium bundle boxes containing 0.9-in. wide fuel bearing Zircaloy elements separated by water channels. Fuel bearing extensions were pulled into a slab as the control rods were withdrawn. The slab size was such that the rod-free value of k_{eff} (determined from extrapolated rod worths) exceeded unity by less than $\frac{1}{2}\%$. The fueled height of each assembly was 36 in. Activation shapes were measured with U-235-Zircaloy wires located in Zircaloy wands inserted horizontally into the assembly through slots in the bundle box walls.

In the two-dimensional description of a horizontal slice through the assembly, each bundle box was represented as a homogeneous, fuel bearing rectangle surrounded by a homogeneous metal and water region. This representation was necessary for adequate eigenvalue and activation shape calculations. The calculated eigenvalues listed in Table I lie with +0.5 and -0.9% of the experimental values. The calculated peak to average activation

TABLE I. Description of Experiments and Results of Calculations

Experiment	No. of Boxes	Metal/Water Volume Ratio	g/U-235	k_{calc}^2
1	7 x 20	1.17	58.1	1.003
11	7 x 23	1.75	77.5	1.001
15	6 x 19	1.32	96.8	0.998
24	6 x 26	0.95	52.5	1.005
26	5 x 22	1.13	106	1.005
30	6 x 19	1.40	116	1.001
32	5 x 27	1.38	155	0.991
75	5 x 23	1.41	232	1.004

² k_{calc} is normalized to an experimental value of unity.

rates were from 0 to 10% above the measured values which have uncertainties of $\pm 5\%$. No correlation has been observed between the errors in the calculated eigenvalues and the parameters listed in Table I.

¹A. F. Henry, "54 Group Library for P-1 Programs," WAPD-TM-224 (Apr. 1960).

²R. J. Breen, "A One Group Model for Activation Calculations," *Nuclear Sci. and Eng.*, 9 (1961), pp. 91-93.

³H. J. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged Over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCAT Code," WAPD-TM-72 (1957).

Analysis of Highly Enriched Water Moderated Critical Assemblies—Part II; Analysis of Slab Experiments with Absorbing Septa and Varying Temperatures, H. J. Litke, G. F. Bogar, R. J. Breen, D. R. Connors, D. S. Rampolla, and J. A. Smith (W-BAPD).

The methods outlined in the preceding paper have been applied to the analysis of assemblies containing self-shielded absorbing septa. Eigenvalues were calculated for several slab experiments run at room temperatures which were transected parallel to the long dimension by boron or hafnium septa. The boron septa consisted of layers of boron-impregnated mylar tape in an aluminum sandwich. The hafnium septa were sheets of natural hafnium metal. Control was achieved as described in the preceding paper by means of rods with fuel bearing extensions. The rod free value of k_{eff} was again in excess

of unity by less than $\frac{1}{2}\%$. The details of the experiments and the calculated eigenvalues are listed in Table I. The good agreement between the calculations and the experiments in these cases and in the cases described in the preceding paper indicate that the absorption rates in the septa are calculated by the model to within $\pm 5\%$.

Activation shapes, critical rod positions, and temperature coefficients were calculated for a slab experiment which was operated at various temperatures in the Bettis High Temperature Test Facility. Control was achieved with four Ag-In-Cd blade rods inserted axially in 1.5-in. water gaps. The slab contained boron septa which were 0.090-in. boron-stainless steel plates with a 2200 m/sec optical thickness of 1.0.

Table II gives a comparison of the measured and calculated temperature coefficients and eigenvalues between 74° and 464° F. The temperature defect between 74° and 464° F was estimated from integrated rod worths to be about 3%. Rod worths and temperature coefficients were based on a calculated effective delayed neutron fraction $\bar{\beta}$ of 0.0074 (i.e., $\bar{\beta}/\beta = 1.16$). In addition to having a small absolute bias (all calculated eigenvalues are within + 0.5 and - 0.9% of measured values), the model describes the temperature effects quite accurately.

Activation shapes were measured at 464° F perpendicular to the boron septa and perpendicular to the water channels. The agreement between measured and calculated peaks was within $\pm 5\%$, which is comparable with that for the room temperature experiments.

TABLE I. Description of Room Temperature Experiments and Results of Calculations

Experiment	Number of Boxes	Metal/Water Volume Ratio	g/liter U-235	Septum	k_{calc}	Septum Worth Δk
12	9 x 18	1.17	58.1	1.0 B ^a	1.005	0.10
13	10 x 23	1.17	58.1	2.0 B	1.003	0.14
27	7 x 18	1.13	106	1.0 B	1.008	0.10
84	7 x 18	1.41	232	2.0 B	1.002	0.12
33	9 x 18	1.38	155	2.0 Hf	0.991	0.19
57	10 x 21	1.38	155	1.0 Hf-W ^b -1.0 Hf	0.990	0.24
60	10 x 25	1.38	155	0.5 Hf-W ^c -0.5 Hf	0.985	0.26

^a1.0 B → Boron septum, optical thickness = 1.0 at 2200 m/sec

^bW → 0.5-in. water gap

^cW → 1.4-in. water gap

k_{calc} is normalized to an experimental value of unity

TABLE II. Comparison of Analytical and Experimental Results of Temperature Coefficients and k_{eff} for a Slab Experiment

Temp (°F)	Four Rod Bank Critical Position (in.)	Calculated k_{eff} for Experiment at Critical Rod Height	Temperature Coefficient ($\times 10^{-4}/^{\circ}\text{F}$)	
			Experimental ^a	Calculated
74°	14.230	1.0020		
12°	14.314	1.0020	-0.39	-0.40
337°	16.796	1.0026	-1.06	-1.30
464°	22.655	1.0020	-2.61	-2.25

^aCorrected for a calculated $\bar{\beta} = 0.0074$.

Critical Experiments II

Experiments and Calculations on Unmoderated Uranium-Molybdenum Alloy Critical Assemblies, J. T. Mihalcz and W. E. Kinney (ORNL).

Critical dimensions, fission rate distributions, and void coefficients of reactivity of three unmoderated, unreflected assemblies of a 10 w/o molybdenum-90 w/o uranium (93.2 w/o U-235) alloy have been measured to support the design of an unmoderated and unreflected research reactor capable of producing 50 μ sec bursts of $\sim 10^{17}$ fissions.¹ The critical dimensions of nine assemblies with plexiglas (methacrylate plastic; $\rho = 1.2$ g/cc) and 0.025-cm thick cadmium sheet as reflectors were determined. Multigroup transport theory calculations were made to predict critical dimensions of the unreflected assemblies; void coefficients of reactivity were computed by one-group and multigroup transport theory and by perturbation theory.

The unreflected critical assemblies studied were:

- 1) A solid, 20.32-cm diam U-Mo alloy cylinder;
- 2) A 20.32-cm diam U-Mo alloy cylinder with a 5.08-cm diam axial hole; and
- 3) A 20.32-cm diam U-Mo alloy cylinder with a 5.08-cm diam axial stainless steel core.

The reflected assemblies resembled (2) but with the lower 12.70 cm of the hole enlarged to 8.89 cm and with various conditions of reflection.

Calculations were performed with a multigroup transport code² using the S_n method of solution with an S_4 approximation and the group structure and cross sections as listed by Mills.³ Measured and calculated critical heights of the unreflected assemblies are compared in Table 1. Measured and calculated fission distributions and void coefficients of reactivity are given in Figure 1.

TABLE 1. Critical Parameters of Various 20.32-cm diam Uranium-Molybdenum Alloy Cylinders (Density, 17.08 g/cc)

	Critical Height (cm)		Critical Mass (kg of U-235)
	Experimental	Calculated	Experimental
Unreflected			
Solid cylinder	14.78	13.86	68.6
Annular cylinder:			
5.08-cm diam axial hole	19.74	18.22	85.9
5.08-cm diam steel core	18.92	17.57	82.4
Reflected			
Annular cylinder; plexiglas reflector ^a	12.57	-	47.2
Annular cylinder; plexiglas plus cadmium reflector ^b	15.80	-	61.2

^a8.89-cm diam axial void; 5.08-cm thick plexiglas on all outer surfaces.

^bTwo-diam void as described in text; 5.08-cm thick plexiglas on all outer surfaces; 0.025-cm thick cadmium between cylinder and reflector.

The calculations predict heights which are 6 to 8% less than those experimentally determined. From the measured reactivity change resulting from a variation in the height of each assembly, the calculated multiplication constants of the critical configurations are about 2% too large. Although the cross sections which were used give excellent results for fast, unalloyed uranium systems, the neutron spectrum in U-Mo is softer than that in pure uranium,

Fig. 1. Fission Distribution and Void Coefficients in a 20.32-cm Diam Cylinder of U-Mo with a 5.08-cm Diam Steel Core

and group constants, especially w_{0f} , characteristic of a somewhat lower energy should ideally be used. The errors in cross sections and the uncertainties in obtaining group constants could easily account for the differences between the calculated and experimental values of the critical parameters.

¹G. Breidenbach et al., "Preliminary Design of the ORNL Fast Burst Reactor," NDA-2136-1 (July 30, 1960).

²B. Carlson et al., "The DSN and TDC Neutron Transport Codes," LAMS-2346 (Feb. 12, 1960).

³C. B. Mills, "Neutron Cross Sections for Fast and Intermediate Nuclear Reactors," LAMS-2255 (Jan. 16, 1959).

Analysis of a Beryllium Moderated Critical Experiment,* G. Srikantiah (BNL).

Using the recently available data on beryllium cross-sections, a study was made of the beryllium moderated critical experiment: CA-18 of the Oak Ridge National Laboratory.¹ The nuclear data available at the time these

experiments were carried out were inadequate for accurate calculation of criticality. In view of the recent improvement in cross-section data for both beryllium and U-235, the estimated value of effective multiplication factor is 1.015 which exceeds the measured value of 1.002 by about 1%. Taken into account in the estimation of this value were the effect of anisotropic scattering, the fast effect in beryllium, the self-shielding effect, and the effect of reflection due to the assembly structure. The anisotropic group scattering cross-sections were calculated from the angular distributions compiled by Parker² by numerically integrating over the various groups. The $(n, 2n)$, (n, α) cross sections are also tabulated by Parker. These, in conjunction with the recent data of Levin and Cranberg as quoted by Sher and Moore,³ were utilized in the present calculations. The distribution of secondary neutrons from the $(n, 2n)$ reaction in beryllium is not well known; the distributions used here were those recommended by Parker which follow closely the recommendation of Goldstein.⁴ By suitably arranging these data in the multi-group analysis, the calculated value of the beryllium fast effect for the CA-18 assembly was $\epsilon = 1.07$. The same cross section data give a value of $\tau = 1.075$ in an infinite beryllium medium with a fission source. This value may be compared with those obtained by Moore & Chernick, Haffle, Goldstein, and Rief which average to a value $\epsilon = 1.075 \pm 0.020$.

The difference in the effective multiplication constant due to the assumption of isotropic and nonisotropic scattering has been estimated to be 4.1%.

The value of age calculated on the basis of these cross sections is $\tau(1.4 \text{ ev}) = 86 \text{ cm}^2$. If isotropic cross sections are assumed, the value is $\tau(1.4 \text{ ev}) = 73 \text{ cm}^2$. Including the effect of nonelastic scattering by means of a two-group treatment yields $\tau(1.4 \text{ ev}) = 79 \text{ cm}^2$. There is only one measurement of age in beryllium, made in 1947, which is quoted in literature:⁵ $\tau(1.4 \text{ ev}) = 80.2 \pm 2\%$.

The self-shielding factor for this assembly has been calculated to be 2.9%, of which only 1% is contributed by resonance neutron self-shielding. This was estimated from the results of a thermal spectrum calculation which gave the self-shielding contribution due to thermal neutrons.

The values of η for U-235 in the intermediate energy range are not definitely known, and the multigroup calculation on CA-18 indicates that the change in average η for U-235 from 1.7 to 1.8 results in only a small change in k . The sensitivity to intermediate energy η values is much more pronounced in less thermal assemblies such as CA-1.¹

¹ E. L. Zimmerman, "Two Beryllium Moderated Critical Assemblies," ORNL-2201 (1958).

² K. Parker, "Neutron Cross-Sections of Beryllium In The Energy Range 0.025ev - 15 Mev," AWRE 0-27/60 (1960).

³ R. Sher and S. Moore, "Neutron Cross-Section Evaluation Group Newsletter No. 1, Beryllium," BNL-607-T178 (1960).

⁴ H. Goldstein and H. Mechanic, "Penetration of Neutrons From a Point Fission Source Through Beryllium and Beryllium Oxide," NDA 2092-9 (1958).

⁵ D. J. Hughes, *Pile Neutron Research*, Addison Wesley, Cambridge, Mass., (1953), pp. 126-127.

Pressure Tube Reactor Critical Experiment,* R. R. Powell and P. M. Williams (AMF).

As part of a development program for a very high power pool reactor, known as the Pressure Tube Reactor,¹ a critical experiment was performed by installing a temporary core structure in the pool reactor facility of Industrial Reactor Laboratories, Inc., Plainsboro, New Jersey, of which AMF is a part-owner. The PTR is light water moderated and fueled with an aluminum clad uranium-aluminum alloy. The volume composition of the core is similar to that of a typical pool reactor as shown in Table I.

The main components of the core assembly consist of fuel, control rods and guide assemblies and a false grid. Figure 1 shows the false grid, partially loaded with dummy fuel elements, with the control rods and guide assemblies in place. The fuel elements have an active length of 2 ft, consist of six concentric segmented cylinders housed in a 3.51-in. O.D. by 36-in. long aluminum cylinder, and are spaced on a $3\frac{1}{4}$ -in. triangular pitch. The control rods have Y-shaped absorber sections, comprised of a cadmium-stainless steel sandwich, $\frac{3}{16}$ -in. thick, which are guided between fuel elements by assemblies which also act as sway bracing for the control rod magnet guide tubes. The false grid receives and secures the fuel in the desired lattice arrangement. Four legs on the false grid occupy fuel positions in the IRL grid, and can be remotely locked by means of a cam device in each leg.

The IRL core was dismantled and the PTR experiment initiated on July 5, 1961. The final shutdown and disassembly of the PTR core occurred on July 13. During this period, a total of 29 core loadings and 96 criticals were performed. Figure 2 shows two of the core loadings performed, the initial core loading and a flux trap loading. Some of the more important data from these loadings are

*Sponsor: J. MacPhee

TABLE I. Composition of PTR

Material	PTR Core Volume Fraction	IRL Core Volume Fraction
H ₂ O	0.639	0.623
Al	0.358	0.375
U-235	0.00208	0.00239
U-238	0.00016	0.00018

Fig. 1. PTR False Grid, Control Rods and Dummy Fuel Elements

Fig. 2. PTR Critical Loadings

summarized in Table II. In general, experimental data agreed very well with the data calculated for the hazard report.² With respect to the cold-clean critical mass the experimentally determined value of 3.25 kg U-235 agreed surprisingly well with the predicted value of 3.2 kg U-235.

TABLE II. Measured Results for PTR Critical

	Initial Critical	Flux Trap Core
U-235 loading	3.337 kg	4.354 kg
Excess reactivity	+0.67% $\Delta K/K$	+1.60% $\Delta K/K$
Worth of control rods	-9.3% $\Delta K/K$	-7.0% $\Delta K/K$
U-235 loading per elements		
Standard 6 fuel cylinders	202 g U-235	202 g U-235
Partial 6 fuel cylinders	108 g U-235	108 g U-235
Maximum thermal flux at center of core at 1 watt power level (unperturbed)	1.75×10^7 n/cm ² /sec	4.7×10^7 n/cm ² /sec
Cd ratio at center of core	2.1	7.0
Reactivity effects of materials placed in central core position		
$\frac{1}{2}$ " x 24" SS rod	-0.5% $\Delta K/K$	-0.2% $\Delta K/K$
$\frac{1}{2}$ " x 24" borated Pyrex rod	-1.4% $\Delta K/K$	-0.5% $\Delta K/K$
$\frac{1}{2}$ " x 24" void	-0.07% $\Delta K/K$	+0.05% $\Delta K/K$
$\frac{1}{2}$ " x 24" Al rod	-0.08% $\Delta K/K$	+0.06% $\Delta K/K$

¹ J. MacPhee et al., "The 25 Mw Prototype PTR Design" Paper 32-1, *these Transactions*.

² AMF Atomics PTR Critical Experiment-Hazards Summary Report, NP-9603 (Nov. 10, 1960) 71 pp.

Critical Experiments on 4%-Enriched UO_2 Lattices Moderated by D_2O - H_2O Mixtures, N. L. Snidow, R. H. Clark, C. E. Barksdale, R. H. Lewis, and T. C. Engelder (B & W, Lynchburg).

A series of thermal and epithermal critical experiments was performed to demonstrate the reactivity hold-down of D_2O in conventional light water lattices and study the nuclear properties of lattices moderated by D_2O - H_2O mixtures. The fuel rods consisted of 4%-enriched UO_2 swaged in stainless steel, 0.444 in. ID, 0.475 in. OD and 66.7 in. active length. Each rod contained 56.61 gms U-235. The rod to moderator volume ratio in the lattice was 1.0 in all cases. The experiments were performed in a 60-in. diam aluminum tank so that a finite radial reflector existed.

As the concentration of D_2O in the moderator was changed, reactivity was balanced in two ways: by changing the core diameter or by poisoning the moderator with boric acid. The critical parameters for six cores, covering a range of D_2O concentrations from 0 to 76.7%, are listed in Table I. Flux traverses were made with bare and cadmium covered gold foils to demonstrate the existence of an asymptotic region and to yield bucklings from which radial and axial reflector savings were derived. The reflector savings as a function of reflector thickness in the transition between Core III and IV was also measured over a limited range of thickness.

The thermal disadvantage factor was measured in each lattice using bare and cadmium dysprosium-aluminum alloy sector foils. Cadmium ratios of U-235 were also obtained by standard techniques to determine the degree of thermality. The fraction of thermal fissions decreased from 79 to 53% as the D_2O concentration increased from 0 to 76.7%. In selected lattices the asymptotic neutron spectrum up to 2000 ev was measured by activating calibrated resonance foil detectors.

The cadmium ratio of U-238 was measured in two cores with the highest D_2O concentrations to separate thermal and epithermal absorptions. Two different tech-

TABLE I. Critical Parameters

Core No.	D ₂ O Conc (Mole%)	Boron Conc (g B/l)	No. of Fuel Rods	Critical Radius (cm)	Critical Height (cm)	Critical Buckling ($\times 10^{-4} \text{ cm}^{-2}$)	Thermal Disadvantage Factor	Cadmium Ratio	
								U-235	U-238
I	0	0	484	18.76	153.7	87.9 $\pm$ 0.3	1.26 $\pm$ 0.02	4.78 $\pm$ 0.10	-
II	0	3.39	4904	59.71	145.5	17.2 $\pm$ 0.2	1.26 $\pm$ 0.02	-	-
III	76.7	0	5284	61.98	153.2	10.8 $\pm$ 0.1	1.18 $\pm$ 0.02	2.14 $\pm$ 0.02	1.08 $\pm$ 0.02
IV	70.0	0	2252	40.46	151.0	18.6 $\pm$ 0.3	1.16 $\pm$ 0.01	2.37 $\pm$ 0.01	1.10 $\pm$ 0.02
V	70.0	0.422	5284	61.98	145.0	14.2 $\pm$ 0.1	1.14 $\pm$ 0.02	2.29 $\pm$ 0.01	-
VI	50.0	1.79	5284	51.98	147.6	15.8 $\pm$ 0.1	1.19 $\pm$ 0.01	2.82 $\pm$ 0.04	-

niques were used, both employing direct measurements on the fuel itself to avoid the problems associated with the use of U-238 foils. Fission product interference was eliminated in one technique by rapid chemical separation and β counting the U-239 activity and in the other by γ - γ coincidence measurements of the Np-239 decay. The results are listed in Table I.

Since the desired experimental parameter is ρ_{28} , which is defined as the reciprocal of $(C_{28}-1)$, and since C_{28} is close to unity, relatively large errors in ρ_{28} could not be avoided. Therefore, direct measurements of ρ_{28} were attempted using a third technique by which epicadmium absorptions in U-238 were compared with thermal absorptions in another material possessing a high cadmium ratio in these lattices. By this technique $C_{28}-1$ was measured to be 0.059 ± 0.002 in Core III and 0.090 ± 0.004 in Core IV. These results suggest that the technique is much more sensitive than the standard cadmium ratio measurements, but the possibility of systematic errors has not been eliminated completely.

Nuclear Analysis of 4%-Enriched UO_2 Lattices Moderated by D_2O - H_2O Mixtures, W. A. Wittkopf and K. E. Roach (B & W, Lynchburg).

Nuclear parameters of 4%-enriched UO_2 lattices moderated by D_2O - H_2O mixtures were calculated and these results were compared with experimental measurements. The critical assemblies used were clean, uniform lattices of approximate cylindrical geometry. Physical properties common to all assemblies are given in Table I. The assemblies differ in D_2O concentration, boron concentration, core radius, and reflector thickness.

For the analysis of these assemblies, the BPG code¹ was used extensively along with a four-group, one-dimensional neutron diffusion code. The thermal flux depression factors were obtained from a one-velocity Monte Carlo program.² A resonance integral program³ was used to calculate epithermal absorption parameters for U-238 and U-235.

The 40-group BPG code solves the transport equation for neutron flux, current, and slowing down density as a

TABLE I. Physical Properties of 4%-Enriched Assemblies

Fuel rod to moderator volume ratio	1.00
Square lattice pitch (in.)	0.595 $\pm$ 0.002
Fuel rod diameter (in.)	0.4755 $\pm$ 0.0015
Fuel diameter (in.)	0.444 $\pm$ 0.003
Cladding material	stainless steel (304)
Cladding thickness (in.)	0.0158 $\pm$ 0.0007
Average active fuel length (in.)	66.7
Weight of UO_2 per rod (g)	1600 $\pm$ 2
Weight of U per rod (g)	1408 $\pm$ 2
Weight of U-235 per rod (g)	56.61 $\pm$ 0.10
Lattice volume fractions:	
Moderator	0.4983
Fuel	0.4373
Cladding	0.0644

function of lethargy in a homogeneous medium characterized by a buckling, B . From these quantities, few-group constants are generated for use in a spatially-dependent, few-group diffusion code. The BPG code has the options of using either the B_1 or P_1 approximation, along with either the consistent or inconsistent method, and each of these with either the Greuling-Goertzel or Fermi age model of slowing down for each element of mass less than or equal to 27. For elements of mass greater than 27 the Fermi age model was used.

Fig. 1. Comparison of Calculations with Experimental Measurements

TABLE II. Comparison of Measured and Calculated Nuclear Parameters

D ₂ O conc (mole %)	0	0	76.6	70.0	70.0	50.0
Boron conc (g B/l)	0	3.39±0.07	0	0	0.422-0.008	1.79±0.04
No. of fuel rods	484	4904	5284	2252	5284	5284
Core radius (cm)	18.76	59.71	61.98	40.46	61.98	61.98
Calculated parameters:						
k_e	0.995	1.014	0.991	0.990	0.991	1.002
C_{25}	4.70	3.87	2.12	2.36	2.29	2.83
C_{28}	1.207	1.164	1.066	1.081	1.079	1.108
ϕ_m/ϕ_F	1.27	1.27	1.34	1.32	1.32	1.27
Measured parameters:						
k_e	1.00	1.00	1.00	1.00	1.00	1.00
C_{25}	4.78±0.10	-	2.14±0.02	2.37±0.01	2.29±0.01	2.82±0.04
C_{28}	-	-	1.059±0.002	1.090±0.004	-	-
ϕ_m/ϕ_F	1.26±0.02	1.26±0.02	1.18±0.02	1.16±0.01	1.14±0.02	1.19±0.03
B^2 (x 10 ⁻⁴ cm ⁻²)	87.91±0.28	17.2±0.2	10.77±0.06	18.58±0.31	14.23±0.1	15.79±0.09
B_L^2 (x 10 ⁻⁴ cm ⁻²)	3.59±0.01	3.92±0.01	3.24±0.04	3.46±0.02	3.55±0.02	3.53±0.02

The resonance integral program calculates the infinitely dilute, homogeneous, and lumped resonance integrals from the parameters of individual resonance lines. The program is based on an extension of the methods of Roe,⁴ and was used to generate energy distribution functions for resonance absorption in U-238. The single rod resonance integral measurements of Hellstrand⁵ were used, and Dancoff factors were calculated by Bell's method.⁶

Using the BPG code with the Greuling-Goertzel slowing down model, the fission neutron age in D₂O-H₂O mixtures was calculated. The calculations are compared with experimental measurements^{7,8} in Figure 1, which indicates the excellent agreement.

Using the methods described, the effective multiplication factor (k_e), U-235 fission cadmium ratios (C_{25}), U-238 cadmium ratios (C_{28}), and thermal disadvantage factors, the ϕ_m/ϕ_F for Assemblies I through VI were calculated and compared with experiment.⁹ Table II presents this comparison.

¹ L. D. Gates et al., "The BPG Code," BAW-115 (to be published).

² R. N. Kubik, "Monte Carlo Calculation of Neutron Absorption in a Square Lattice," BAW-154 (1961).

³ K. E. Roach, "Calculations of Infinite Dilute, Homogeneous and Lumped Resonance Integrals," BAW-TM-72 (1961).

⁴ G. M. Roe, "The Absorption of Neutrons in Doppler Broadened Resonances," KAPL-1241 (1954).

⁵ E. Hellstrand, "Measurements of the Effective Resonance Integral in Uranium Metal and Oxide in Different Geometries," *J. Appl. Phys.*, 28 (1957).

⁶ G. I. Bell, "A Simple Treatment for Effective Resonance Absorption Cross Sections in Dense Lattices," *Nuclear Sci. and Eng.*, 5 (1959), p. 75.

⁷ J. W. Wade, "Neutron Age in Mixtures of D₂O and H₂O," *Nuclear Sci. and Eng.*, 4 (1958), p. 12.

⁸ W. G. Pettus, "Neutron Age Measurements in Water and in Metal-Water Lattices," BAW-146 (1960).

⁹ N. L. Snidow et al., "Critical Experiments on 4%-Enriched UO₂ Lattices Moderated by D₂O-H₂O Mixtures," paper 28-4, *these Transactions*.

Initial Operation and Experimentation with the Consolidated Edison Thorium Reactor, Russell M. Ball (B & W, Lynchburg).

Prior to the final installation of the reactor core in the Consolidated Edison Thorium Reactor at Indian Point, New York, the completed nuclear portion was checked at the Babcock & Wilcox Critical Experiment Laboratory. Initial criticality was reached on June 29, 1961.

The plan view of the core is shown in Figure 1. There are three zones of fuel with contents as indicated in Table I.

The fuel, a mixture of thorium oxide and highly (93%) enriched uranium oxide, is in pellet form with a density of 9.25 g/cc. The core contains a total of 17,207 kg of ThO₂ mixed with the UO₂ in six different ratios.

Fuel rods are made of Type 304 stainless steel (200-255 ppm natural boron) with an OD of 0.772 cm and a clad thickness of 0.052 cm. They are brazed together to form square lattices of 195 pins each. The fuel bundles are placed in 14.5-cm square Zircaloy-2 cans. The active core contains 120 fuel elements with an active height of 254 cm. There are 21 movable hafnium control rods, fixed boron steel shims, and flux depressor plates or followers located between the fuel cans.

Excess reactivity was evaluated by the partial water height method¹ as $16.5 \pm \$1$. This is the condition with

TABLE I

Zone	No. of Elements	U-235/element (kg)	Total U-235 (kg)
I	32	6.75	216
II	44	8.79	387
III	44	11.30	497
Total mass U-235 in core			1100 kg

all fixed shims and movable control rods replaced by followers, at room temperature, and with no soluble poison in the water.

The subcriticality, with four fixed shims and all movable hafnium rods inserted, room temperature, and no soluble poison in the water, was $-2.56 \pm \$0.13$. This was determined using the pulsed neutron source technique.² The $1/\beta$ was 2.46 ± 0.04 msec.

Fig. 1. CETR Core Hardware Position Designations

Flux and power distributions near room temperature were made with Mn-Cu wire and a special bundle of individually removable pins. These measurements show good agreement with machine calculations.

¹ A. L. MacKinney and R. M. Ball, *Trans. Am. Nuclear Soc.*, 2, 1(1959), p. 169.

² B. E. Simmons and J. S. King, "A Pulsed Neutron Technique for Reactivity Determination," *Nuclear Sci. and Eng.*, 3, 5(1958), pp. 595-608.

Some Recent Zero-Power Physics Measurements on the Organic Moderated Reactor Experiment, W. B. Wolfe and R. J. Mack (AI, Idaho).

The third core loading of the Organic Moderated Reactor Experiment (OMRE) has been recently completed. A complete description of the OMRE may be found elsewhere.¹ The third core loading is an elliptical array of 31 fuel elements, four of which are developmental fuel element experiments. The remaining 27 elements are the standard 16-plate, stainless steel clad, cermet driver elements used in the previous core loadings. A fairly comprehensive group of zero- and low-power physics measurements have been performed in conjunction with the loading of this present core.

The core reached criticality with 20 driver elements; it was subsequently loaded to 25 driver elements, then to 31, and finally to the configuration including the developmental elements. At each configuration, rod calibrations were performed and excess k determined. Some necessarily limited flux distributions were measured using gold wire attached axially to six or more representative fuel elements. The excess reactivities at 300°F measured with each core configuration are given in Table I.

Temperature coefficients were measured over the range of 250°-400°F at each loading step to determine the effect of the neutron leakage component on the total reactivity coefficient. The temperature coefficient, a strong function of the leakage term, was predicted to become less negative with core size; instead, the experimental results showed a decrease. A re-examination of the buckling of the core offers a reasonable explanation. Although the radial size of the core increases as fuel elements are added, the effective axial length of the core is decreased due to the partial insertion of all of the control rods caused by the increase in reactivity of the larger loading. This decrease in core length is illustrated by the axial flux profiles shown in Figure 1. The buckling of this truncated core is actually increased. The change in the temperature coefficient as a function of the number of fuel elements as calculated and observed experimentally is shown in Figure 2. For purposes of clarity, only the results at 300°F are shown since the same trend prevailed over the entire temperature range measured.

Fig. 1. Axial Flux vs Control Rod Position

TABLE I. Excess Reactivity vs Number of Fuel Elements at 300°F

No. of Fuel Elements	Excess Reactivity (% $\Delta k/k$)
20	0.15
25	4.18
31	7.85

The worth of the B_4C control rods at 540°F was found to be approximately 17% greater than at 345°F. This is a slightly greater increase than observed by Kunze and Poole in a similar study.² Although the hydrogen density of the coolant does increase with the amount of high-boiling decomposition products present in the coolant, essentially no change in rod worth was observed with coolant containing 0% and 22% of these products.

¹ C. A. Trilling, "Organic Moderated Reactor Experiment First Progress Report October, 1955 - July, 1956," NAA-SR-1700 (1957).

² J. F. Kunze and R. N. Poole, "K_{excess} and Control Rod Worth Versus Temperature For A Heterogeneous Core," Paper 21-8, *Am. Nuclear Soc.*, 4, 1(June 1961), p. 106.

Fig. 2. Temperature Coefficient vs Core Size

Nuclear Reactor Physics

Measurements of Reactor Parameters in Subcritical and Critical Assemblies: a Review, *I. Kaplan (MIT)*.

The purpose of this session is to consider the status of measurements of nuclear parameters of reactors in subcritical and critical assemblies. The emphasis will be on similarities and differences between the results obtained from measurements of individual parameters in these two types of assemblies. Measurements in both fast and thermal assemblies will be discussed.

The purpose of this paper is to point out some of the problems and to provide the background for the more detailed discussions of the later papers.

Since Fermi's early work on exponential experiments the design of reactors has depended heavily on the experimental determination of nuclear properties of multiplying assemblies in subcritical or critical experiments. In thermal assemblies the nuclear properties studied have included macroscopic flux traverses, from which the material buckling can be obtained, and ratios related to quantities which enter into the four-factor formula for k_{∞} . One ratio measured in uranium fueled assemblies is:

$$\delta_{238} = \frac{\text{fission rate in U-238}}{\text{fission rate in U-235}}$$

which is related to ϵ , the fast fission effect. Another quantity which can be measured is R_{238} , the average U-238 cadmium ratio for a fuel element. It is sometimes convenient to use the ratio, ρ_{238} , which is related to R_{238} :

$$\rho_{238} = \frac{1}{R_{238} - 1} = \frac{\text{epithermal capture rate in U-238}}{\text{thermal capture rate in U-238}}$$

Either ratio, R_{238} or ρ_{238} , can be related to the resonance escape probability or the conversion ratio. Microscopic (intracellular) flux distributions can be measured, and values of disadvantage factors and thermal utilization can be obtained from them. In lattices of slightly enriched uranium rods in ordinary water it has also been possible to determine, indirectly, the migration area or age, and thus to obtain an experimental link between the macroscopic and microscopic properties of such lattices.

The measurements mentioned have been used for two purposes: to develop semiempirical calculational schemes, "recipes", for the design of real reactors; and to provide experimental information with which theory can be compared, in the hope of arriving, eventually, at a real understanding of the physics of thermal reactors. A serious problem that arises is that of the possible difference between results of measurements of a particular quantity when made in a critical assembly and in a subcritical assembly. For example, recent work at the Savannah River Laboratory has indicated significant differences between values of the material buckling obtained from exponential and critical assemblies moderated by heavy water. A similar effect does not seem to have been observed in assemblies moderated by graphite or ordinary water.

It seems appropriate, therefore, to review work done up to now on thermal reactors to see where differences do exist and to seek the reasons for such differences. The papers in this session will treat some of these problems. The systems to be considered are: uranium-graphite,

uranium (or uranium oxide)—ordinary water, and uranium (or uranium oxide)—heavy water.

Experiments on fast critical and exponential assemblies have been made to determine critical buckling, neutron energy spectra and other nuclear properties, and to develop sets of cross sections and other nuclear parameters for use in detailed calculational schemes. Recent work has again indicated significant differences, both between theoretical predictions and experimental results and between experimental results obtained in critical and exponential assemblies. A careful consideration of the state of affairs in this field also seems worthwhile.

Measurements of Reactor Parameters in Subcritical and Critical Assemblies—Lattices of Slightly Enriched Uranium in Water, *Glenn A. Price (BNL)*.

A number of subcritical measurements involving slightly enriched uranium fuel rods in water have been performed at BNL in the past 10 years. They are part of a parametric study of the various neutron reactor factors as a function of uranium enrichment, fuel rod diameter, and water-to-fuel volume ratio. Generally speaking, the measurements are interpreted in terms of the four factor critical reactor formula, $\eta \epsilon p f$, with either an age theory or one group leakage probability. Techniques have been developed to measure each of the constants in the critical equation more or less independently, with the exception of η . Thus the value of η can be inferred from a knowledge of all the other constants of the critical equation.

Many of the lattices which were investigated at BNL as subcritical assemblies were also studied in critical assemblies at Westinghouse (Bettis).¹⁻³ In some cases the techniques at the two laboratories were similar, and in other cases the techniques were different. There were also a few critical experiments at BNL. The subcritical experiments at BNL fall into two classes: miniature lattice measurements and exponential measurements. Values of ϵ , p , and f were obtained in small assemblies (miniature lattices) which have a neutron multiplication of about 3, and in which the central neutron spectrum is characteristic of a critical assembly. Values of M^2 and B^2 were measured in larger assemblies (exponential assemblies) in which the neutron multiplication is between 3 and 50. Thus, there are considerable data to compare reactor parameters as measured in similar lattices by subcritical and critical techniques.

The data in this paper refer to uranium-metal fuel rods with aluminum cladding, immersed in ordinary water. The uranium is slightly enriched to concentrations

TABLE I. Comparison of Criticality Studies with 0.600-in. Rods of 1.3% Enrichment

Vol. H ₂ O Vol. U	Critical Number of Rods	
	BNL (predicted)	Bettis (observed)
1.5	486.6 ± 7.4	478
2	337.4 ± 2.9	335
3	265.6 ± 1.9	266

TABLE II. Comparison of Material Bucklings with 0.387-in. Rods of 1.3% Enrichment

Vol. H ₂ O Vol. U	B ² (cm ⁻² × 10 ⁴)	
	BNL	Bettis
2	52.19 ± 0.36	53.55 ± 0.48
3	59.25 ± 0.33	58.2 ± 1.0

TABLE III. Comparison of Migration Areas (One Group Leakage Probability) with 0.387-in. Rods of 1.3% Enrichment

Vol. H ₂ O Vol. U	BNL (exponential)	BNL (critical)	Bettis (critical)
2	34.84	35.2	34.2
3	32.02	32.2	36.2
4	30.72	32.3	

TABLE IV. Comparison of Reflector Savings with 0.387-in. Rods of 1.3% Enrichment

Vol. H ₂ O Vol. U	λ, cm	
	BNL	Bettis
2	7.56 ± .08	7.40 ± .17
3	6.94 ± .07	7.22 ± .31

TABLE V. Comparison of Thermal Utilization, f, with 1.3% Enriched Rods

Rod Diam.	Vol. H ₂ O Vol. U	BNL (miniature)	Bettis (critical)
0.600"	3	0.846	0.843 ± .003
0.387"	2	0.904	0.904 ± .003
0.387"	3	0.865	0.857 ± .004

TABLE VI. Comparison of Resonance Escape Probability, p

Rod Diam.	Vol. H ₂ O Vol. U	1% BNL (miniature)	1.15% Bettis (critical)	1.3% Bettis (critical)
0.600"	1	0.720 ± .027		0.660 ± .013
0.600"	2	0.838 ± .004	0.824 ± .006	0.833 ± .017
0.600"	3	0.871 ± .003	0.859 ± .007	0.877 ± .017
0.387"	2	0.787 ± .009		0.788 ± .006
0.387"	3	0.847 ± .005		0.838 ± .005

TABLE VII. Comparison of Fast-to-Thermal Fission Ratio, δ₂₈

Rod Diam.	Vol. H ₂ O Vol. U	Enrichment	BNL (subcritical)	Bettis (critical)
0.600"	2	1.15%	0.113	0.104 ± .015
0.600"	3	1.15%	0.086	0.081 ± .012
0.387"	2	1.3%	0.108	0.099 ± .012

of 1.15% and 1.3% U-235. Uranium rod diameters are 0.600 in. and 0.387 in. The aluminum cladding is 0.028 in. thick, with a nominal 0.005 in. air gap between fuel and clad. These data include only a small fraction of the subcritical and critical experiments, but are typical of those areas where the data may be compared directly.

There have been a few instances in which a variation in technique has led to systematic differences in the measurements. For example, the use of aluminum foil holders

in the thermal utilization measurements at BNL was found to perturb the neutron flux. When the measurements were repeated with Lucite foil holders, the subcritical and critical measurements were found to agree with each other.

In the case of some measurements, it is possible to interpret the data in several different ways. For instance, measurements of migration area may be interpreted by age-diffusion, age, one group, or multigroup theory. Numerical values of M² as computed by the different methods will vary considerably, although each will form a consistent set of reactor parameters in its own critical equation. In addition, there appears to be agreement in values of M² as measured by subcritical and by critical methods. Thus, it is necessary to make any comparison between critical and subcritical experiments with a common theoretical interpretation.

It is always possible that a systematic error may not be common to both subcritical and critical measurements. For instance, recent investigations at BNL have revealed a minor difference between buckling measurements which were done at long relaxation lengths and those which were done at short relaxation lengths. The effect was traced to a source of high-energy gamma radiation from the thermal column which caused a distributed source of photo-neutrons throughout the lattice. The effect on the buckling measurements was small in this particular case, although very disconcerting.

As experimental techniques continue to improve, systematic errors in technique or interpretation of data will be revealed, but it appears that these systematic errors are also rapidly approaching a negligible limit. At the present time it appears that the reactor parameters of slightly enriched-uranium water lattices, as measured in subcritical assemblies and in critical assemblies, are in agreement within the limits of experimental errors.

¹ H. Kouts et al., "Exponential Experiments with slightly Enriched Uranium Rods in Ordinary Water" *Proc. 1st UN Conf. on Atomic Energy—Geneva 1955*, 5, 183. United Nations, New York (1955).

² H. Kouts et al., "Physics of Slightly Enriched, Normal Water Lattices," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 446, United Nations, New York (1958).

³ G. A. Price, "Migration Areas and Effective Delayed Neutron Fractions by Critical Experiments," *J. Nuclear Energy*, 10, 111 (1959).

Measurements of Natural-Uranium, Heavy-Water Lattices, J. L. Crandall (DuP-SR).

Physics experimentation on heavy water reactor lattices is currently a very active field with at least a dozen independent laboratories performing measurements. However, owing to the high cost of D₂O and the comparatively large size of critical D₂O lattices most experiments are performed by exponential or substitution techniques or in reflected criticals. Studies at Savannah River have investigated the possibility of systematic errors arising from such techniques. Some fifty full-scale lattice loadings have been made in the Process Development Pile at SRL to provide reference buckling numbers for simple one-region criticals.¹ Both natural uranium metal and oxide fuel were used over a wide range of fuel assembly sizes and lattice pitches. Although generally good agreement was found with other existing work, a detailed comparison with exponential studies also made at SRL showed definite anomalies in the exponential bucklings.² Such results had also been observed earlier between NAA exponentials³ and AECL criticals.⁴ The effects were traced

to variations in the radial buckling of the exponential in excess of those which would be predicted by a two-group calculation of changes in the neutron extrapolation distance. The measured variations depended both on the moderator-to-fuel ratio and the degree of heterogeneity in the lattice. Rather similar anomalies were also found in studies of reflected exponential lattices.

Techniques for studying detailed neutron processes are generally similar in all reactor systems. However the studies in D_2O lattices emphasize the essential heterogeneity of such lattices. At normal fuel spacings of six inches or greater, fast fission is almost exclusively a function of the individual fuel assembly design rather than of the lattice design. Resonance capture also is often found to be more an intra- rather than an intercell process. In fact, in many cases the fuel assemblies can be sufficiently separated so that they are coupled primarily by thermal neutrons, i.e., complete neutron thermalization takes place between them. However in these cases large fuel assemblies are generally used and there is a compensating effect of flux hardening within the fuel.

A feature which D_2O shares with other liquid moderators is large density changes on heating or boiling. However, the resulting effects differ strongly from those in neutron absorbing moderators like H_2O .

In H_2O , the thermal utilization aspects of the density changes are controlling, directly in neutron absorption changes and indirectly in forcing the lattice designs towards the undermoderated side where moderator density changes strongly affect neutron thermalization and leakage.

In D_2O , these effects are minimized. In the temperature coefficient studies the main problems arise in evaluating the changes in η . In the boiling or void studies the major problems are in resonance capture determinations. The complication of void streaming also enters. Currently the experimental data on this latter point seem best matched by a theory due to Benoist.⁵ The effects of void location have been a particular problem and have received extensive study by a number of groups. A one-group perturbation theory analysis developed by Persson has proven a particularly powerful tool in attacking these problems.⁶

¹T. J. Hurley et al., "Experimental Bucklings of Heavy Water Moderated Lattices of Natural Uranium Rod Clusters", *Nuclear Sci. and Eng.* 12, 341 (1962).

²E. C. Wingfield and E. J. Hennelly, "Exponential Measurements of Natural Uranium Rods in Heavy Water and Comparison with Critical Experiments", *Nuclear Sci. and Eng.*, 12, 348 (1962).

³E. R. Cohen, "Exponential Experiments on D_2O -Uranium Lattices", *Proc. 1st UN Conf. on Atomic Energy—Geneva, 1955*, 5, 268 United Nations, New York, 1955.

⁴D. W. Hone et al., "Natural-Uranium Heavy-Water Lattices, Experiment and Theory", *Proc. 2nd UN Conf. on Atomic Energy—Geneva, 1958*, 12, 351 United Nations, New York, 1958.

⁵P. Benoist, "Formulation Générale et Calcul Pratique du Coefficient de Diffusion Dans un Réseau Comportant des Cavités", *Reactor Science*, 13, 97 (1961).

⁶R. Persson, "One-Group Perturbation Theory Applied to Substitution Measurements with Void", *RFX-74* (Jan. 1961).

Measurements of Reactor Parameters in Subcritical and Critical Assemblies: Measurements in Uranium-Graphite Lattices, R. E. Heineman (GE-HAPO).

At the present time there exists an exceedingly large number of measurements of reactor parameters for graphite uranium systems.

The literature on the correlation of these data is in itself large. The problems involved in this correlation are complex because of the interrelations between the various parameters, the varying methods of obtaining them, the large number of such parameters, and the fact that there exist many published values of questionable accuracy.

The scope of this subject is here limited to a comparison of the values of the buckling B^2 and the excess infinite multiplication factor $k_{\infty}-1$, which are variously derived from critical and subcritical experiments. These two integral parameters can be measured more or less directly. They are about equally important to the reactor physics design of low-enrichment graphite reactors, since the real question is the size of the assembly which gives a sufficient margin of excess effective k to allow the reactor to reach the planned operating conditions.

There have been but a few critical experiments for natural uranium-graphite systems. Some of these critical systems were designed for high power operation and could not be compared directly with the relatively clean exponential pile assemblies. One such series¹ of criticals required corrections of from 11 to $34 \times 10^{-6} \text{ cm}^{-2}$ in B^2 . The residual differences from this study of the order of 5×10^{-6} (about 3×10^{-3} in k) are not at all unreasonable in view of the possible errors. These include,² but are not limited to: ± 2 in statistical precision on the exponential data, ± 2 in estimates of extrapolation distances for the exponential piles (8 ft $\times$ 8 ft size), ± 1 in the precision of the critical data, and at least ± 1 to 2 in the corrections made to the values from the critical systems, all in units of 10^{-6} cm^{-2} . Comparison³ of the critical size of one of the Calder reactor loadings to calculations based on exponential bucklings also involved substantial corrections. The difference of 4×10^{-3} in k which remained after refined calculations were made was believed to be barely significant. Two additional, and more direct, comparisons⁴ with slightly enriched uranium fuel agreed to within the errors of the measurements. However, in these cases the exponential piles were not far from critical.

A relatively new and unique facility at Hanford Laboratories, the Physical Constants Testing Reactor (PCTR), is used to obtain⁵ values of $k_{\infty}-1$ with much smaller samples of a lattice than those used in exponential piles. The values of $k_{\infty}-1$ from critical experiments depend upon measured bucklings and the delayed neutron fractions. Comparisons⁶ of values of $k_{\infty}-1$ derived from these types of experiments show them to be in good agreement. The experimental errors on these values do not preclude systematic differences of the order of 2 to 5×10^{-3} in k for natural uranium systems. An additional comparison⁷ is now available for a paraffin-moderated, enriched uranium system. After measurements of k_{∞} were completed, a critical assembly was stacked at Oak Ridge National Laboratory. The two measured values of $k_{\infty}-1$ agreed within their experimental errors of $\pm 13 \times 10^{-3}$ and $\pm 12 \times 10^{-3}$ for this rather high- k medium.

The measurements of $k_{\infty}-1$ in the PCTR and those of B^2 from the exponential piles have been compared,⁸ by use of a calculated migration area M^2 . The use of standard formulae for M^2 gave values of $k_{\infty}-1$ from exponential bucklings which appeared to be systematically high by

about 4×10^{-3} on the average. However, the lattices measured were not exactly the same in many of the cases and corrections or interpolations were necessary. In one case, where the disagreement was very large, the buckling was measured in a four foot exponential pile, in which the uncertainty in the extrapolation distance easily accounts for the difference. More recent comparisons¹⁰ for two lattices have shown agreement to within 3×10^{-3} in k . The above comparisons are, of course, all limited by the uncertainties in the values of M^2 of 4 or 5%.

The question of critical versus subcritical measurements can not be discussed without speculation on possible reasons for differences. The fundamental difference between the systems pertains to neutron leakage. Although the net leakage in an exponential pile and in a critical pile is the same, $M^2 B_m^2$, the signs and magnitudes of the various spatial components are different in the two piles. These differences are the larger, the smaller the exponential pile. This has not been studied to the writer's knowledge. Other reasons for differences are primarily associated with the subcritical technique or the pile size. These are related reasons since the techniques must become more sophisticated as the exponential pile decreases in size. Is the pile so small that the homogeneity assumptions are not valid, that the flux distribution does not reach equilibrium, or that the uncertainties in extrapolation distance are excessive? These questions have been the subject of intensive study recently.¹¹⁻¹³

The net leakage of neutrons of all energies from a lattice cell in the PCTR is deliberately reduced to as low a value as possible. Thus, the difference between k_∞ and k_{∞}^* , as introduced by Weinberg and Wigner,¹⁴ is in question. However, calculations of f^*-f for one lattice show entirely negligible differences. The same conclusion may be drawn from PCTR measurements, at least for cells in which the flux gradient is close to zero. Values of f measured in a lattice in the PCTR are essentially the same as the values of f_p in the same lattice poisoned at the cell edge with a thermal absorber such that all excess neutrons "leak" into the poison as thermal neutrons.

It can be concluded that no differences in B^2 or $k_{\infty}-1$ have been found in these critical and subcritical experiments which are outside of the present experimental errors.

¹P. F. Gast, "Normal Uranium, Graphite-Moderated Reactors: A Comparison of Theory and Experiment—Water-Cooled Lattices," *Proc. 1st UN Conf. on Atomic Energy—Geneva 1955*, 5, 288, United Nations, New York (1955).

²D. E. Davenport, "Exponential Experiments in Graphite Systems," *Proc. 1st UN Conf. on Atomic Energy—Geneva 1955*, 5, 309, United Nations, New York (1955).

³B. Cutts et al., "Graphite Moderated Thermal Reactor Calculations for the Calder Hall Reactors," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 612, United Nations, New York (1958).

⁴C. G. Campbell and I. S. Grant, "Critical and Sub-Critical Experiments with a Two-group Correlation of Results," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 728, United Nations, New York (1958).

⁵D. J. Donahue et al., "Determination of K_∞ from Critical Experiments with the PCTR," *Nuclear Sci. and Eng.*, 4, 297 (1958).

⁶R. E. Heineman, "Experience in the Use of the Physical Constants Testing Reactor," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 650, United Nations, New York (1958).

⁷V. I. Neeley, " k_∞ Measurements of 2 Per Cent U-235 Enriched UF₄ in Paraffin," *Trans. Am. Nuclear Soc.*, 2, 1, 67 (1959); also HW-62031 (1959).

⁸D. D. Lanning et al., "The Nuclear Parameters of Some Graphite, Natural Uranium Lattices Measured in the PCTR," *Proc. Conf. on Graphite Reactors*, BNL-489, p. 85 (1958).

⁹D. E. Wood, "Material Buckling Measurements on Graphite-Uranium Systems at Hanford: A Summary Tabulation," HW-69525 (May, 1961).

¹⁰D. E. Wood, private communication.

¹¹D. E. Wood, "Exponential Piles in Teaching and Research," *Proc. Univ. Subcritical Assemblies Conf.*, TID-7619, p. 222 (1961).

¹²D. E. Wood, "Small Source Theory Oscillations in Exponential Pile Flux Distributions," *Trans. Amer. Nuclear Soc.*, 4, 2, 284; See also HW-70716, p. 22 (1961).

¹³G. W. R. Endres and D. E. Wood, "Investigation of Extrapolation Lengths for Graphite Uranium Lattices," *Nuclear Physics Research Quarterly Report*, HW-67219, p. 75 (1960).

¹⁴A. M. Weinberg and E. P. Wigner, *The Physical Theory of Neutron Chain Reactors*, The Univ. of Chicago Press, p. 173 and p. 733 (1958).

Measurements in Fast Assemblies at Argonne, R. Avery (ANL).

During the past seven years many fast exponential and critical experiments have been performed at Argonne National Laboratory. The interest in fast reactors has mainly been related to power-breeder systems operating with fuel recycle on the U-238-Pu chain. The systems that have been experimentally studied differ from the systems of ultimate interest mostly in that they are smaller, less dilute with harder neutron spectra, and fueled with U-235 rather than Pu-239. The continuing trend has been to study systems progressively larger and more dilute, and in particular to focus attention on the problems that are of most importance in large dilute systems, such as the Doppler and sodium void coefficients. Also, the study of larger systems leads to an increased interest in the low energy end of the neutron spectrum, < 50 kev.

The experiments have been performed mostly on three facilities: ZPR-IV, a standard exponential facility; ZPR-V, an internal exponential driven by an annular water-moderated thermal system; and ZPR-III, a two-half flexible critical facility. Two larger flexible critical facilities, ZPR-VI and ZPR-IX, will soon be available.

The experiments that have been performed are standard. Critical configurations have been determined for many different compositions. Reactivity coefficients have been determined by period measurements or by use of control rods calibrated by period measurements. Reactivity worths have also been determined by analysis of the response to oscillator and rod drop experiments. Rossi- α measurements have been made on a number of assemblies to determine the ratio of the effective delayed neutron fraction to the prompt neutron lifetime and thus indirectly to give information on the neutron spectrum. Detector responses, both as a function of detector material and as a function of position, have been made to determine data relevant to power distributions, bucklings, reflector savings, and neutron spectra. Spectrum measurements have been made by use of emulsion plates.

The systems studied on the various facilities have included a wide range core and blanket compositions, fuel enrichments, sizes, and shapes.

The analysis has been almost exclusively done through the application of multigroup theory, with the calculated integral data compared to the measured values. The multigroup constants, which for the most part have been obtained from microscopic basic data, are sometimes modified to improve agreement with experiment. In general this is done only if it leads to systematic improvement over a wide range of systems and also only if the modification does not conflict with measured cross sections. Predictions of reactor behavior on other systems or on characteristics not directly measured are then made on the basis of multigroup calculations and not by way of direct application of the integral data.

Analysis of all of the experimental data leads to the conclusion that when the best available microscopic data are used to generate multigroup constants, there are serious systematic errors between theory and experiment. In particular, there is strong evidence both from Rossi- α and various detector measurements that the true spectrum is almost always softer than calculated. Also the systems are usually less reactive than predicted. Some of the systematic discrepancy has been removed by use of more rigorous derivations of transport and elastic transfer cross group constants which more properly account for the resonance structure of light materials. Recent measurements yielding a lower rate for the variation of k_{∞} with energy has also substantially improved agreement. Nevertheless, the experimental assemblies are still often found to be from 2 to 5% less reactive than predicted. The largest discrepancies occur in systems with substantial amounts of steel.

Comparison between the results of the critical and exponential experiments are difficult, since there has not been too much overlap of the assemblies investigated. The indirect use of the integral data which, in the analysis, serves only as a check on the assumed multigroup constants, also leads to difficulties in the comparison since the quantities measured in the two types of experiments are usually different. No systematic difference of accuracy has been observed in the extent of agreement between theory and measurement in the two types of experiment. The critical experiments have been far more useful, however, in that the precision of the experimental results is much greater.

Future exponential experiments will almost certainly be of the multizone type where the central test region is surrounded by a driver region, perhaps with an intervening buffer region, with compositions quite similar to that of the central zone. Such loadings will be particularly useful in the study of plutonium-fueled systems, in that the amount of plutonium required will be much reduced. Numerical studies indicate that with large systems, central regions can be made sufficiently large to yield good results and still lead to very appreciable reductions of material of the composition of interest.

It is also planned to use multizone loadings to study the Doppler effect in large dilute systems. The central heated zone will be sufficiently large to yield an equilibrium spectrum characteristic of the test zone temperature over most of its volume.

The Los Alamos Fast Critical Program, Hugh C. Paxton (LASL).

The Los Alamos fast critical program has been directed toward the development of realistic parameters to

feed detailed computational schemes. This objective has materialized as the Hansen-Roach cross-section set.¹ Typical of the high-energy realm, are comparisons of U-233 metal assemblies with results of earlier calculations. Measured critical masses of U-233 (98.1%) at 18.6 g/cm³ are 16.1 kg for a bare sphere and ~5.5 kg for a sphere reflected by 7½ in. thick natural uranium, as compared with S_4 values of 16.0 kg and 5.6 kg.²

Because of such satisfactory agreement, our work with small, simple assemblies now concentrates on experimental discrepancies (e.g., uncertainties about the effectiveness of Pu-240, and absolute values of spectral indices), certain elusive parameters (such as the ratio of capture to fission in a thorium reflector, and effective extrapolation distance for shape conversion), and effects of dilution (as various metals mixed with U-235). Some recent results in these categories follow.

(1) In a 1.6-kg Pu sphere centered within a U reflected U(93) core, each gram of Pu-240 behaves like 0.69 g Pu-239 within the same volume.³ An interpretation in terms of the most sensitive nuclear parameter is $k_{\infty}(\text{Pu-240, 2 Mev}) = 3.32 \pm 0.14$.

(2) In an 8.4-in. thick Th reflector about a Pu core, Grundl observes capture-to-fission ratios for Th-232 ranging from 8.3 at a distance of 0.9-in. outside the core, to 53 at 5.9-in.

(3) Table I, due to Hansen and Wood, gives a few reflector savings measured on the plane surfaces of 15-in. and 21-in. diameter U(93) discs, compared with computed values for infinite slabs.

TABLE I. Reflector Savings per Face of
U(93.2) Slabs - kg U/in²

Reflector (6" thick)	15" Diameter Slab (obs)	21" Diameter Slab (obs)	Infinite Slab (S_4)
Water	0.277 ± .002	0.288 ± .002	0.306
Polyethylene	0.283 ± .002	0.290 ± .002	0.304
Lucite	0.297 ± .002	0.308 ± .002	0.326
critical values in kg U/in ² :			
None	0.942 ± .003	0.873 ± .003	0.813

(4) Table II summarizes critical data of Jarvis for some diluted U(93) systems.

TABLE II. Bare Critical Spheres of Diluted U(93.2)

Values of n in: $(U \text{ critical mass}) \propto (\text{vol. fraction } U)^{-n}$
= F^{-n} (from data for bare cylinders)

diluent:	Al	Fe	Ni	Cu	Zn	Zr	Ta	W
n :	1.33	1.20	1.10	0.97	1.16	1.16	1.30	1.03
min. obs. F :	0.42	0.38	0.38	0.38	0.38	0.39	0.49	0.31

A specialized example is a mockup to assist Livermore with the design of a high yield burst reactor. The assembly was an annulus of U averaging 27% U-235, 15-in. ID × 21-in. OD × 11.8-in. high, capped with a 4.9-in. thick U-U(27) composite. Calibration techniques are illustrated by Orndoff's measurement of reactivity vs. width of gap across the median plane.

To discuss very high energy systems as a distinct class would be a deceptive oversimplification. The Hansen-Roach set has potential applicability to the entire family of homogeneous systems that is represented in this session, and it seems appropriate to include a final illustration in the intermediate energy range. Near-homogeneous graphite-hydrogen-U(93) assemblies were set up by Hoogterp to check results of Stratton's DSN survey for nuclear safety guidance. For a bare sphere at H/U-235 = 12.1, C/U-235 = 97.5, and $\rho(\text{U-235}) = 0.261 \text{ g/cm}^3$, the experimental critical mass (adjusted to a sphere) is 66.3 kg U-235 against 67.1 kg computed. With a Lucite reflector the corresponding values are $\sim 28.9 \text{ kg}$ and 30.4 kg .

¹ G. E. Hansen, W. H. Roach, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543 (1961).

² G. E. Hansen, "Status of Computational and Experimental Correlations for Los Alamos Fast Neutron Critical Assemblies," *Proc. IAEA Seminar on the Physics of Fast and Intermediate Reactors*, Vienna (1961).

³ D. M. Barton et al., "Critical Masses of Composites of Oy and Pu-239-240 in Flattop Geometry," LAMS-2489 (1961).

Reactor Theory—I

The Influence of Core Shape on Fast Reactor Criticality, W. B. Loewenstein and G. W. Main (ANL).

A large fraction of the reported fast neutron critical experiments were investigated in cylindrical geometry. Analytical efforts to interpret the experimental data and test microscopic multigroup constants utilize reflected spherical geometry. It has been demonstrated that spherical calculations do give the spectral properties of the cylindrical critical assemblies. However, there must be a basis for converting the predicted spherical critical mass to cylindrical geometry for comparison with the experimental data. This is generally accomplished by the use of a suitable Shape Factor.

The Shape Factor (S.F.) is defined as the ratio of the spherical critical mass of a system to the critical mass of a given assembly. Both systems have identical core compositions and similar reflector parameters.

Four fast reactor systems have been extensively analyzed for typical Shape Factor data. These represent systems that may be encountered in reactor design as well as in analysis of critical experiments. The systems are combinations of high and low-density cores and reflectors. Table I summarizes the compositions of the four systems.

Reactor No. 1 is similar to ZPR-III Assembly 25.¹ Results from this analysis probably will apply to ZPR-III Assemblies 22 and 24,¹ as well as to the 16.25% enriched Jemima.² Reactor No. 2 represents a typical fast power breeder reactor system. Reactor No. 3, with a low-density core and a high-density reflector, is typical of many systems investigated at ZPR-III (Assemblies 15, 29, 30, 31).^{1,3} Reactor No. 4 has no experimental analogue.

The Shape Factor analyses utilized three-energy-group diffusion theory in spherical and two-dimensional r,z geometry. Table II is a summary of the analyses. It may be seen that the optimum geometry Shape Factor is a function of the reactor system. It is clearly not an invariant quantity. Furthermore, the variation of critical mass with core geometry also is a function of the specific reactor system.

The net neutron leakage from the core is an important contributor to the Shape Factor determination. It is therefore reasonable to expect that the Shape Factor (S.F.) is strongly dependent upon the ratio of the spherical core surface area (S_s) to the cylindrical core surface area (S_c). All the data in Table II were found to satisfy

$$S.F. = a \frac{S_s}{S_c} + b,$$

where a and b are constants. A fair fit may be obtained with $a = 1$ and $b = 0.1$; a good fit requires $a > 1$.

TABLE I. Core and Reflector Compositions of the Four Reactor Systems

Reactor number	Density		Volume Fractions ^a							Approx. spherical core volume (liters)
			Core			Reflector				
	Core	Reflector	U-235	U-238	Fe	Na	U	Fe	Na	
1	High	High	0.0712	0.7405	0.0928	0	0.835	0.0731	0	560
2	Low	Low	0.0447	0.2553	0.20	0.50	0.60	0.20	0.20	850
3	Low	High	0.0581	0.0914	0.245	0.376	0.835	0.0731	0	380
4	High	Low	0.0712	0.7405	0.0928	0	0.60	0.20	0.20	550

^a Volume Fractions $\times 100$ = Volume %; the total volume fraction is not unity because voids exist.

TABLE II. Summary of Shape Factor Analyses

Reactor No.	L/D	V(liters) ^a	Shape Factor	Reactor No.	L/D	V(liters) ^a	Shape Factor
1	0.30	946	0.554	3	0.32	586	0.624
	0.49	678	0.773		0.45	480	0.762
	0.68	590	0.888		0.59	431	0.849
	0.78	571	0.918		0.78	406	0.901
	0.97	568	0.926		1.17	414	0.884
	1.32	602	0.871		1.48	437	0.837
	1.97	732	0.716				
2	0.27	1696	0.501	4	0.50	662	0.790
	0.36	1283	0.661		0.68	581	0.900
	0.57	993	0.855		0.78	570	0.917
	0.78	923	0.920		0.96	569	0.919
	0.87	909	0.934		1.32	599	0.873
	0.96	923	0.920				
	1.44	995	0.853				
	2.16	1203	0.705				

^a Core volume.

It is of considerable interest to determine to what extent rather simple results may be used for future analyses. The criticality equation

$$\kappa^2 = \left(\frac{\pi}{S + s} \right)^2 = \left(\frac{2.405}{R + \delta} \right)^2 + \left(\frac{\pi}{L + 2\Delta} \right)^2,$$

where S , R , and L are the calculated spherical radius, cylindrical radius, and core height, respectively, and s , δ , and Δ are the spherical, radial cylindrical, and axial cylindrical "reflector savings," respectively, was applied to the results of the detailed analyses. The material buckling, κ^2 , is obtained from a multigroup fundamental mode analysis. It was found that, in general, axial and radial reflector savings are not equal. However, the differences are small in near optimum geometry systems. Therefore, the self-consistent one dimensional diffusion theory analysis³ is found to be adequate for all but the most extreme and exacting investigations. Larger errors may be incurred by applying the spherical reflector savings in a cylindrical analysis. In almost all cases the spherical reflector saving is less than those appropriate for cylindrical systems.

¹ J. K. Long et al., "Experimental Results on Large Dilute Fast Critical Systems with Metallic and Ceramic Fuels," Paper Presented at IAEA Sponsored Seminar on the Physics of Fast and Intermediate Reactors, Vienna, (1961), Paper SM-18/48.

² G. E. Hansen, "Properties of Elementary Fast Neutron Critical Assemblies," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 84, United Nations, New York (1958).

³ J. K. Long et al., "Fast Neutron Power Reactor Studies with ZPR-III," *Proc. 2nd UN Conf. on Atomic Energy—Geneva 1958*, 12, 119, United Nations, New York (1958).

Critical Experiments

Critical Mass Calculations of a Multi-spectrum Critical Assembly, L. A. Mountford and V. E. Kistler (AI).

The Advanced Epithermal Thorium Reactor critical assembly¹ is used for making physics measurements to aid in the design of a power reactor based on the Th-U-233 fuel cycle. Since the available supply of U-233 is less than that required for a full-scale critical assembly, a multiregion assembly is used. The regions are the central test region and surrounding concentric buffer, decoupler, driver, and reflector regions. Buffer and decoupler regions produce a change from the thermal neutron spectrum of the driver region to the fast or epithermal spectrum of the test region. The driver region provides most of the fissions to make the reactor critical. The reflector serves to reduce the size of the driver region necessary for criticality. Figure 1 shows the median absorption energy in the reactor as a function of radius.

Fig. 1. Median Flux Energy as a Function of Radius in AETR Critical Assembly Core 1—an Unmoderated U-235 Fueled Assembly.

Although measurement of the critical mass of a full-scale AETR is not part of the experimental program, the critical mass of each assembly has to be calculated and measured to guide the fuel loading. For these calculations the spherical reactor is divided into concentric regions, with the composition and size of all except the driver being determined by their functions. Individual regions are divided into sub-regions to reduce errors caused by homogenizing them. The size of the driver region necessary for the reactor to be critical is then calculated. An 18 energy group cross section library is used with the AIM-6 diffusion code.² Cross sections are basically from the LAMS-2255 and LAMS-2543. Self-shielding corrections are applied to the Th, U-233, and U-235 cross sections. Thorium cross sections are corrected by evaluation of the effective resonance integral for each resonance. For the uranium isotopes, the differences between calculated infinitely dilute and self-shielded resonance cross sections are subtracted from cross sections given in the above library in order to obtain the self-shielded cross sections.

TABLE I. Description of AETR Cores 1-5 and Comparison of their Calculated to Measured Critical Systems

Core	Fuel	Test Region		k_{eff}^a (calc)
		Moderator/fuel ratio	Median fission energy (kev)	
1	U-235	0	422	1.0158
2	U-233	10.6 (C)	120	1.0013
3	U-235	4.57 (C)	140	1.0150
4	U-233	30 (C) 370 (Be)	0.0099	1.0122
5	U-233	24.2 (C)	66	1.0057

^a Measured $k_{eff} = 1.000$.

Table I gives a brief description of the first five cores and indicates the differences between their calculated and experimental critical masses. Accuracy of the calculations is given by the k_{eff} calculated for the measured critical reactor configuration.

¹ H. A. Morewitz et al., *Trans. Am. Nuclear Soc.*, 4, 55 (1961).

² H. P. Flatt and D. C. Baller, "AIM-6 Code, NAA Program Description" (1961).

Description and Analysis of Further Critical Experiments with Double Annular Fuel Elements in Superheater Geometries, P. G. Klann, C. O. Dechand, E. Fein, and M. F. Gormley (CE).

The previously reported¹ critical experiments and analysis of low enriched, light water moderated, double annular fuel elements in clean uniform arrays have been extended to more complex geometries. The cylindrical cores consisted of 19 hexagonal-shaped subassemblies, each containing 19 fuel elements spaced on a 1.915-in. triangular pitch. Figure 1 shows a typical subassembly. Measurements were made with Y-shaped control rods completely inserted, partially inserted, and completely withdrawn, and with the steam coolant spaces either flooded or voided.

Also measured were the reactivity effects of columns of pyrex spheres from 3/8 to 7/8-in. diam and circular pyrex rods from 5/16 to 1-in. diam. It was found that, for the same height, a column of spheres has the same reactivity effect as the rod of the same volume. The lateral projected area of the rod is thus 4% greater than that of the equivalent column of spheres.

The method of analysis was similar to that used for the clean lattices.¹ The two-dimensional transverse diffusion calculations were made using the PDQ Code. The analysis thus provided a test of the validity of representing the

Fig. 1. First Ring Subassembly.

hexagonal geometry of the fuel assemblies in a rectangular coordinate system to take advantage of the relative simplicity of the PDQ Code. For cores containing partially inserted rods, a three-dimensional representation was obtained with axial synthesis calculations using flux weighted constants and group-dependent bucklings obtained from the transverse PDQ calculation.

TABLE I. Calculated Eigenvalues for Double Annular Elements Arranged in Superheater Geometry

Description of core	Water height (in.)	Calculated eigenvalue	Experimental eigenvalue
Clean core; 361 flooded elements	35.01	1.009	1.000
Central subassembly elements voided; remainder flooded; 361 elements	36.24	1.007	1.000
Center water hole; 360 flooded elements (central element removed)	35.94	1.010	1.000
7/8" diam Pyrex sphere column substituted for central fuel element; 360 flooded elements	36.85	1.013	1.000
Fully inserted cadmium Y rod	41.56	1.010	1.000
Three central boron Y rods inserted to 29.95" from bottom of fuel	Full up	1.011	1.000
Boron Y rod inserted to 9.87"	Full up	1.012	1.000

In Table I the calculated and experimental eigenvalues are compared. The eigenvalues which were calculated were generally 1% too high. Power distributions were calculated to within $\pm 7\%$ for the unrodded cores while the

agreement for rodded cores ranged from ± 7 to $\pm 12\%$ with the power generally overestimated near inserted control rods. The calculations of the unsymmetrically rodded cores proved to be an especially severe test of the rectangular PDQ representation of the hexagonal core geometry.

¹P. G. Klann et al., "Critical Experiments with Low Enriched Double Annular Superheater Fuel Elements in Light Water and their Analysis," Paper 27-5, *Trans. Am. Nuclear Soc.*, 4, 2 (Nov. 1961) p 298.

Critical Studies with ZPR-III for the Enrico Fermi Fast Reactor Core B,* Thomas A. Doyle (APDA) and Allen L. Hess (ANL, Idaho).

The critical experiment on the second core for the Fermi sodium-cooled, fast reactor yielded basic critical details, reactivity effects of various reflectors, and further physics data.

The Fermi Core B will contain plates of fully enriched UO_2 -stainless cermet clad in stainless steel, with an available volume of 600 liters. The radial blanket has a depleted uranium-molybdenum alloy; the axial blanket is depleted UO_2 .

Investigations were made in Argonne's Zero Power Reactor III on a "clean core" mockup. In these studies engineering features were averaged into regional compositions and their reactivity effects separately determined. Plates of uranium and Fe_2O_3 mocked up the UO_2 . The equivalent compositions were 9.38 v/o fully enriched UO_2 , 30.8 v/o Na, 49.4 v/o steel in core, 39.4 v/o depleted U, 2.2 v/o Mo, 28.8 v/o Na, and 18.4 v/o steel in a uniform blanket. These are essentially the Fermi reactor compositions in core and radial blanket respectively. With the fixed core height of 36 in., the core was built out radially and went critical at 667 liters and 505 kg U-235. To achieve criticality with 600 liters it was necessary to replace the depleted U blanket for a thickness of 2 in. around the core radial boundary with nickel reflector of 61.2 v/o Ni, 16.4 v/o Na, and 12.2 v/o steel. The possible need for such a reflector modification was foreseen because of uncertainties in calculations.

Reactivity worths of major reactor materials were measured in this 600-liter, nickel-reflected core. Average worths found were: U-235, 36.5 lh/kg; U-238, -1.3 lh/kg; 0, 21.7 lh/kg; Na, 13.7 lh/kg; and steel, 2.2 lh/kg (relative to void). Local sodium worths were extensively investigated since this was the first large volume, high sodium content core constructed in ZPR-III. Sodium worths were found to decrease from 18.8 lh/kg at the core center to 12 lh/kg at the radial edge of the core in the center plane. Experiments to determine the effect of sodium density on its worth showed little correlation.

*Sponsor: John K. Long

TABLE I. Critical Reflectors for a 600-Liter Core

Reflector	Composition g/cc						Radial width for criticality on 600-liter core
	Ni	O	SS	Fe	Na	Be	
Ni	5.44	--	0.954	--	0.159	--	2.01
NiO	2.69	0.631	0.626	--	(Al 0.710)	--	1.32
Fe_2O_3	--	0.692	1.35	1.74	0.159	--	2.15
Be	--	--	0.954	--	0.159	1.15	1.47
BeO	--	0.428	0.833	1.07	0.080	0.766	1.21

TABLE II. Comparison of Core Parameters in Ni and NiO Reflectors

Measurement	Value in Ni reflector	Value in NiO reflector
Worth of enriched uranium per column at radial core edge	17.1 lh/kg	18.1 lh/kg
Worth of enriched uranium at radial core edge near midplane	20.1 lh/kg	22.7 lh/kg
Ratio of fission rates - core edge to core center - for enriched uranium	0.60	0.57

Some results from the basic physics program were fission ratios at the core center of 0.029 for σ_f U-238/ σ_f U-235, 1.07 for σ_f Pu-239/ σ_f U-235, and 0.23 for σ_f U-234/ σ_f U-235, and a neutron lifetime of 4×10^{-7} sec from Rossi alpha measurement with a β_{eff} of 0.0065.

Experiments with varying reflector thicknesses and materials were also performed in a 90° sector around half the core length. Extrapolating the results linearly to a full reflector, it was found that doubling the above nickel-reflector thickness to 4 in. would be only 1.3 times as effective. Further results are given in Table I.

In the reflector, BeO apparently provides the greatest reactivity gains, but doubt was cast on this result by strong fuel proximity effects observed. The most promising reflector for the Fermi reactor was hence judged to be NiO. Final experiments were made by replacing the Ni with a NiO reflector completely around the core and repeating some comparative measurements. The results are given in Table II.

Experiments and Analysis of Beryllium Oxide Moderated Cores Containing Hydrogen. Part I: Experiments, W. B. Wright and A. J. Goodjohn (GA).

Reactivity and neutron activation measurements have been made on a series of BeO-moderated cores containing hydrogen and are compared with calculated results obtained through multigroup transport theory. These assemblies were built in support of the nuclear design of EBOR¹ and had polyethylene distributed throughout the cores to simulate water flooding. Because of the severe heterogeneities of the unit cells, part of the investigation was devoted to detailed neutron activation measurements in unit cells with varying amounts of fuel and moderator lumping. In the remainder of the experiments, critical cores were built having either unit cell rearrangement or containing discrete slab rods.

Fig. 1. Diagram of the Core Face of the Critical Assembly.

Fig. 2. Manganese Activation Curves Through Unit Cells.

Part I of this series describes the experiments. Part II describes the analytical methods used in unit cell homogenization and reactivity calculations.

All cores were composed of 24 unit cells (6×4.6 in. in cross section and 38 in. long) in a honeycomb assembly constructed of rectangular aluminum tubes. The assemblies were reflected on all sides with 7 in. of BeO. Shutdown of the facility was obtained by two large vertical rods that extended through both the core and reflector. In addition, six BF₃ fuses were incorporated in each core. A diagram of the core face is shown in Fig. 1. Each unit cell contained BeO, polyethylene, fuel (fully enriched uranium in aluminum) and boron stainless steel. The fuel inventory in all cores was 46.1 kg of U-235 and the BeO/U ratio was 110.

Because of the perturbations in the core reactivity created by fuse voids, control rod voids and the slight nonuniformity (in some cases) of the distribution of boron-stain-

TABLE I. Core Descriptions and Reactivity Values

Core	Unit cell	Rod type	Rod location	Rod worth (% β_k/k)	Core ρ_{ex} (% β_k/k)
I	Be-O-U-CH ₂ -B	--	--	--	3.8
II	BeO-CH ₂ -U-B	--	--	--	4.0
III	BeO-U-B-CH ₂	--	--	--	4.0
IV	BeO-B-U-CH ₂	--	--	--	4.6
V	BeO-U-CH ₂ -B	DY ₂ O ₃	One at center	3.5	3.4
VI		B-SS	One at center	3.5	3.4
VII		Bofal	One at center	5.9	2.7
VIII	Same	Boral	Two at center	7.0	2.6
IX		Boral	Two each side	11.1	3.0
X		Boral	Four in reflector	4.7	2.4

less steel shims. these three effects were measured so that the experimental core reactivity could be corrected to a core having neither fuse nor rod voids and a uniform distribution of boron-stainless steel.

Four cores were constructed having varying amounts of lumping of the fuel and moderator in each of the 24 unit cells. Detailed bare manganese activation measurements were made in each case within the unit cell near the core center. The experimental data from two of these measurements are shown in Fig. 2 where the unit cell structure is indicated at the bottom of the graph. The manganese activity was determined from the total gamma activity above 600 kev, and the standard deviation on total foil counts was less than 1% in all cases.

Reactivity measurements were made on ten cores as described in Table I. In each case the reactivity bias due to voids and nonuniform content of boron-stainless steel in some cells have been added to the measured core reactivity. In the cores containing rods, the rod worths were estimated by subcritical measurements and by the amount of distributed boron-stainless steel removed in order to return the core to criticality. Each unit cell was symmetric around its center line, and the order of the components in each half cell is listed.

¹A 10 Mw(t) Experimental Beryllium Oxide Reactor being designed by General Atomic.

Experiments and Analysis of Beryllium Oxide Moderated Cores Containing Hydrogen. Part II: Analysis, W. B. Wright and A. J. Goodjohn (GA).

The manganese activity, as a function of position, was calculated in multigroup transport theory for each unit cell rearrangement reported in the previous paper (Part I). For each configuration, multigroup microscopic cross sections were generated by GAM¹ and GATHER² calculations using the volume weighted number densities of the cell. With this cross section library, DSN³ calculations were made. In all of the DSN calculations, P-1 transfer was included in the epithermal range, and in the thermal range P-1 transfer (within group) was approximated by setting the within-group term equal to $\bar{\mu} \sigma_g$. Manganese absorption cross sections were obtained in each group and used in conjunction with the group fluxes from DSN to obtain the manganese activation rate. All of the experi-

Fig. 1. Comparison of Calculated and Experimental Manganese Activations.

TABLE I. Broad Group Structure for Reactivity Calculations

Groups	Group upper limit (ev)
1	1×10^7 -
2	1.35×10^6 -
3	1.11×10^5 -
4	4.31×10^3 -
5	61.4 -
6	10.68 -
7	2.1 -
8	0.8 -
9	0.5 -
10	0.15 -
11	0.08 -
12	0.02 - 0

mental configurations were calculated in the DSN S_8 approximation using 22 energy groups covering the complete energy range 10^7 ev to 0.005 ev. Comparison of these calculated results with experiment is shown in Fig. 1 for Configuration II where the calculated and experimental points have been normalized to the cell average.

In this configuration the dip in the fuel-poison region (compared with the moderator) was the most pronounced of all the experiments, so that a comparison was made in this case between various DSN approximations. In addition, an investigation was made regarding the effect of changing the number of thermal groups in the DSN problems. Some of the configurations were calculated in the S_8 approximation using 12 epithermal groups and 1, 2, 3, 6 and 10 thermal groups. It was shown that 10 thermal groups were necessary for an adequate prediction of the experimental results.

The total core reactivities of all the cores described in Part I were also calculated. In each case the unit cells were homogenized by DSN calculations. From the results of the DSN problems, the average flux for each group in each region was used to determine fine group self-shielding factors for each of the elements present in the cell. Macroscopic, self-shielded cross sections were then obtained for each of the cores investigated. These cross sections were generated in a 12-group set with the group structure shown in Table I and were utilized in GAZE, 2DXY and/or DSN calculations.

The k_{eff} of all of the various configurations was determined with radial GAZE⁴ problems using group-dependent axial bucklings. For the cores that contained discrete rods, the resulting eigenvalues were representative of the "rods out" condition. In two of the cores, the

TABLE II. Comparisons of the Calculated and Experimental Core Reactivities

Core	% Difference in $\delta k/k$ (Calc-Exp)
I	+1.29
II	+1.42
III	+1.04
IV	+1.10
V	+0.97
VI	+0.97
VII	+0.61
VIII	+0.69
IX	-0.12
X	+0.34

core was represented explicitly in 2DXY problems which included P-O transfer only. The first of these was the unrodded configuration of Core I. The second, Core IX, contained two 12-in. boral slab rods symmetrically located around the center line.

Using the eigenvalues from the GAZE problems and a constant correction factor of 1% in ρ_{ex} from the GAZE-2DXY⁵ comparison, the experimental and calculated experimental and calculated excess reactivity values for all of the cores are compared in Table II.

¹G. D. Joanou and J. S. Dudek, "GAM-I: A Consistent P-1 Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants," GA-1850 (June, 1961).

²A thermal spectrum code developed by General Atomic (report unpublished).

³B. Carlson, et al., "The DSN and TDC Neutron Transport Codes," LAMS-2346 (Oct, 1959).

⁴A one-dimensional diffusion theory code developed by General Atomic (report unpublished).

⁵J. Bengston, et al., "2DXY Two-Dimensional, Cartesian Coordinate SN Transport Calculation," AGN TM-392 (June, 1961).

Measurements of Bucklings and Void Effects in D₂O Moderated Lattices of UO₂ Rod Clusters, W. E. Graves and G. F. O'Neill (DuP-SR).

The bucklings of 25 D₂O moderated lattices of natural UO₂ rod clusters were measured in order to provide a firm experimental foundation for the calculation of (a) cold-clean bucklings of D₂O cooled reactor lattices, and (b) the buckling changes obtained on voiding the fuel assembly housing tubes in such lattices. Knowledge of the cold-clean buckling of a lattice is necessary to predict criticality and core life, and knowledge of the void effect is necessary to predict the stability of a boiling lattice.

The UO₂ rods consisted of 0.500-in. diam sintered pellets stacked in 6063 aluminum tubes 0.020 in. thick and 0.547 in. OD. The oxide density averaged 10.39 g/cc. These rods were used to build up three types of clusters, containing 19, 31, and 48 rods in triangular arrays. The six flat boundaries of the 48-rod cluster alternated between rows of four rods and five rods. The center-to-center rod spacing averaged 0.650 in. in the 19- and 31-rod clusters and 0.656 in. in the 48-rod clusters. Three sizes of 6063 aluminum housing tubes were used. They had outer diameters of 4.0, 5.0, and 6.0 in. and wall thicknesses of 0.054, 0.056, and 0.058 in. respectively.

The fuel assemblies were charged into the Process Development Pile (PDP) as uniform loadings on triangular lattice pitches. The inner diameter of the PDP tank is 4.94 m, and in one loading the full size of the PDP was used. In the other 24 loadings, the radial nuclear boundary of the PDP was reduced by loading lithium-aluminum poison rods into the outer region of the pile. This produced a closer match between radial and vertical bucklings and hence a smaller error in the sum of the two. The radial and vertical bucklings were measured by making flux traverses throughout the pile with gold pins and least-square fitting the results to the appropriate functions. The geometrical bucklings obtained in this way were corrected for the poisoning effects of the gold pins, the poisoning effects of a small amount of aluminum structural material in the lattice, and for slight deviations of the moderator temperature from the reference temperature of 22°C.

TABLE I. Bucklings of Clusters of UO₂ Rods in D₂O (22°C and 99.59 mol % D₂O except where noted)

No. of rods in cluster	Triangular lattice pitch (in.)	OD of housing tube (in.)	D ₂ O or air in housing tube	B_r^2 , m ⁻²	Corrected B_m^2 , m ⁻²
19	8.08	None	--	3.38 ± 0.05	6.02 ± 0.06
19	9.33	None	--	3.51 ± 0.05	5.86 ± 0.06
19	9.33	4	D ₂ O	2.12 ± 0.03	5.01 ± 0.05
19	9.33	4	Air	2.13 ± 0.01	4.48 ± 0.03
19	9.33	5	D ₂ O	2.13 ± 0.01	4.68 ± 0.04
19	9.33	5	Air	1.76 ± 0.01	3.21 ± 0.02
19	12.12	None	--	2.18 ± 0.01	4.11 ± 0.02
19	12.12	4	D ₂ O	2.17 ± 0.02	3.61 ± 0.03
19	12.12	4	Air	2.14 ± 0.02	3.76 ± 0.03
19	14.00	None	--	1.53 ± 0.01	3.01 ± 0.02
31	9.33	None	--	2.08 ± 0.04	5.35 ± 0.05 ^a
31	9.33	5	D ₂ O	2.10 ± 0.03	4.46 ± 0.04
31	9.33	5	Air	2.02 ± 0.04	3.39 ± 0.05
31	11.10	None	--	2.44 ± 0.01	5.25 ± 0.04
31	12.12	None	--	2.17 ± 0.03	4.06 ± 0.04 ^a
31	12.12	5	D ₂ O	2.16 ± 0.02	4.28 ± 0.04
31	12.12	5	Air	2.13 ± 0.03	4.28 ± 0.04
31	14.00	None	--	2.22 ± 0.02	3.83 ± 0.02
31	16.16	None	--	0.88 ± 0.01	2.81 ± 0.02
48	12.12	None	--	2.14 ± 0.02	4.90 ± 0.03
48	12.12	6	D ₂ O	2.16 ± 0.01	4.13 ± 0.03
48	12.12	6	Air	2.07 ± 0.01	3.91 ± 0.02
48	14.00	None	--	1.52 ± 0.01	4.28 ± 0.04
48	14.00	6	D ₂ O	1.52 ± 0.03	3.73 ± 0.04
48	14.00	6	Air	1.48 ± 0.01	4.03 ± 0.03

^aModerator purity = 99.68 mol % D₂O.

The radial bucklings, B_r^2 , and the corrected material bucklings, B_m^2 , are presented in Table I. All results except for the two buckling values noted are for 99.59 mol % D₂O. The assigned errors should be regarded as standard deviations.

Although the survey was planned as a general study covering a wide range of lattice parameters, a few of the loadings could be considered attractive candidates for power reactor lattices. In particular, the 31-rod cluster assembly on a pitch between 9 and 11 in. would satisfy conditions of reasonably low moderator-to-fuel ratios, high buckling, and strongly negative void coefficient of buckling.

Reactivity and Subcriticality Determinations

Critical Mass Experiments with Plutonium Nitrate Solutions, E. D. Clayton, R. C. Lloyd, and W. A. Reardon (GE-HAPO).

A series of criticality experiments was begun in mid-1961 with plutonium solutions in the new Hanford Plutonium Critical Mass Laboratory. This laboratory was built for the purpose of obtaining needed criticality data on plutonium solutions and precipitates of plutonium in support of chemical processing.

Criticality was first achieved on July 21, 1961, with $\text{Pu}(\text{NO}_3)_4$ solution in a 14-in. diam stainless steel sphere reflected with a $\frac{1}{2}$ -in. layer of paraffin. Since then criticality data have been obtained for the bare sphere (from extrapolation only), for the sphere nominally reflected (with $\frac{1}{2}$ -in. of paraffin and 1-in. of paraffin), and with a 4-in. thick concrete reflector. One of the experimental assemblies is shown in Fig. 1. Plutonium solutions with acid molarities in the range of ~ 1.2 to 6, with plutonium concentrations of ~ 33 to 300 g Pu/l have been studied.

Fig. 1. Paraffin Being Positioned on Criticality Vessel for Studies of the Effects of Neutron Reflectors on Criticality.

A variety of criticality data were obtained for the vessel in different states of neutron reflection, with plutonium solutions of various compositions. Data were obtained for evaluating the effect of the stainless steel shell of the vessel on criticality and to evaluate the effect of the $\frac{1}{2}$ - and 1-in. paraffin reflectors. In one case, the vessel was covered with a 30-mil layer of cadmium and then reflected with 1-in. of paraffin. To make the results more useful,

Fig. 2. Critical Mass vs. Pu Density for $\text{Pu}(\text{NO}_3)_4 \cdot \text{H}_2\text{O} + \text{HNO}_3$ Bare Spheres.

Fig. 3. $\ln_e$ Critical Mass vs. Nitrate.

these data have been corrected for reflector savings, and the experiments reduced to equivalent bare critical systems. The data were also corrected to the nearest even integer of nitric acid molarity. The results, corrected to acid molarities of two, four, and six, are presented in Figure 2.

The data in Fig. 2 do not include the results for the concrete reflected sphere; these are now described briefly. The plutonium concentration and acid molarity were varied so as to obtain criticality in the nearly full sphere (constant volume) for acid molarities of 5.6, 2.9, and 1.2.

These data permit an evaluation of the critical mass for a hypothetical plutonium-water mixture in the 14-in. sphere reflected with 4-in. of concrete. The $\ln$ (critical mass) is plotted vs total nitrate in Fig. 3. From these data, the critical mass for the plutonium-water mixture is then estimated to be ~ 633 g Pu (Pu concentration of ~ 28 g Pu/l). These results indicate the 4-in. concrete reflector to be essentially equivalent to an effectively infinite water reflector. In subsequent experiments an additional layer of concrete will be used to increase the reflector thickness to 10-in.

It was possible in a few cases to derive a coefficient relating the nitrate ion concentration to critical volume. The coefficient derived is the change in critical volume per gram of (NO_3^-) change as nitric acid. The conversion to a correction for total (NO_3^-) concentration is simply performed by assuming that the plutonium is present as $\text{Pu}(\text{NO}_3)_4$. For plutonium concentrations in the range of ~ 47 to 98 g/l, the coefficient varied between ~ 0.025 to ~ 0.029 l/g NO_3^- .

Operating Experience in the Hanford Plutonium Critical Mass Facility, R. C. Lloyd, E. D. Clayton, and W. A. Reardon (GE-HAPO).

Since June of 1961, the Critical Mass Group at the Hanford Critical Mass Facility has acquired over six months operating experience while performing approximately 60 critical mass determinations. The purpose of these experiments is to determine the criticality properties of plutonium solutions of various concentrations of plutonium and nitrate, and the effects of various reflectors.

The laboratory consists in essence of a heavily shielded cell, a mixing room, and a control room.¹

The mix tank ($72 \times 24 \times 2\frac{1}{2}$ -in.) in the mixing room was the subject of the first experiment. As a check on the previous nuclear safety evaluation for the tank, an approach-to-critical procedure was employed when the 300 g/l plutonium solution was transferred to the tank. The critical mass of the tank was determined to be > 20 kg of plutonium. The tank is reflected with a 0.030-in. Cd layer between the polyethylene reflector and the 0.25-in. stainless steel tank wall.

The approach-to-critical procedure is employed in each critical mass determination, and the rod worth is simultaneously determined for each experiment. The control rod worth, in terms of solution volume, varies from about 60 ml (at high plutonium concentrations) to about 560 ml (at low plutonium concentrations) as shown in Fig. 1. When the control rod is started into the solution (sphere not full), the flux is observed to rise, then fall off as expected. This is because the first portion of the rod is worth more as a volume displacement (improved geometry) than as a neutron absorber. The effect is estimated to be about 5¢ positive reactivity experimentally, and a perturbation calculation provides an estimate of 4.3¢ $\Delta k/k$.

While the difficulties encountered to date have been relatively minor mechanically, the presence of plutonium makes any equipment change difficult. The transfer and fast fuel addition pumps were originally gear pumps, but failure to self-prime led to their replacement by rotary displacement pumps. These latter pumps have been successful.

Calling of screw-type valves (stainless steel on stainless steel) have been troublesome, and the valves will be replaced with ball-valves.

Air leaking into the lines has given some trouble with airlocking of the pumps. These leaks are covered adequately with neoprene rubber cement.

Accurate measurement of the liquid level has been troublesome. A liquid level manometer of the servo-following type was installed. It was found that the tubes must be flushed with the new solution at the start of each new experiment, because the reading is very sensitive to

Fig. 1. Control "Rod" Worth vs. Plutonium Concentration in 14-Inch Diameter Stainless Steel Sphere.

the specific gravity of the solution. The manometer has, in addition, suffered from mechanical failures in the selsyn drive mechanism.

Contamination control is an ever-present job of detection and housekeeping. One sizeable spill and several minor ones have occurred to date. In each case prompt detection and clean-up have prevented the spread of the contamination. Clean-up is usually accomplished by quickly covering the affected areas with strippable coating and changing the equipment. Strippable coating is used to fix the contamination until more adequate cleaning can be accomplished. Sodium disulfate has been effective in cleaning exposed stainless steel surfaces.

¹W. A. Reardon et al., "Hazards Summary Report for the Hanford Plutonium Critical Mass Laboratory," HW-66266 (Aug 1, 1960).

Experiments Involving Gaps in Critical Assemblies, Arthur G. Cole (UCRL-L).

The series of experiments¹ involving gaps or voids in critical assemblies with simple configurations has been extended to include two-dimensional geometries. Three methods of treating the gaps in neutron diffusion code calculations are presented, together with the results obtained.

The new experiments (Fig. 1) include an assembly stacked on a 30-in.-square base with a 6-in.-square void centered on the vertical axis of symmetry and extending the full 20.6 ± 0.2 in. critical height of the stack. This assembly has 8-mil or alloy (93.2% U-235) foils arranged in planes separated by 1-in. blocks of BeO to give an effective BeO density of 2.86 gms/cm³ and a moderator-to-fuel ratio, BeO: U-235 = 247.

TABLE I

Assembly	Void worth (Reactivity units)		Method 1 (Zoom)		Method 2 (Angle)		Method 3 (Angle)	
	Zoom	Angle	K_{eff}	% effect predicted	K_{eff}	% effect predicted	K_{eff}	% effect predicted
Base problem								
BeO: U-235 = 493	---	---	0.9980	---	0.9976	---	0.9976	---
Base problem								
BeO: U-235 = 247	---	---	1.0005	---	1.0040	---	1.0040	---
Cruciform void								
BeO: U-235 = 493	0.2037	0.2033	1.0695	64.9	1.0426	77.9	1.0012	100.7
Square well void								
BeO: U-235 = 247	0.1531	0.1546	1.0085	93.3	1.0377	74.1	1.0165	91.0

The other two assemblies were mock-ups of a typical Hot Box² experiment. They were stacked on 24×25 in. bases and had "voids" (aluminum hexcell, $\rho = 0.053$ gms/cm³) arranged in the form of a cross. The fuel loading was BeO: U-235 = 494 and the foil thickness and spacing was 4-mils per in. In one case half-density graphite was simulated in the cruciform by half-inch graphite blocks with a density of 1.70 g/cm³.

These experiments have been treated in calculations with the 18-group neutron diffusion codes, 9-Zoom³ and 9-Angle,⁴ by the following three methods:

1) The void volume is weighted according to the neutronic "worth" of its position, and the assembly density reduced by the ratio of the volume of solids to volume of solids plus the weighted void volume. The weighting is done by reference to a curve developed from calculated spatial density variation effects.

2) The void is eliminated by smearing sufficient material from the adjacent regions to make the resulting uniform density of the smeared region one-half that of the remainder of the assembly.

3) The void region is treated as filled with core material at several densities to develop a curve of k_{eff} as a function of relative density of filler material. A line drawn tangent to the curve at $\rho/\rho_0 = 0.5$ is extrapolated to zero relative density to give k_{eff} for the void assembly.

The results of the calculations are presented in Table I. Two previously reported⁵ basic assemblies having the same fuel loadings, foil thicknesses, and spacings as the void experiments are used as base problems. The "void worth" is taken to be the difference between k_{eff} calculated with the dimensions of the void critical assembly but with the void region filled with full-density core materials, and k_{eff} calculated for the base problem. This permits an evaluation of the calculated results in terms of percent of the void effect predicted.

Calculations carried out by method 3 are shown to predict the void effect very well.

Fig. 1. Critical Assembly.

¹E. Goldberg et al., "Critical Experiments Involving Gaps in Simple Configurations," UCRL-6120T, (Aug, 1960).

²R. Finke et al., "Summary Report on a High-Temperature Beryllium-Oxide Critical Experiment," UCRL-6329, (March, 1961).

³S. P. Stone et al., "9-Zoom-A One-Dimensional, Multigroup, Neutron Diffusion Theory Reactor Code for the IBM 709," UCRL-5682, (Aug, 1959).

⁴S. P. Stone, "9-Angle-A Two Dimensional, Multigroup, Neutron Diffusion Theory Reactor Code for the IBM 709 or 7090," UCRL-6076, (Oct, 1960).

⁵F. A. Kloverstrom et al., "Critical Measurements on Near-Homogeneous BeO-Moderated, Oralloid-Fueled Systems," UCRL-5369 Pt. 1, Revised, (July, 1959).

Experiments with Nickel as a Reactor Reflector Material, Arthur G. Cole (UCRL-L).

Experiments demonstrated conclusively that displacement of reflector BeO with a density of 2.86 g/cm³, by nickel metal with a density of 8.90 g/cm³, resulted in an increase in assembly reactivity. Calculations using the 9-Zoom¹ neutron diffusion code and standard cross sections in use at this laboratory, gave very good agreement with the experiments.

The experiments were carried out with the Spade² critical assembly facility and made use of the pulsed-neutron technique³ to give a high degree of sensitivity. The assembly was stacked on a base 24 × 24 in. with 6 × 6 × 1 in. BeO blocks having an effective density of 2.86 g/cm³, and an effective thermal neutron absorption cross section of 7 ± 4 mb. Fuel was present in the form of 8-mil or alloy foils (93.5% U-235) arranged in planes spaced one-inch apart to give a moderator-to-fuel ratio, BeO/U-235 = 247. The measurements determine α , the neutron decay constant, and have an estimated accuracy of ± 3%.

The first measurements were made on the bare core at various stack heights, and the results are shown in Fig. 1. The measurements were then extended to include assemblies with a fueled core capped with a reflector of unfueled BeO and nickel metal in various combinations. The experimental results are presented in Table I together with values calculated with 9-Zoom.

Zoom calculations relating to the four bare systems yielded a functional relation between K_{eff} and α_{exp} . This function, $\alpha(K_{eff})$ was employed in the determination of $\alpha(\text{predicted})$ for the composite systems. The decay constant so determined by reference to the bare systems is a simple function of the prompt neutron lifetime, l^* . The eighth column of Table I lists the values of $\alpha(\text{predicted})$. However, a small adjustment in $\alpha(K_{eff})$ must be made because of the slightly different values of l^* in the reflected systems compared to the bare system. In the ninth column, $\alpha(\text{predicted})$ is adjusted according to the ratio of the values of l^* which were determined with the Zoom code.

The lower decay constant measured with nickel replacing BeO in the reflector indicates the superiority of nickel on an inch-for-inch basis. The values of the decay constant from calculations are in good agreement with the experiments.

Fig. 1. Results of Measurements on Bare Core at Various Stack Heights.

¹S. P. Stone et al., "9-Zoom - A One-Dimensional, Multi-group, Neutron Diffusion Theory Reactor Code for the IBM 709," UCRL-5682 (1959).

²F. A. Kloverstrom et al., "Critical Measurements on Near-Homogeneous BeO-Moderated, Oralloy-Fueled Systems," *Nuclear Science and Eng.* 8, 221 (1960).

³O. C. Kolar and F. A. Kloverstrom, "Pulsed Neutron Measurement of Control Rod Worths," UCRL-6070, (1960).

TABLE I

No.	Assembly composition				K_{eff} calculated	$\alpha(\text{sec}^{-1})$ predicted from bare assemblies	$\alpha(\text{sec}^{-1})$ corrected for l^* ^a	$\alpha(\text{sec}^{-1})$ experiment	$\Delta\alpha$ predicted minus experiment ^a
	Core height (inches)	Unfueled BeO inches lb/ft ²	Nickel inches lb/ft ²						
1	18.5	---	---	---	0.9516	---	---	1628	---
2	19.0	---	---	---	0.9662	---	---	1224	---
3	19.5	---	---	---	0.9800	---	---	802	---
4	20.0	---	---	---	0.9931	---	---	443	---
5	16.5	3.0	44.4	---	0.9729	1020	948	940	+8
6	16.5	2.5	37.0	0.5	0.9797	827	767	724	+43
7	16.5	2.0	29.6	1.0	0.9809	792	741	735	+6
8	16.5	1.5	22.2	1.5	0.9793	835	802	845	-43

^a $\Delta\alpha$ per inch of fueled BeO $\approx 390 \text{ sec}^{-1} \text{ in.}^{-1}$ at $\alpha \approx 800 \text{ sec}^{-1}$.

Use of Gadolinium Septa in Slab Cores, J. J. Kepes and J. A. Mitchell (W-BAPD).

The nuclear characteristics of gadolinium in septa form have been predicted and compared with experiment in slab cores. Three critical assemblies were constructed to provide different spectral environments for the gadolinium. These experimental critical assemblies were built using square Zircaloy-2 fuel boxes with an effective wall thickness of 0.053 in. and an outside dimension of 1.563 in. Each box contained eight fuel plates of 9 w/o enriched uranium in Zircaloy-2, $0.060 \times 1.386 \times 30$ in., clad with 0.015 in. of Zircaloy-2. The average fuel loading was 169 g/l of U-235 while the metal-to-water ratio was 1.2. The gadolinium septa were 4.27 w/o gadolinium in Zircaloy-2, they were 0.090 in. thick, and they extended the length of the core in each experiment. Their optical thickness was 12 at 0.0253 ev. The three gadolinium cores were: (1) a 5×11 box core with a gadolinium sheet between the second and third rows of boxes; (2) a 7×13 core with one row of empty boxes in the center (this row of boxes was lined with gadolinium septa); and (3) a 7×10 array similar to the 7×13 core except that zirconium plates were added to the unfueled row of boxes, which fact resulted in a metal-to-water of 1.12. These cores are depicted in Fig. 1. For each of the three cores the excess reactivities were obtained for a clean critical condition at 80°F.

Natural gadolinium is a non-1/v absorber, due primarily to the presence of large thermal resonances at 0.0268 and 0.0314 ev.¹ Gadolinium has an 0.0253 ev absorption cross section of about 49,000 b, which falls off much faster than 1/v to about 110 b at 0.625 ev. The integrated resonance absorption is estimated to be less than 400 barns. Diffusion theory calculations using these gadolinium constants were done for comparison with experimental results.

Two techniques were used to obtain thermal macroscopic constants in these calculations. The first technique was an iterative method used to arrive at an appropriate gadolinium blackness coefficient² for further use in a diffusion theory calculation. A series of KATE³-WANDA⁴ problems were run to determine the spectral environment of the gadolinium and to predict the thermal flux shape entering the septa; both of these were necessary to calculate the blackness coefficient. The cross section deck for gadolinium used in the Wigner-Wilkins KATE program was generated from data reported in Reference 1. The macroscopic constants for the rest of the critical assembly were obtained from KATE using flux-weighted thermal number densities. The second technique differs in that thermal mixed number density constants were employed,⁵ and a variational technique was used to describe the spectrum around the gadolinium; a discussion of a variational treatment can be found elsewhere in Reference 6. The fast constants in each case were obtained from the slowing down code, MUFT-5.⁷ Gadolinium epithermal blackness coefficients were also generated from the MUFT-5 code.

Fig. 1. Gadolinium Septum Cores.

An explicit core calculation was done in xy geometry using a multigroup diffusion code, PDQ-4,⁸ with constants obtained from the variational technique, (see Table I). The iterative and variational methods were compared with one another using one-dimensional diffusion calculations (WANDA) in the short dimension of each assembly and then normalized to the two dimensional problem (Table I). The importance of delayed neutrons to the measured reactivities was taken into account. The comparisons shown in this table indicate that the use of gadolinium in clean slab geometries has been described by the iterative method to within $-2.0\% \Delta k/k$. The variational treatment appears to give more consistent agreement, particularly for the water gap cores, where the results were within $-0.6\% \Delta k/k$ and $+0.3\% \Delta k/k$.

¹ H. B. Moller et al., "Low-Energy Neutron Resonances in Erbium and Gadolinium," *Nuclear Sci. and Eng.* 8 (1960), pp 183-192.

² C. W. Maynard, "Blackness Theory and Coefficients for Slab Geometry," *Nuclear Sci. and Eng.* 6 (1959), pp 174-186.

³ H. J. Amster and J. B. Callaghan, "KATE-1—A Program for Calculating Wigner-Wilkins and Maxwellian Averaged

TABLE I. Summary of Experimental and Calculated Reactivity Results of Three Gadolinium Assemblies

Core	Effective delayed neutron fraction, β	Physical core size (in.)	Gd septum (w/o)	Measured k_{eff}	Variational treatment k_{eff}	Normalized iterative treatment k_{eff}
5 x 11	0.0079	7.930 x 17.334	4.28	1.0177	1.0032	0.993
7 x 13	0.0077	12.751 x 20.499	4.27	1.0084	1.0111	0.997
7 x 10	0.0077	12.736 x 15.712	4.26	1.0186	1.0124	0.998

Thermal Constants on the Philco-2000," WAPD-TM-232 (1960).

*C. J. Marlowe and M. C. Suggs, "WANDA-5—A One Dimensional Neutron Diffusion Equation Program for the Philco 2000 Computer," WAPD-TM-241 (1960).

*R. J. Breen, "A One-Group Model for Thermal Activation Calculations," *Nuclear Sci. and Eng.* 9 (1961), pp 91-93.

*G. P. Calame and F. D. Federighi, "A Variational Procedure for Determining Spatially Dependent Thermal Spectra," *Nuclear Sci. and Eng.* 10 (1961), pp 190-201.

*H. Bohl, Jr. and A. P. Hemphill, "MUFT-5—A Fast Neutron Spectrum Program for the Philco 2000," WAPD-TM-218 (1961).

*W. R. Cadwell et al., "PDQ-4—A Program for the Solution of the Neutron Diffusion Equations in Two Dimensions on the Philco-2000," WAPD-TM-230 (1961).

Reactor Parameter Measurements

k_{∞} Measurements of Natural UO_2 , Heavy Water Moderated Lattices in the Pawling Lattice Test Rig, R. D. Schamberger, G. G. Foster, and W. L. Brooks (UNC, NY).

The Pawling Lattice Test Rig (PLATR)¹ is a heavy water moderated, zero power reactor designed to simulate in its central region the neutron environment found in an infinite system of the lattice under examination. A 44-in. long section of seven experimental lattice cells, occupying the center of the core, is surrounded by a spectrum matching region which is, in turn, surrounded by an enriched uranium driver region. The principles of operation are similar to the Physical Constants Testing Reactor (PCTR) at Hanford,² which is its graphite moderated counterpart and the original device of this kind.

With the experimental lattice in the proper neutron environment and poisoned so that $k_{\infty} = 1$, homogeneous theory indicates that replacing a section of the cell with void will not alter the reactivity of the system. The effect of cell inhomogeneities on this conclusion will be discussed.

The conditions which must be satisfied to establish the proper neutron environment are that the flux shall be flat i.e., constant parallel to the cell axis, over the test region) and that the spectrum shall be the same at corresponding points in the central cell and adjacent cells. The experimental criteria which have been employed in PLATR to insure that these conditions have been satisfied are in terms of the activation of bare and cadmium covered gold foils.

After the establishment of the proper environment and the determination of the amount of thermal neutron absorber required to reduce k_{∞} to 1, the intracell flux distribution is determined to permit the determination of values for the thermal utilization, f , and the infinite multiplication factor, k_{∞} , for the unpoisoned lattice.

An extensive series of measurements has been made with clusters of $\frac{1}{2}$ -in. diam. natural UO_2 rods clad in aluminum. A major portion of the series dealt with a 37-rod cluster on an 11.1-in. lattice pitch, the NDA-S&L design.³ Among the parameters studied were: lattice pitch, rod-rod pitch, cluster size, insulation gap between housing tubes, and clad material. The coolant void coefficient was also determined. A corresponding series of lattices has been evaluated using the ROCLAND-A code.⁴ The agreement between the calculations and measurements is satisfactory; however, the calculations yielded are generally lower values than the measurements by 0.5-1.0% in k_{∞} .

Since August 1961 the experimental program has been concerned with tubular natural UO_2 elements which were fabricated at the Savannah River Laboratories of the USAEC. Lattices of single tube elements and coaxial tube

clusters have been studied. The parameters reviewed in this study have been the lattice pitch, the cluster size and the coolant material. Particular emphasis has been placed on the determination of partial and total coolant void coefficients.

The effect of small changes in a configuration, changes which make negligible alterations in the neutron environment, can be evaluated by changing only the central part of the experimental lattice. In the case of a liquid (such as the coolant), such a change can be made while the system is in operation, thus greatly increasing the sensitivity of measurement. Comparison of the results of coolant void coefficients obtained in this fashion with the coefficient obtained as the difference between independent k_{∞} measurements with and without coolants showed good agreement for changes in k_{∞} of a few percent or less.

The coolant void coefficient is, in general, positive with a value of 0.025 in k_{∞} for the standard 37-rod cluster on an 11.1-in. lattice pitch.

With a representative tubular cluster having a fuel/coolant volume ratio of 2.20, the coolant void coefficient increased from 0.007 to 0.014 as the lattice pitch was increased from 7.5 to 13.22 in. The major portion of the change was produced by the coefficient in the outer coolant channel for which the void coefficient was a negative 0.0007 at a lattice pitch of 7.5-in. and increased to +0.006 at 13.22-in. lattice pitch.

As expected, in a clustered rod lattice the value of k_{∞} decreased as the spacing between rods was increased. The magnitude of the effect, about 2% in k_{∞} on increasing the rod center-to-center spacing from 0.580 to 0.720 in., was larger than anticipated on the basis of the ROCLAND-A calculations.

The value of k_{∞} increased with increasing lattice pitch. Similar shapes have been observed for the rod cluster and tubular elements. The moderator/fuel ratio does not, however, appear to be a convenient parameter to employ in obtaining a correlation.

¹ P. Anthony et al., "The Pawling Lattice Test Rig Design Report," NDA 2131-25 (Apr. 1961).

² D. J. Donahue et al., "Determination of k_{∞} from Critical Experiments with the PCTR," *Nuclear Sci. and Eng.*, 4, (1958), p. 297.

³ "Design Study—Heavy Water-Moderated Power Reactor Plants," SL-1565 (Jan 28, 1959).

⁴ H. Soodak and R. Sullivan, "A Method for Calculating the Reactivity of D_2O -Moderated Natural Uranium Lattices with Clustered Rod Fuel Elements," NDA 2131-38 (June 1961).

The Determination of k_{∞} for Three Percent U-235 Enriched UO_3 and Slightly Enriched Uranium Metal from Monte Carlo Calculations, C. R. Richey (GE-HAPO).

Two quantities of special interest for nuclear safety have been determined by Monte Carlo Calculations: (1) the value of k_{∞} for unmoderated, 3.04 w/o enriched UO_3 and whether this value is greater or less than unity, and (2) the limiting enrichment for uranium metal, i.e., the enrichment at which k_{∞} is just unity for unmoderated metal. Multiplication factors for 3.04 w/o enriched UO_3 systems having hydrogen-to-uranium atom ratios in the range 3.57 to 43.87 have previously been measured in the Hanford PCTR.¹ The k_{∞} at the lowest H/U for which measurements were made (3.57) was found to be in excess of unity (1.310 ± 0.023). For nuclear safety considerations, additional data was needed in the H/U range from zero to 3.57. Due to the increased amount of material required to measure k_{∞} at very low H/U and the difficulties in obtaining and sampling these mixtures, Monte Carlo methods were used to determine k_{∞} in this range.

The calculations were carried out utilizing the HISMIC code (a Monte Carlo code for tracing neutron lifetimes from birth to capture in an infinite homogeneous system) on an IBM-7090 digital computer. Point cross section values were stored on a data tape at those energy intervals necessary to accurately describe the cross sections as a function of energy by a linear interpolation of the stored values. For neutron energies in excess of 0.1 Mev, the cosine of the elastic scattering was obtained from a stored table of inverted partial integrals of the differential elastic scattering cross section.² Inelastic scattering events were assumed to be isotropic in the center-of-mass system, and the resultant neutron energy was determined from the statistical model. For neutron energies less than 0.1 Mev, the scattering was assumed to be isotropic in the center-of-mass system. In the experiments, "dry" UO_3 powder (approximately 1 w/o H_2O) was mixed with polyethylene to obtain the desired H-to-U atomic ratios. For the Monte Carlo calculations, the scattering model for the bound hydrogen in the polyethylene molecule was considered to be identical to that for free hydrogen.

Figure 1 illustrates the infinite multiplication factor as a function of hydrogen-to-uranium atom ratio for the UO_3 systems as obtained from Monte Carlo calculations. For comparison, the measured values as obtained from the PCTR experiments are also given.¹ It is noted from Fig. 1 that the value of k_{∞} decreases quite rapidly at low H-to-U ratios; k_{∞} for dry UO_3 salt is $k_{\infty} = 0.584 \pm 0.019$. To determine the effect of moderation by oxygen for dry UO_3 , a calculation was made for 3.04% enriched uranium metal at a density of 18.9 g/cc. Calculations were also made for enrichments of 5 and 6%. The net effect of oxygen in the dry UO_3 was found to be ~ 136 milli k. Interpolating the enriched metal calculations (see Fig. 2) another quantity of special interest in nuclear safety is determined—the limiting enrichment of uranium metal. The value of 5.9 w/o U-235 found for the limiting enrichment of uranium metal is in agreement with the conclusion drawn by H. C. Paxton, obtained by the extrapolation column data, that the limiting enrichment is between 5 and 6%.

¹V. I. Neeley, "k_∞ of Three Weight Percent U²³⁵ Enriched UO_3 and $\text{UO}_2(\text{NO}_3)_2$ Hydrogenous Systems," HW-66882.

²J. R. Lamarsh et al., "Selected Nuclear Data for Monte Carlo Calculations," NP-8216 (Oct 1958).

³H. C. Paxton, "Critical Masses of Fissionable Metal as Basic Nuclear Safety Data," LA-1958 (Jan 1955).

Fig. 1.

Fig. 2.

The SL-1 Accident

SL-1 Recovery Operation, C. L. Storrs (GE-NMPO, Idaho) and C. W. Bills (AEC-IDO).

On January 3, 1961, the SL-1 reactor at the National Reactor Testing Station experienced an excursion that killed the three operators on duty at the time. Immediate efforts were devoted to the removal of the victims and to assuring the safety of the reactor. In May 1961, the General Electric Company was called upon to undertake recovery operations with the objectives of gathering evidence as to the causes of the accident and of cleaning up the area and restoring it to useful service.

The main problem confronting the recovery operation was, of course, the high radiation level. Gamma doses were over 100 r/hr, with a ratio of beta to gamma dose up to 15:1, so that the beta exposure was operationally limiting. The radioactive material was widely distributed throughout the reactor operating room and the fan room above it. The operating room, 21 feet above the ground, was congested with equipment and shield blocks, while the fan room was accessible only by cutting through the wall. Additional complications resulted from the necessity of avoiding any possibility of renewed criticality of the reactor, as for example by flooding it with water, and the requirement to examine all material for its possible value as evidence.

The over-all recovery plan started with steps to lower radiation levels, concurrent with the gathering of evidence from the reactor building. Following this, much of the equipment and structure above the reactor pressure vessel was removed, permitting the final step of removing the pressure vessel and transporting it to the Hot Shop in the north end of the National Reactor Testing Station, where specialized equipment was available to dismantle and examine the core.

Some use was made of remote handling techniques, principally with an Austin-Western crane fitted with a horizontal boom 26 feet above the ground and with shielding for the operators. This equipment was used for photographic and radiation surveys, placing shielding, and removing debris with an electromagnet.

Main reliance in the recovery operation, however, was placed upon manual entries. Personnel were given normal occupational exposures, the usual limit being the quarterly permissible dose to the skin. Approximately 475 people were used, including volunteers from the U. S. Army and Atomic Energy Commission in addition to our own personnel. Extensive photographic coverage was maintained during the cleanup, throughout the building and above and below the core. The removal of the vessel was greatly expedited because the pipes connected to the vessel had been sheared during the accident. Subsequent to the removal of the pressure vessel and reactor, the reactor building is being completely dismantled and the remainder of the area decontaminated and restored to usefulness.

Experience with the SL-1 recovery has demonstrated that a highly contaminated area can be cleaned up, using mostly ordinary manual techniques for the fastest and most economical results. General purpose apparatus which is of value in such an operation includes devices for remote television, photography, and radiation surveys, and powerful vacuum and steam cleaning equipment. Efficient health physics and radioactive waste disposal provisions are also essential.

Critical Experiments

Precise Criticality Determinations in the Solid Homogeneous Assembly, F. Feiner, S. Weinstein, W. C. Oaks, K. V. Cooper (KAPL).

Properly designed critical-assembly experiments can often be used to determine the adequacy of group constants used in reactor calculations. But because of the assumptions that need to be made about fuel self-shielding and the homogenization of core-support structure and channels, the usual heterogeneous core comparisons involve several percent uncertainties. Homogeneous-solution experiments have been used to determine critical sizes very accurately but can be used only for low-metal-fraction compositions because of the limited solubility of the metallic salts in water. To accommodate large metal fractions, but still retain the advantages of homogeneity, the Solid Homogeneous Assembly (SHA) is designed with the metallic elements and fuel as very fine powders dispersed in paraffin.

Figure 1 shows a picture of the reactor. The core, comprised of a stack of fuel and reflector blocks, rests on a low-density base to reduce floor reflection. The two halves of the reactor are separately movable, and are held apart by springs. The tables are coupled to the drive mechanisms by magnetic latches which release on scram allowing the springs to separate the halves. The springs provide an acceleration of about 1 m/sec² which produces fast safety shutdown of more than 5β in 200 msec.

Shim reactivity control is provided by the reflectors at the two ends of the reactor as well as by the table-closure motion. The entire reflector assemblies may be readily removed to perform unreflected experiments.

The fuel elements are primarily 4-in. $\times$ 4-in. $\times$ 2-in. compacts of fine UO₂ and ZrO₂ powder uniformly distributed in paraffin. On the basis of statistical sampling, the mean uranium and paraffin densities of a core are expected to be known to $\pm 1\%$ with 90% confidence. There is 80% confidence that 90% of all the 5-cm³ volumes of the fuel blocks have uranium and paraffin concentrations within $\pm 2\%$ of the mean.

The composition of the blocks currently available is as follows:

	Number densities, atoms/barn-cm					
	U-235	Zr	O	C	H	Al
UO ₂ -ZrO ₂ -CH ₃ fuel blocks	$.116 \times 10^{-24}$	.01565	.03156	.01235	.02568	$.248 \times 10^{-24}$
ZrO ₂ -CH ₃ reflector blocks		.01584	.03167	.01263	.02575	$.247 \times 10^{-24}$
Polyethylene				.03930	.07861	

A series of reproducibility checks performed with the assembly has demonstrated that rearranging or replacing major segments of the core leads to reactivity changes of less than 10¢.

Experiments on an unreflected critical configuration, 50 $\times$ 50 $\times$ 45 cm³, demonstrate that the reactivity effect of reflections from the room and the reactor structure is about 65¢.

A series of criticality experiments has been performed with rectangular cores with 4-in. reflectors. The effective multiplication of five of the configurations are listed in Table I along with the multiplication that has been calculated using few-group neutron-diffusion programs. The group constants were obtained from MUFT-V¹ and KATE² using "as built" number densities.

KLAGE³ (a few-group flux-synthesis program) was used for the calculation of configurations I and II because of the three-dimensional nature of the problems. The errors introduced by these calculations are being studied and a variational procedure using a convolution of slowing-down kernels is being developed to perform three-dimensional calculations more accurately.

It can be seen from Table I that discrepancies of about 2.5% exist between the calculation and the experiments. Possible sources of the discrepancies are the choice of thermal-scattering kernel, slowing-down model and poor high-energy non-elastic cross sections. Calculations of these effects are being performed. It is hoped that these calculations plus experimental determination of energy-flux distributions and position dependent reactivity coefficients will serve to isolate the effects responsible for the miscalculation.

TABLE I

Configuration	Effective Multiplication	
	Experiment	Calculations
I 14.05-in. $\times$ 16.00-in. $\times$ 16.00-in. core; 4.00-in. polyethylene reflector on all sides	1.003	1.0229 KLAG ³
II 12.575-in. $\times$ 16.00-in. $\times$ 16.00-in. core; 4.00-in. zirconia-paraffin reflector on all sides	1.006	1.0336 KLAG
III 14.05-in. $\times$ 16.00-in. $\times$ 20.00-in. core; 4.00-in. polyethylene reflector on four sides	1.000	1.0278 KARE ⁴
IV 18.10-in. $\times$ 16.00-in. $\times$ 20.00-in. core; 4.00-in. polyethylene reflector on two sides	1.008	1.0259 KARE
V 18.10-in. $\times$ 20.00-in. $\times$ 20.00-in. bare core	1.002	1.0228 Hand Calculation

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2. AMSTER, H. J., and CALLAGHAN, J. B., "KATE-I A Program for Calculating Wigner-Wilkins and Maxwellian Averaged Thermal Constants on the Philco-2000" WAPD-TM-232.
3. WACHSPRESS, E. L., "Digital Computations of Space-Time Variation of Neutron Fluxes," *Nuc. Sci. & Eng.*, 12, 381, (1962).
4. ARCHIBALD, J. A., Jr., "KARE Input" KAPL-M-JA-6, (June 1961).

Fig. 1.

Reactor Physics II

On the Physics of Semi-homogeneous Graphite U-235 Reactors, A. J. Goodjohn, G. D. Joanou and N. F. Wikner (GA).

The bare and reflected graphite/U-235 assemblies described by H. L. Reynolds¹ have been analyzed. Careful attention has been given to both the selection of input cross section data and to the adequacy of various approximations to Boltzmann's equation in solving for the basic reactivity.

For graphite, existing experimental information on the high-energy scattering cross sections has been analyzed and compiled in suitable calculational form. The data have been checked against Fermi age measurements in graphite and in hydrocarbons;² excellent correlations have been obtained. In addition, thermal-neutron interactions with bound carbon atoms have been the subject of an intensive investigation³ leading to excellent correlations with thermal-neutron-spectra measurements in poisoned graphite.⁴

There are not sufficient experimental data on all of the partial cross sections of U-235. Experimental measurements have been made of the total and fission cross sections for most of the pertinent energy range. Consequently, it is possible to deduce a set of energy-dependent cross sections for reactor calculations. The procedures we have used are the following: For energies below 5 eV, all of the partial cross sections are well determined from the experimental work of Shore and Sailor⁵ and from the multilevel analysis of Vogt.⁶ Vogt's work, in particular, leads to the same 2200 m/sec values of ν , α , $\sigma_{n,\gamma}$, $\sigma_{n,f}$ and $\sigma_{n,n}$ as those obtained by a least-squares fit to all of the relevant 2200 m/sec measurements.⁷ For the energy range 5 to 61 eV, we have arbitrarily used a single-level Breit-Wigner resonance formulation to fit the total, capture, and fission cross sections at and near the resonance energies, and we account for fission interference in the wings and tails of the resonances by simple reverting to the measured data whenever the calculated fission cross section falls below the experimental measurements. Between 61 and 10⁷ eV, the capture cross sections were deduced from the measured fission cross sections and the

α measurements of Diven and Hopkins.⁸ We assumed that $\alpha = 0.5$ between 61 eV and 10,000 eV. Above 1.0 MeV, we assumed that $\sigma_{n,\gamma}$ varies as 1/E.

The analysis of the graphite/U-235 assemblies proceeded by computing slowing-down and thermal spectra and associated few or multigroup cross sections using the data for graphite and U-235 discussed above. These group parameters were then used with appropriate one- and two-dimensional diffusion and transport approximations to the Boltzmann equation to calculate k_{eff} . The difference between (a) thermal spectra computed with a free-gas and a bound scattering kernel, (b) diffusion theory and DSN transport theory, (c) isotropic scattering and linearly anisotropic scattering options in DSN and (d) the S-4 and S-8 discrete angle approximations in DSN were evaluated. Some of our results for bare assemblies are given in Table I.

Binding effects are significant for the more thermal systems as indicated in Column 4. Rather good results are obtained with diffusion theory, particularly if the effects of chemical binding are treated in a reasonable way. (For these reactors, a 2% error in k_{eff} yields a 25% error in U-235 loading.) Treating the anisotropy of the scattering in the laboratory system in the P - 1 approximation makes a $\frac{1}{2}$ % change over the P - 0 approximation. The treatment of the anisotropy of the flux in higher order than the DS₄ approximation is unimportant. Similar results were obtained for the reflected assemblies.

From these studies we can see that the cross sections for graphite and U-235, as presented in this paper, are quite adequate for an accurate description of the physics of graphite/U-235 reactors in this range of absorption. The effect that must be treated most thoroughly is that of chemical binding on thermal-neutron spectra below 2 eV. Transport and anisotropic effects are of secondary importance, although they must be treated accurately in definitive calculations.

TABLE I. Values of K_{eff} for Bare Graphite Critical Assemblies

C-U-235 Atom Ratio	k_{eff}							
	Diffusion Theory			Transp. Theory (DS ₄)			Transp. Theory (DS ₄)	
				Crystal			Crystal	
	Free Gas	Crystal	% Diff.	P - 0	P - 1	%	P - 1	% Diffusion DS ₄ - DS ₄
297	1.00087	1.00015	.072	.99538	1.0016	.62	1.00238	.078
600	.99781	.99393	.39	.99323	.99991	.67	1.0004	.046
1200	1.00612	.99567	1.05	.99586	1.00178	.61	1.0023	.051
2340	1.01569	.99605	1.97	.99681	1.00203	.52	1.0027	.062

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2. JOANOU, G. D., et al., "Moments Calculations of the Fermi Age in Moderators and Moderator-Metal Mixtures," *Nuclear Sci. and Eng.*, 13, pp. 171-189 (1962).
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Critical Experiments and Nuclear Parameter Measurements

Experiments and Analysis of Water-Reflected, Undermoderated Zirconium-Hydride Critical Assemblies—Part I: Experiments, *L. I. Moss (AI)*

These experiments were done to evaluate the consequences of water immersion of reactor cores of the SNAP 2/10A type. Of particular interest was the measurement of reactivities associated with various configurations.

Figure 1 is a diagram of the core region of the assembly. The fuel material was an alloy of zirconium/10 wt% of 93.2%-enriched uranium, hydrided to a concentration of 6.5×10^{22} atoms of hydrogen/cm³. The fuel rods were 12.225 in. in length and 1.212 in. in diameter, clad in Hastelloy N. They were arranged in a triangular array on 1.260 in. centers. The cross-section of the fully loaded (37-rod) core was a hexagon 7.80 in. across the flats and 8.81 in. across the corners. The core vessel consisted of a 0.032 in. cylindrical shell of 316 stainless steel. Inserts of beryllium filled the volume between the hexagonal fuel region and the cylindrical shell. The water reflector was physically separated into three regions—a base tank, a top tank, and a bottom control cap.

Critical loadings were measured with several combinations of lucite rods in vacant lattice locations, beryllium inserts, internal water, neutron-poison annuli at the core-reflector interface and ammonium pentaborate in the water supply. A few of these configurations and fuel loadings are listed in Table I.

The reactivity worth of the top tank water as a function of height, and the incremental worth of substitution of fuel rods for lucite rods was measured by the pulsed-neutron

Fig. 1. Core Region of the Test Assembly.

TABLE I. Critical Loadings and Extrapolated Excess Reactivities of Configurations with an Effectively Infinite Water Reflector (6 in.)

Assembly Configuration					Critical Loading, Number of Fuel Rods	Extrapolated Excess Reactivities, Fully Loaded (37 Rod) Core dollars
Lucite Rods in Vacant Lattice Locations	Beryllium Inserts	Internal Water	Binal Annulus, 10.2 wt% B, 0.75 in. thick	Ammonium Pentaborate in Water, 3.33×10^{20} atoms B/cm ³		
		X			27.8 ± 0.1	
X ^a	X	X			24.5 ± 0.2	13.7 ± 1.3
X ^b	X	X	X		34.4 ± 0.1	1.6 ± 0.5
	X	X		X	37.0 ± 0.1	0.0 ± 0.1
X ^c	X				30.95 ± 0.05	6.4 ± 0.6

^aIn locations 16, 23, 24, 29, 30, 31, 32, 33, 34, 35, 36, and 37 of Figure 1.

^bIn locations 35, 36, and 37 of Figure 1.

^cIn locations 32, 33, 34, 35, 36, and 37 of Figure 1.

method¹ as the loading was continued past the critical point. From these measurements an extrapolation for the excess reactivities of fully loaded, fully water-reflected assemblies was obtained, as listed in Table I.

The ratio of the effective delayed neutron fraction (β_{eff}) to the effective prompt-neutron generation time (l^*) was measured for several unpoisoned configurations, employing both pulsed neutron and noise analysis^{2,3,4} methods. The two methods were in satisfactory agreement, giving a best value of $4.76 \times 10^2 \text{ sec}^{-1}$. The corresponding value for the beryllium-reflected reactor has been measured⁵ as $1.38 \times 10^3 \text{ sec}^{-1}$. The much greater l^* for the water-reflected assembly is attributable to reflector delayed neutrons.

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Experiments and Analysis of Water-Reflected, Undermoderated Zirconium-Hydride Critical Assemblies—Part II: Analysis, B. W. Colston (AI)

The experiments described in Part I (previous paper) were analyzed using existing nuclear codes and cross-section libraries. Effects of the geometrical representation were considered, including the difference between one- and two-dimensional models and the homogenization of the hexagonal core structure to a cylindrical geometry. Calculations were accomplished with and without representations for an anisotropic neutron flux. In addition, the effect of anisotropic scattering with hydrogen was considered. Thermalization calculations, including the effects of spatially dependent spectra, were accomplished. Cell effects in these reactors are not significant and were not considered.

Transport-theory effects were evaluated using the one-dimensional codes DSN,¹ DTK² and the two-dimensional code TDC.¹ One-dimensional diffusion-theory calculations were accomplished with FAIM.³ The slowing-down and thermalization calculations, and cross-section group reductions were accomplished using ULCER⁴ and QUICK-IE.⁵ The one-dimensional and two-dimensional criticality calculations were accomplished using sixteen and four groups respectively. The spectrum calculations (ULCER and QUICKIE), were done with twenty-three groups, fourteen fast and nine thermal. In general, the cross sections used were from the sixteen-group set generated by Hansen and Roach.⁶ Changes were made to the U-235 cross sections of Reference 6 to give a fission resonance integral of 271 barns.⁷ Thermal cross sections were changed to include averaging over the calculated thermal spectra. Thermalization cross sections used were those available from the ULCER library, and include the representations for zirconium hydride and water reported in Reference 8.

The one-dimensional calculations were done using the buckling iteration techniques described in Reference 9.

The two-dimensional calculations resulted in reactivity increases of as much as 1.5% above the one-dimensional cases. In addition to this geometrical effect, the problem of representing the hexagonal core in a cylindrical geometry was shown to introduce as much as a 2.0% difference in reactivity depending upon the choice of homogenization techniques, i.e., cell or rubber-band model. (The cell model resulted in higher reactivities.) The final method was a combination of the above techniques and reproduced the surface-to-volume ratio at the core vessel and the correct void fractions in the central core region.

The hydrogen content of the core, the relatively large surface-to-volume ratio of these assemblies, and the use of a poison sleeve indicated the importance of an anisotropic flux representation, particularly at the edge of the cores. In addition, it appeared that the anisotropy of the neutron scattering with hydrogen would also be important. These effects were investigated using the S_N codes. The reactivity effect of the anisotropy of the flux varied from 3.0% to 6.0% for the various assemblies. The largest difference was in the poison-sleeve case, for which consideration of the anisotropy of the flux led to an increase in reactivity of 6.0%. The remaining cases all resulted in increases in reactivity of about 3.0%. The use of anisotropic scattering cross sections for hydrogen introduced additional 1.0% to 4.5% differences in reactivities, although opposite in sign to those of the anisotropic flux effects. Table II presents some comparisons of the differences between transport- and diffusion-theory calculations for reactivity differences between the configurations presented in Table I. These results were obtained with one-dimensional, sixteen-group calculations. A β of 0.008 was used to convert to units of dollars.

TABLE II. Experimental and Calculated Reactivity Worths of Core and Reflector Perturbations

	Diffusion Calculation	Transport Calculations (S_N)		Experiment
		Isotropic Scattering	Anisotropic Scattering	
Water in Core	\$ 4.93	\$ 4.96	\$ 5.66	\$7.3 ± 1.9
Borated Water in Core and Reflector	\$18.0	\$14.8	\$13.98	\$13.7 ± 1.4
B ₂ C Sleeve	\$20.60	\$15.0	\$12.80	\$12.1 ± 1.9

Because of the differences in the chemical binding characteristics of hydrogen in zirconium hydride and water, the spatial dependence of the spectrum was considered. This was accomplished using ULCER, a one-dimensional diffusion-theory calculation which allows neutron upscatter. It was found that at the core edge the spectrum was significantly distorted from the characteristic hydride spectrum established in the core interior. These results were incorporated in the two-dimensional calculations by defining separate core regions for the center and edge of the core and using the appropriate spectrum weighting indicated by ULCER.

The excess reactivities of the four assemblies in Table I were calculated to within 2.0% when proper treatments of the geometrical, transport and spectral problems were included.

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Experimental Determination of Anisotropy in D_2O /Uranium Lattices Containing Voids, W. B. Rogers, Jr. (DuPont-SR)

Anisotropy in the neutron migration area of a reactor lattice can be determined by comparing bucklings measured in different lattice configurations.¹ As part of an exchange program between A. B. Atomenergi, Studsvik, Sweden and the Savannah River Laboratory, measurements were made in each facility in D_2O and air-filled fuel assemblies in heavy water. The assemblies consisted of four coaxial uranium metal tubes in aluminum pressure tubes (Table I). The tubes were first sent to Sweden for buckling measurements in the 1-meter-diameter exponential ZEBRA² and then returned to SRL for measurement in the 1.52-meter-diameter exponential SE.³ Complete data on the ZEBRA measurements have been reported by A. B. Atomenergi.⁴

The anisotropy is given by: $\frac{M_z^2 - M_r^2}{M_z^2} =$

$$\frac{[B_r^2(\text{air}) - B_r^2(D_2O) + \kappa_z^2(D_2O) - \kappa_z^2(\text{air})]_1 - [B_r^2(\text{air}) - B_r^2(D_2O) + \kappa_z^2(D_2O) - \kappa_z^2(\text{air})]_2}{[B_r^2(\text{air})]_1 - [B_r^2(\text{air})]_2}$$

where B_r^2 = radial buckling; κ_z^2 = vertical buckling; M_r^2 and M_z^2 = radial and vertical components of M^2 , respectively; (air) and (D_2O) = coolant channel material; and subscripts 1 and 2 refer to the facility in which the measurement is made.

Figure 1 shows the results of anisotropy determinations with the air-filled lattices. Although the points show

TABLE I. Dimensions of Fuel Assembly

Tube	Tube Dimensions, Inches	
	OD	ID
Aluminum pressure tube	3.840	3.570
Aluminum insulating tube	3.450	3.390
Uranium tube 1	3.080	2.830
Uranium tube 2	2.410	2.150
Uranium tube 3	1.730	1.480
Aluminum cladding	1.120	1.060
Uranium tube 4	1.060	0.800

Fig. 1. Anisotropy in Migration Area Air-filled Assemblies.

Fig. 2. Material Buckling D_2O -Filled Assemblies Mod. Purity - 99.62 mol % D_2O

considerable spread, the trend is as expected with lower anisotropy at the lower void fractions.

Figure 2 compares the results of the two series of material-buckling determinations for the nearly isotropic D_2O -filled lattices. The SE and ZEBRA points agree within the experimental error although the SE points are generally lower than the ZEBRA points.

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Bucklings and Disadvantage Factors of Several Water-Moderated, 1.25%-Enriched Uranium Metal Slab Lattices, *Henry Windsor (BNL)*

Measurements of bucklings, reflector savings, indium-cadmium ratios and thermal disadvantage factors were performed on water-moderated exponential assemblies made up of bare uranium metal slabs of 1.25% enrichment. The slabs are $0.122 \times 2 \times 48$ inches, and for the macroscopic measurements were arranged in continuous sheets $\frac{1}{8}$ inch thick by 48 inches high of selected widths spaced in a row to give the desired volume ratios. The water-to-uranium ratios investigated were nominally 4:1, 3:1, 2:1, 1.5:1, and 1:1. The disadvantage factors were measured in smaller lattices made up of varnished 16-inch long, but otherwise similar, fuel elements. The values of the measured bucklings, reflector savings, indium-cadmium ratios, and disadvantage factors are given in Table I.

Buckling and reflector-savings measurements were performed by both the standard flux-shape technique and the "variable loading" technique in which a series of axial relaxation lengths were measured as a function of lattice size and fitted to the equation

$$B_m^2 = \frac{\pi^2}{(x + 2\lambda)^2} + \frac{\pi^2}{(y + 2\lambda)^2} - \frac{1}{L^2} \quad (1)$$

where x and y are the known lattice widths, L is the measured axial relaxation length and B_m^2 and λ are determined by a least-square fit.¹

The flux shape was measured in lattices consisting of enough fuel to give a 40-cm axial relaxation length except in the 1:1 case where only enough fuel was available to obtain an 18-cm relaxation length. A series of approximately 20 axial relaxation lengths was measured in each lattice as its size was decreased until a 10-cm relaxation length was obtained. The 40-cm relaxation length corresponded to lattice dimensions in the range of $18 \times 20 \times 48$ inches, while the 10-cm relaxation length is in the neighborhood of a $10 \times 12 \times 48$ inch assembly.

The slab-lattice disadvantage factors were measured in a miniature lattice. This is a small rectangular exponential assembly measuring 16 inches high by 12 inches wide by 12 inches long contained in an aluminum tank surrounded on all sides, except the top, by a $\frac{1}{16}$ -inch of cadmium sheet and 3 inches of paraffin. The whole assembly was exposed in the animal tunnel facility of the Brookhaven Graphite Reactor which supplied a flux of 10^9 neutrons/sec-cm².

The disadvantage factors were determined from intracellular flux distributions measured with 0.007-in.-thick, $\frac{1}{8}$ -inch in diameter, 93% Al/7% Dy foils. Seven or five foils, depending on the volume ratio, were supported in the water by 0.015-inch-thick Lucite foil holders. One fuel element contained a $\frac{3}{8}$ -inch-diameter $\frac{1}{8}$ -inch-thick uranium disk insert. This disk was split in half, and

TABLE I. Bucklings and Disadvantage Factors of Several Water-Moderated, 1.25%-Enriched Uranium Metal Slab Lattices

Bucklings are given in units of 10^{-4} cm ⁻² ; reflector savings (λ) in cm.									
Volume Ratio V_W/V_U	Direction	Variable Loading				Flux Shape ¹			
		B_m^2	δB_m^2	λ	$\delta \lambda$	B_m^2	δB_m^2	λ	$\delta \lambda$
4:1 (4.16:1)	x	49.8 ± 0.7		6.76 ± 0.48		51.97 ± 0.5		6.51 ± 0.05	
	y	49.8 ± 0.7		6.77 ± 0.44				6.25 ± 0.13	4.23
	x=y	50.1 ± 0.41		6.73 ± 0.06					4.098:1
3:1 (3.14:1)	x	55.66 ± 0.46		7.08 ± 0.08		58.99 ± 0.26		6.39 ± 0.06	
	y	55.96 ± 0.26		7.02 ± 0.04				6.33 ± 0.05	3.46
	x=y								3.074:1
2:1 (2.12:1)	x					54.12 ± 0.33		6.76 ± 0.08	
	y							6.75 ± 0.07	2.54
	x=y	51.61 ± 0.61		7.43 ± 0.13					2.049:1
1.5:1 (1.61:1)	x					43.830 ± 0.34		6.95 ± 0.10	
	y							6.91 ± 0.11	2.42
	x=y	41.27 ± 0.61		7.75 ± 0.18					1.537:1
1:1 (1.09:1)	x	20.09 ± 1.18		8.04 ± 0.17		22.35 ± 0.3		7.2 ± 0.07	
	y	19.76 ± 0.68		8.10 ± 0.10					1.98
	x=y	19.91 ± 0.98		8.11 ± 0.15					1.025:1
									$\bar{\phi}_W/\bar{\phi}_U^2$

¹ The reflector savings quoted were computed assuming the lattice boundaries to be the metal in the x direction and half a water gap out in the reflector in the y direction.

² Estimated correction for fission products, not applied, is +1%. Estimated statistical error ±2%.

diagonally across its flat surface were milled five 0.010-inch-deep foil holes. An aluminum catcher was placed between the foils and the upper half of the split fuel disk, but, because of the size of the foils, no catchers were placed below them. The sides of the foils that faced down were marked, and only the upper sides were counted. This way, it was hoped to minimize the amount of fission-product activity picked up during exposure.

The foil activities were then determined by a least-squares fit with a parabola in the slab, and by Simpson's rules fit in the water. The resulting disadvantage factors are listed in Table I.

From Table I it can be seen that the reflector savings measured in the x and y directions are the same, but that the "variable" loading technique gives values from 0.5 to 1.0 cm larger than those found by the flux-shape technique.

Differences of this sort have been observed before in other BNL measurements using these techniques and there is, in addition, a theoretical basis² for expecting a difference in the reflector savings inferred by the two methods. Whether or not the magnitude of the difference observed can be properly accounted for by the theory will be shown by the detailed analysis now in progress.

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Measurements of Bucklings and Void Effects in D₂O-Moderated, Organic- or H₂O-Cooled Lattices of UO₂ Rod Clusters, F. D. Benton and W. E. Graves (duPont-SR)

Cost studies¹ have indicated that D₂O-moderated, organic-cooled reactors are attractive power-reactor can-

didates. The purpose of the present work was to provide normalization points for reactivity calculations on such reactors.

Bucklings were measured for fuel assemblies consisting of clusters of 19, 31, or 37 UO₂ rods in housing tubes filled with organic, H₂O, or air. These assemblies were chosen to cover an extreme range of hydrogenous coolant volumes rather than to mock up candidate power-reactor fuel assemblies. The UO₂ rods consisted of 0.500-in.-diameter sintered pellets of 10.4 g/cm³ average density stacked in aluminum tubes 0.020 in. thick and of 0.547 in. OD. These rods were assembled in triangular arrays to form clusters. The clusters were placed in aluminum housing tubes, which in most cases were selected to allow only a small portion of the coolant to lie between the outer cluster boundary and the housing wall. Dowtherm A (a mixture of 26.5% diphenyl and 73.5% diphenyl oxide) and H₂O were used to simulate coolants. Clusters in air-filled housings were also measured.

The experiments were carried out in the Process Development Pile (PDP).² Bucklings were determined by the substitution method, in which fuel assemblies to be measured replace assemblies in a one-region pile of known nuclear properties. The reference one-region loading consisted of 31-rod UO₂ clusters and had been used in previous experiments.³ Both reference and test lattices were at a 9.33-inch triangular pitch, which is in the range of practical interest. In addition to the measurement of the critical moderator height of the reference one-region pile, three successive measurements were made with one, three, and seven test assemblies in the central region for each type of test assembly.

Two analysis techniques were used: a one-group perturbation method⁴ and a two-group flux-matching method.⁵ In the perturbation method, the lattice-containing test assemblies was assumed to consist of a test region and a reference region, separated by an intermediate region between the test fuel assemblies and the reference assemblies. Each set of successive substitutions was expressed as a function of statistical weights, diffusion-coefficient ratios, and buckling differences between the one-region

TABLE I. Bucklings of Clusters of UO₂ Rods - 9.33-In. Lattice Pitch
(Moderator at 22.8 C and 99.58 mol % D₂O)

No. of Rods in Cluster	Rod Spacing Center-to-Center, In.	Coolant	Housing Tube Dimensions		Buckling, m ⁻²		
			OD, In.	Wall Thick, In.	Pert. Theory	Two Group	Average ^c
19	0.607	Organic	3.080 ^a	0.030	3.53	3.53	3.53
19	0.607	H ₂ O	3.080 ^a	0.030	2.06	2.07	2.07
19	0.607	Air	3.080 ^a	0.030	5.42	5.48	5.45
19	0.598	Organic	2.679 ^b	0.030	4.54	4.60	4.57
19	0.598	Air	2.679 ^b	0.030	5.60	5.60	5.60
31	0.607	Organic	4.000 ^a	0.050	2.89	2.96	2.93
31	0.607	H ₂ O	4.000 ^a	0.050	1.17	1.08	1.13
31	0.607	Air	4.000 ^a	0.050	4.91	4.90	4.91
37	0.607	Organic	4.350 ^a	0.050	2.45	2.58	2.52
37	0.607	H ₂ O	4.350 ^a	0.050	0.64	0.75	0.70
37	0.607	Air	4.350 ^a	0.050	4.36	4.33	4.35
37	0.607	Organic	4.746 ^a	0.050	0.28	0.33	0.31
37	0.650	Organic	4.746 ^a	0.050	1.05	1.16	1.11
37	0.650	Organic	4.970 ^a	0.162	-0.90	-0.70	-0.80
37	0.650	D ₂ O	4.746 ^a	0.050	3.99	4.01	4.00
37	0.650	Air	4.746 ^a	0.050	3.64	3.65	3.65

^a OD of cylindrical housing.

^b Outer dimension across the flats of a hexagonal housing.

^c Since there is no reason to choose one method over the other, the measured buckling is taken to be the average of the two results.

and three-region piles; each set was extrapolated to a condition at which the pile was a one-region lattice of test assemblies. The two-group method utilized only the seven-assembly substitution. In addition to the measured pile height and radius, it required knowledge of fast- and slow-diffusion coefficients, thermal-diffusion area, neutron age, and resonance-escape probability for each of the two regions. Results of both methods of analysis, along with a description of the test fuel assemblies, are summarized in Table I.

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Migration Area in Slightly-Enriched- UO_2 Light-Water-Moderated Lattices, M. Imai, T. Kanai, and H. Kobayashi (Hitachi)

Migration areas were derived from a one-group critical equation with experimental buckling coefficients accompanied by changes of lattice size in the OCF (a critical assembly designed and constructed by Hitachi Ltd. and located at Hitachi Central Research Laboratory, Tokyo, Japan). The first core loading of this assembly attained criticality on October 2nd, 1962 and the number of critical fuel pins for the fully reflected core was 401 instead of theoretical value of 407. The lattice reported here was uniform and cylindrical, consisting of 2.5%-enriched UO_2 fuel pins. The ratio of water-to- UO_2 volume was 2.5.

The size changes resulting in reactivity worths were accomplished by increasing water height of 1-5 mm above critical levels or adding a fuel pin at the cylindrical periphery. Reactivities were measured by the positive-period method with a BF_3 or a fission counter. The ratio of effective β to β for converting period to reactivity was assumed to be 1.10 by the calculation. Increments of water height were performed at various critical water levels, and buckling coefficients were determined from extrapolation of measured reactivities for various amounts of water-height increase. For these small water increases, reactivity worth was a linear function of water level. In the calculation of buckling coefficient, radial and bottom reflector savings were determined to be 7.5 cm and 6.3 cm from neutron flux fittings.

A top-reflector saving of 6.1 cm in a dry lattice above a wet core was calculated from the critical geometrical buckling of a fully reflected core having both top and bottom reflector savings of 6.3 cm. This was verified by the relation of the critical water level being proportional to the inverse cubic root of $\partial\rho/\partial H$. These results on axial buckling coefficients are shown in Table I together with radial buckling coefficients described below.

TABLE I. Buckling Coefficient

Core Geometry	Critical Core Height cm	B^2 axial cm^2	B^2 radial cm^2
Cylinder	80.00	---	32.54
	70.27	28.86	29.60
	66.19	30.32	31.08
	63.98	31.22	---
Rectangular	55.43	33.01	---
Parallelopiped	55.24	33.01	---
	49.66	33.00	---

Fuel pins around the periphery of the cylindrical core were removed from the lattice and reinserted to give size changes. These pins indicated different worths depending upon the arrangement of lattice around these pins, even if these were at the same distance from the core center. The slight irregularity on cylindrical periphery was averaged out, and the averaged value of pin worths at the same distance from the center were used for the calculation of the radial buckling coefficient.

Using a one-group critical equation (with constant reflector savings assumed) migration areas of 44.6 cm^2 in radial (M_r^2) and 46.0 cm^2 in axial direction (M_a^2) were determined. Theoretical value for migration area for this lattice is 42.29 cm^2 using Mercury code in which Deutsch's Method is used.¹

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Criticality of Solutions and Radioactive Shipments

Critical Interaction Potentials of Water-Mod-
erated UF₆ Arrays, C. E. Newlon (UCNC)

The critical spacings of individually subcritical units in large arrays may be computed by use of an interaction potential method which is based on the solid angle of neutron interaction. The method has been written in a Fortran code for two- or three-dimensional arrays for use with the IBM-7090.

When two or more identical subcrits are considered as an interacting array, their nuclear reactivity may be expressed as an interaction potential I_m ,

$$I_m = (k_{\infty} - 1)U/(1 - U) \quad (1)$$

where $(k_{\infty} - 1)$ is the excess multiplication factor and $U/(1 - U)$ is the ratio of the neutron nonleakage to leakage probabilities of the units. The interaction potential curves shown in Figure 1 were determined for individually safe units of water-moderated UF₆ at various U-235 enrichments (see Table I).¹

At criticality, the interaction potential of the subcrits I_m must equal the neutron leakage probability I_g of an n -unit array, which is given as:

$$I_g = 1 - \sum_{i=1}^n (q_i \Omega_i) \quad (2)$$

Fig. 1. Interaction Potential, I_m - Infinite Length Cylinders.

TABLE I. Nuclearly Safe Units For UF₆ Systems
(Optimum water moderation and reflection
maximum uranium density, 3.2 g U/cm³)

U-235 Enrichment (wt %)	Infinite- Length Cylinder Diameter (in.)	Sphere	
		Diameter (in.)	Volume (liters)
93.0	5.0	8.2	4.8
40.0	6.0	9.2	6.7
10.0	8.2	11.8	14.0
5.0	10.25	14.6	27.0
2.0	16.0	22.3	95.1
1.5	20.0	27.9	186.0
1.1	29.0	39.4	524.0

Note: All dimensions refer to internal diameters.

Fig. 2. Minimum Critical Interaction Probability vs. Enrichment, Safe Units of Table 1.

where Ω_i is the fractional solid angle of neutron interaction subtended at the centralmost unit by the i th unit of the array, and q_i is a flux-distribution factor. Thus, the critical interaction probability is given as:

$$\sum_{i=1}^n (q_i \Omega_i) = 1 - I_m \quad (3)$$

In Figure 2, the minimum critical interaction probability (maximum interaction potential) for the units of Table I is plotted as a function of the U-235 enrichment. At 93%

U-235, the calculated $\sum_{i=1}^n (q_i \Omega_i)$ is 0.561 for infinite-

length 5-in.-ID cylinders, which compares with a value of 0.570 derived from experimental separation measurements of arrays with up to 100 units of enriched uranium solu-

tion;² a calculated value of unity is obtained for $\sum_{i=1}^n (q_i \Omega_i)$

at the minimum critical enrichment of 1.0% for UF₆ systems.

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2. GILLEY, L. W., D. F. CRONIN, J. K. FOX, and J. T. THOMAS, "Neutron Physics Division Annual Progress Report for Period Ending September 1, 1961," ORNL-3193 (October 1961), p. 162.

Interaction in Arrays: A Comparison of The-
ory and Experiment, H. K. Clark (DuPont-SR)

In Reference (1) a method for computing the interaction of fissionable units is described that in comparison with critical experiments performed with small groups of slabs and cylinders has been found to be reasonably accurate

and usually conservative. The present paper compares this method with recent critical experiments² performed with large unreflected and reflected cubic arrays of cylinders containing U-235 as an aqueous solution.

The method involves calculating the neutron current entering a unit from currents leaving other units under the assumptions that the currents are uniform over the emitting surfaces and that the angular distribution of emitted neutrons per unit solid angle is proportional to the cosine of the angle made with the normal to the emitting surface. The current emitted from a unit is obtained as the product of the albedo (>1 for multiplying units) of the unit and the current entering the unit. Albedos are computed from dimensions and compositions for all units; dimensions, compositions, or spacings are adjusted until the determinant of the coefficients of the equations in the emitted currents is zero. It is convenient and conservative¹ to approximate spheres by cubes of the same volume in computing the probability that neutrons from one unit reach another. The fraction of 4π steradians subtended at the center of an emitting sphere is an underestimate of this probability. In arrays the conservative approach is taken that, except for completely shielded units, entire units are "seen" until the sum of the fractions of the neutrons reaching successively further units in an infinite array of the same type is unity.

For purposes of calculation the "Plexiglas" walls of the cylinders employed in the experiments were treated as H_2O and were mixed in with the solution, giving a concentration of 349 g/U (92.6% U-235) per liter. The homogenized units were approximated by spheres having a radius of 11.25 cm. The reflectors were all assumed to be H_2O , and the effect of thickness was calculated from experimental data expressed as albedos³. The material buckling of a unit was calculated to be 0.0306 cm^{-2} which, with bare and water reflected extrapolation distances of 2.6 and 5.8 cm, is consistent with experimental data for similar solutions³.

In Table I results of the calculations are compared with experiment in terms of k_{eff} calculated from the geometric buckling a unit has when the effective reflector saving of its surroundings is assumed to be independent of the size of the unit. Possible reasons for the somewhat excessive conservatism in the calculations include lumping the Plexiglas walls in with the solution, approximating the cylinders by spheres, the treatment of partial shielding in the lattice, and overestimating the reactivity of an isolated unit.

TABLE I. Comparison of Theory and Experiment

No. of Units	Reflector Thickness, cm	Surface-to-Surface Separation, cm	k_{eff} Calculated for Critical Array	
			Cube Approx.	Solid Angle
8	0	1.43*	-	-
	1.27	3.00**	1.031	0.977
	1.27	3.28**	1.027	0.968
	3.81	6.91	1.043	1.001
	7.62	8.48	1.064	1.040
27	0	8.99	1.072	1.046
	1.27	6.48	1.096	0.995
	1.27	8.76**	1.082	1.017
	1.27	9.02**	1.075	1.009
	3.81	13.69	1.045	1.016
64	0	10.67	1.105	1.032
	15.24	16.53	1.063	1.050
125	0	14.40	1.087	1.045

*The equivalent spheres are in contact when STS = 2.5; hence no calculation was made.

**The smaller separation applies to Plexiglas. The other reflectors were paraffin.

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2. THOMAS, J. T., "Critical Cubic Arrays of Neutron Interacting Units," (to be issued as an ORNL report).
3. CLARK, H. K., Handbook of Nuclear Safety, Report DP-532.

Studies of Critical Cubic Arrays of Fissile Materials, J. T. Thomas (ORNL)

This summary reports two critical experiments with cubic arrays of individually subcritical units of fissile materials intended to further establish bases for regulations governing their transport and storage. Preliminary experiments in this program were reported most recently by Gilley and Thomas¹ and by Mihalcz and Lynn.²

In the first of the current experiments the fissile units were right circular cylinders of 5.000 ± 0.003 liters of aqueous uranyl nitrate solution contained in 0.635-cm-thick methyl methacrylate plastic (Plexiglas, $\rho = 1.18 \text{ g/cm}^3$) 20.32 cm outside diameter and 19.05 cm high. The concentration of the solution was 415 g of U(92.6)* per liter and the specific gravity was 1.555 at 25 C. The H:U-235 atomic ratio was 59. The mass of U-235 in each unit was 1.934 kg. Arrays of 8, 27, 64, and 125 units in a cubic lattice were constructed within an aluminum supporting framework.

Metal cylinders of U(93.2) having a diameter and a height of 11.506 and 10.770 cm respectively were similarly latticed in the second set of experiments. The density of the uranium was 18.66 g/cm^3 and each unit contained 19.457 kg of U-235. The cylinders were supported on stainless steel rods passing through two 0.508-cm-dia holes parallel to the axis of the cylinder and centered

*U(92.6) designates uranium of 92.6 wt% U-235 content.

Fig. 1. Dependence of Number of Units in a Critical Array on Surface-to-Surface Separation and Paraffin Reflector Thickness.

TABLE I. Dependence of Constants N_0 and s on Paraffin Reflector Thickness

Type of Unit	Reflector Thickness (cm)	N_0	s
Solution	0	2.506 ± 0.014	1.928 ± 0.028
"	1.27	1.699 ± 0.316	1.831 ± 0.118
"	3.81	0.794 ± 0.125	1.797 ± 0.098
"	15.24	0.606 ± 0.084	1.721 ± 0.083
Metal	0	2.407	1.524
"	1.27	1.673	1.436
"	7.62	0.249	1.473
"	15.24	0.208	1.510

8.547 cm apart. The cylinders were located vertically on these rods by spacers of appropriate length cut from light close-fitting stainless steel tubing. Arrays of 8 and 27 of these units have been constructed.

The critical surface-to-surface spacing, equal in all three directions, was measured in both experiments as a function of the number of units in an array and of the thickness of a paraffin reflector ($\rho = 0.93 \text{ g/cm}^3$) surrounding the array. The inside surface of the reflector was displaced from the outer surface of the peripheral units a distance equal to one half the surface-to-surface separation of the units themselves. The results are plotted in Fig. 1.

The data from tests with the solution units have been found to be adequately represented, over the range studied, by the expression

$$N = N_0(\rho/\rho_0)^{-s}$$

where N is the number of units in a critical array, ρ/ρ_0 is the ratio of the average uranium density of a cell to the uranium density of the unit, and N_0 and s are constants. By applying this expression also to the metal-cylinder data, values of N_0 and s , which are dependent upon the reflector condition, have been obtained and are tabulated.

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2. MIHALCZO, J. T. and LYNN, J. J., *Trans. Amer. Nucl. Soc.* 2, 80 (1959).

A Study of Idealized Plutonium Metal Dissolvers and the "Always Safe" Conditions, W. A. Reardon and F. R. Czerniejewski (GE-HAPO)

This describes a study of the critical-volume/critical-mass relationship for an idealized plutonium dissolver, and derives the "always safe" envelope of the resulting curves derived. In addition, it is shown that, at least for the conditions assumed, it is possible to start dissolution in a system that is subcritical at both the starting configuration and the fully dissolved end point, and achieve a supercritical configuration somewhere in between.

The idealized systems consisted of concentric spheres of α -phase plutonium metal ($d = 19.7 \text{ g/cm}^3$), $\text{PuO}_2 + \text{H}_2\text{O}$ water "solution," and a 14-cm H_2O reflector. A compromise with reality is made in the assumption of a $\text{PuO}_2 + \text{H}_2\text{O}$ "solution." The idealized solution more nearly represent dissolving operations than the idealized plutonium/water systems. The results are directly applicable to many hood-design problems involving operations with plutonium in metal and oxide form.

The series of curves in Figure 1 show the critical masses (total Pu-239 in metal sphere and solution) and

the corresponding critical volume of the dissolver (regions I and II). The "always safe" envelope is drawn tangentially to the various curves and extrapolated to the single point corresponding to a critical mass of 5.5 kg Pu-239, infinitely reflected by light water. This point together with the curve for the homogeneous systems serve as check points for the computations. The "always safe" region is the region below the envelope. The points shown on the curves are computed. All the heterogeneous computations were carried out by an S_2 -approximation transport-theory code¹, while the homogeneous curve was obtained by a multi-group diffusion-theory computation². The point corresponding to a homogeneous solution containing 100 g/liter Pu was computed both ways as a check; it is the point where the envelope joins the homogeneous curve. The cross sections used in the transport-theory computations are the 16-group set in LAMS-2543³. The cross sections used with the multi-group diffusion theory are those in HW-71772⁴. The cross sections and method were tested by computing several known critical masses of bare and reflected plutonium and uranium critical systems⁵.

Fig. 1. Critical Mass (Total Pu-239) vs Volume α -Pu-239 in $\text{PuO}_2 + \text{H}_2\text{O}$ Solution.

The possibility of a system, subcritical at the beginning and end of the dissolution, being critical at some intermediate stage is shown by the example of a 3-kg mass dissolving into a 5-liter volume. The critical masses of systems whose composition corresponds to a partially dissolved 3 kg mass were determined from the data. The critical mass falls below 3 kg total mass at 35 g/liter concentration and reaches a minimum of 2.8 kg at about 100 g/liter and rises above 3 kg again at about 350 g/liter.

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The Effect of Borosilicate Glass Raschig Rings on the Criticality of Aqueous Uranyl Nitrate Solutions, J. T. Thomas, J. K. Fox,* and E. B. Johnson (ORNL)

The capacity, or safety, of processes with fissile materials can be increased significantly by introducing neutron-absorbing elements into the process stream (provided, of course, that the absorber is compatible with process materials). Borosilicate glass has recently been studied for this application by Biding et al.¹ and its use is becoming extensive in large storage vessels and in components of process trains. A program of experiments intended to establish better bases for use of these glasses in nuclear-safety problems was initiated at the Oak Ridge Critical Experiments Facility several years ago.² A more extensive study³ has now been completed in which glass Raschig rings, varying both in dimensions and in boron content, were immersed in cylinders of different diamet-

ers containing enriched uranyl nitrate solutions of several concentrations.

It was impossible to make most of the solution/ring mixtures critical and, in fact, the source neutron multiplication was often too low to allow sensible extrapolation. Addition of only a few inches of solution above a very subcritical mixture established a critical system that served as a source of neutrons distributed over the mixture and had spectral characteristics that allowed a measure of the neutron relaxation length in the mixture. The material buckling estimated for these exponential-experiment data is somewhat uncertain because values of the radial extrapolation distance were unavailable. It has been possible to show, however, that the material buckling of some of the mixtures studied is negative.

The conditions necessary to maintain criticality during the flux measurements in the cylinders are given in Table I. Random irregularities in the top surface of the ring bed introduced an uncertainty of ± 0.3 in. in the glass/solution mixture height and must be considered in any attempt to assign a thickness to the critical slab of solution when reflected on one surface. A portion of the neutron

TABLE I. Results from Experiments with Mixtures of Uranyl Nitrate Solution and Borosilicate Glass U-235 Enrichment = 92.6 wt%

Diameter of Cylinder (in.)	Reflector ^a	Type of Glass	Boron Content of Glass (wt%)	Glass Content of Mixture (vol %)	Glass Height (in.)	Solution Height (in.)	γ (Relaxation Length) ⁻¹ (in. ⁻¹)	Error in γ (95% Confidence Level)
Solution Concentration: 415 g of U per liter (H:U-235 = 59)								
48	Air	EN-1	5.7	24.1	46.5	44.68 ^b	-	-
	H ₂ O	EN-1	5.7	24.1	46.5	44.68 ^b	-	-
	Air	EN-1	5.7	24.1	30.0	34.31	0.189	± 0.022
	H ₂ O	EN-1	5.7	24.1	30.0	34.25	0.179	± 0.027
30	H ₂ O	Pyrex	4.0	20.9	50.0	50.0 ^b	-	-
	H ₂ O	Pyrex	4.0	20.9	35.9	39.49	0.092	± 0.016
	Air	Pyrex	4.0	20.9	35.9	39.75	0.118	± 0.015
20	H ₂ O	EN-1	5.7	24.1	34.6	40.64	0.257	± 0.017
	Air	EN-1	5.7	24.1	34.6	41.03	-	$\pm$ -
	Air	KG-33	4.1	24.1	35.6	41.32	0.219	± 0.007
	H ₂ O	KG-33	4.1	24.1	35.6	40.84	0.198	± 0.018
	H ₂ O	Pyrex	4.0	20.9	33.8	38.53	0.145	± 0.011
	Air	Pyrex	4.0	21.0	33.6	38.67	0.171	± 0.016
	H ₂ O	Pyrex	4.0	21.0	33.6	38.29	0.142	± 0.017
	H ₂ O	KG-33	3.9	30.0	27.2	32.37	0.307	± 0.013
	Air	KG-33	3.9	30.0	27.2	32.69	0.322	± 0.012
	Air	R-6	0.5	24.0	21.4	10.86	{Note that this mixture was critical}	
	H ₂ O	R-6	0.5	24.0	21.4	8.34		
Solution Concentration: 279 g of U per liter (H:U-235 = 92)								
20	Air	KG-33	4.1	24.1	35.6	41.48	0.257	± 0.016
	H ₂ O	KG-33	4.1	24.1	35.6	41.12	0.239	± 0.029
Solution Concentration: 141 g of U per liter (H:U-235 = 191)								
20	Air	KG-33	4.1	24.1	35.6	41.47	0.312	± 0.022
	H ₂ O	KG-33	4.1	24.1	35.6	41.05	0.321	± 0.026
	Air	R-6	0.5	24.0	21.0	25.68	0.112	± 0.007
	H ₂ O	R-6	0.5	24.0	21.0	24.95	0.074	± 0.013
20 ^c	Air	R-6	0.5	24.0	21.0	25.63	0.108	± 0.006
	H ₂ O	R-5	0.5	24.0	21.0	25.29	0.088	± 0.005
Solution Concentration 94.4 g of U per liter (H:U-235 = 290)								
20 ^c	Air	R-6	0.5	24.0	21.0	26.86	0.185	± 0.011
	H ₂ O	R-6	0.5	24.0	21.0	26.51	0.176	± 0.011
Solution Concentration: 63.3 g of U per liter (H:U-235 = 406)								
20 ^c	Air	R-6	0.5	24.0	21.0	28.23	0.264	± 0.016
	H ₂ O	R-6	0.5	24.0	21.0	27.78	0.265	± 0.014

a. In instances of water reflection, the bottom and lateral surfaces to approximately the height of the solution were surrounded by an effectively infinite thickness of water.

b. This assembly was subcritical.

c. 0.032-in-thick cadmium sheet around cylinder.

flux distribution observed along the axis of each glass/solution mixture was fitted by least squares to the relation

$$\phi = \phi_0 e^{-\gamma z}$$

to obtain the values of γ listed in the table.

The material buckling for each exponential system may be calculated by assuming a value of the radial extrapolation distance. It was possible to estimate the radial extrapolation distance for reflected and unreflected mixtures of glass and the most concentrated solution as 5.0 and 2.9 in., respectively. It can be concluded that $k_{\infty} \leq 1$ for every mixture of this solution at a concentration of 415 g of U per liter and borosilicate glass, provided the glass occupies at least 24.1% of the volume, contains a minimum of 3.3 wt% of boron and is uniformly distributed.

The assembly containing rings of 0.5 wt% boron in concentrated solution was the only one that was critical with the solution height equal to or less than the height of the ring bed. These experiments and others with subcritical mixtures of these rings and solutions of lower concentration are reported in Table I. Setting the extrapolation distance equal to zero gives maximum values of B_m^2 so that conservatism is introduced upon application of the results to safety problems. A plot of the maximum values of B_m^2 against solution concentration indicates that any system containing Raschig rings of 0.5 wt% boron which occupy at least 24 volume per cent of the mixture will have a value of $k_{\infty} < 1$ if the solution concentration is ≤ 72 g of U per liter (H:U-235 = 380).

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Impact Testing of Shipping Containers, L. B. Shappert (ORNL)

With the increasing number of shipments of radioactive material being made across country, the problem of ensuring the structural integrity of a shipping cask becomes critical; some government agencies are now setting standards to which casks must conform. Proposed regulations¹ require in part that a cask must withstand both a 15-ft free fall onto an unyielding surface and the puncturing action of a piston. This paper describes the investigation of the behavior of casks under the above conditions. Seven cylindrical lead-filled casks were built. Five were 18 in. OD, 10 in. ID, 36 in. long, had an outer shell thickness 0.312 in. and weighed 1.4 tons. Two were 30 in. OD, 18 in. ID, 60 in. long, had an outer shell thickness 0.500 in. and weighed 6 tons.

These casks were instrumented primarily as shown in Fig. 1. In addition, several were coated with a brittle lacquer, which gave qualitative evidence of the magnitude and direction of the strains at the time of impact.

The impact surface consisted of a 4½-in. thick slab of armor plate backed up by more than a 5-ft thickness of reinforced concrete.

The casks were dropped horizontally from heights below 30 ft, on a corner below 23 ft, on end below 16 ft, and

Fig. 1. Cask Instrumentation and Testing Facility.

on a piston below 6.5 ft. High-speed motion pictures were made of each drop.*

Results indicate that:

- (1) Circumferential weld failures due to corner drops were noted for both cask sizes above a drop height of 15 ft. However, no attempts were made to get high-quality welds and, in some cases, weld penetration was less than 50%.
- (2) The only circumferential weld failure that occurred in a horizontal drop was found on a 1.4-ton cask dropped 28 ft. The cask had been previously dropped 15 ft on the same side.
- (3) End drops caused little visible damage to the 1.4-ton casks.
- (4) Higher stresses were recorded on the inside shell than in a comparable location on the outer shell, except at the area of impact.
- (5) High stresses were found at discontinuities such as lead fill-hole plugs. Care must be taken in welding such holes.
- (6) A sharp-edged 2-in.-diameter piston penetrated the outer shell of a 1.4-ton cask at drop heights above 2 ft.
- (7) The duration of the initial impact was, in general, less than 10 milliseconds, and the maximum number of g's was reached in less than 5 milliseconds.

The casks withstood the majority of impacts well, with the exception of the piston impacts. However, in no test was the inner cavity penetrated. If CFR Title 10, Part 72 continues to require a penetration test, additional impact shields may have to be built around some casks.

1. Proposed CFR, Title 10, Part 72, Federal Register - September 23, 1961.

*Shown during presentation of paper.

Critical Experiments I

Critical Experiment with Borax-V Superheater Elements, K. E. Plumlee, Q. L. Baird, G. S. Stanford, and P. I. Amundson (ANL).

A critical experiment was performed at room temperature with 12 BORAX-V superheater fuel elements surrounded by a peripheral region containing 1228-1525 UO₂ fuel pins, as shown in Fig. 1. Measurements were needed to supplement the available computational information since the neutron energy spectrum, effective neutron cross sections, flux intensity and power production are poorly defined by theory near interfaces such as that between the superheater and the peripheral region, and also in the presence of voids such as exist in the superheater region (26% void).

The experiment was performed in the ZPR-VII experimental facility at Argonne rather than in the BORAX-V reactor.¹ Each superheater fuel element contained five stainless-steel boxes with four fuel plates per box. The inner pair of fuel plates contained 27.9 g U-235 each, and the outer two 15.0 g each. Thin voided regions separated the fuel plates from the moderator and from each other. The fuel plates consisted of highly enriched UO₂ contained in type-304 stainless-steel cermet core with type-304 stainless-steel cladding. The peripheral-zone fuel was enriched to 3 wt% and was contained in aluminum tubing arranged in a 1.27 cm square lattice. Pellet diameter was 0.935 cm, with a density of 10.17 g/cm³.

A well-flattened distribution of epi-cadmium fissions was measured in the superheater. The sub-cadmium distribution contained large perturbations. The radial epi-cadmium peak-to-average fission ratio was 1.02, as compared with RE-6 and PDQ computed values of 1.02 and 1.06 for homogenized superheaters of identical fuel and hydrogen content but at 526.5K. The computations appeared to slightly overestimate buckling and to underestimate the flux significantly near the superheater boundary. Because of the sub-cadmium perturbations, there was an edge-to-average fission ratio of ~1.22 in individual fuel plates, and an outer-fuel-plate to average-fuel-plate ratio of ~1.07, giving a maximum-to-average midplane fission probability in U-235 of ~1.30. Relative midplane fluxes on the line through the center of the fuel clusters were as follows: surface of fuel cluster box, 1.00; average in inter-box moderator, 1.04; average in outer fuel plate, 0.95; average in inner fuel plate, 0.89.

Fig. 1 Boron V. Superheater, Zone in ZPR-VII.

The effect of voiding the central superheater was found to be $-2.2\% \rho$ for 26% void, giving an average coefficient of $-0.085\% \rho/\%$ void in that region. A loss of $0.47\% \rho$ was attributed to voiding of the first quadrant, 0.43% to the one diagonally opposite, and 1.2% to the remaining two quadrants. However, no significant loss was measured when the central cruciform safety guide was voided. Plastic pieces were packed into the moderator space inside the quadrant cans (see Fig. 1).

Interest in spectral information led to measurements of cadmium ratios for the activations of Au, Dy, Cu, In, Lu-175, Lu-176, Mn, and U-235 (fissions) at six positions in the superheater and seven in the peripheral zone, in the midplane. The cadmium ratio for infinitely thin gold was ~1.2, indicating that the spectrum was hard enough that a Maxwellian energy distribution does not apply, and consequently, neither does the Westcott cross-section convention.² Significant deviations from a $1/E$ slowing down spectrum were indicated. Based on the cadmium ratios, the flux per unit lethargy, instead of being constant, was some 15%-30% higher at 500 eV than at 5 eV, the results being generally consistent with GAM-1 calculations.

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Critical Experiments and Calculations with Annular Cylinders of U(93.2) Metal, J. T. Mihalczo (ORNL).

A series of critical experiments with annular cylinders of uranium metal has been performed at the ORNL Critical Experiments Facility to determine the adequacy of the S_n method of solving the transport equation for this geometry. The uranium has a density of 18.76 g/cm³ and is enriched to 93.2% in the U-235 isotope. Unreflected annuli with outside diameters as large as 15 in. and inside diameters as small as 7 in. have been assembled to delayed critical, and the prompt-neutron decay constant has been measured by the Rossi- α technique.¹ The resulting values of the decay constant, together with the measured multiplication constant of the various assemblies, are shown in Table 1.

Multiplication constants of the experimental configurations were calculated by multigroup transport theory using the S_n method² in the S_2 approximation, the TDC computer code,³ and the six-group cross sections of Hansen and Roach⁴ for U-235 and U-238. Fission of U-234 and of U-236 was included, and their absorption and scattering properties were assumed to be the same as those of U-238. The calculated values of the multiplication constant, included in Table 1 are in good agreement with those measured.

The prompt-neutron decay constants obtained from the Rossi- α measurements with the annular cylinders at delayed critical are somewhat smaller than those obtained for solid uranium metal assemblies since the flight time of the neutrons across the center void constitutes a large fraction of the prompt-neutron lifetime.⁵ At delayed crit-

TABLE I. Experimental and Calculated Results for Cylindrical Annuli of U(93.2) Metal

Dimensions		Average Height (in.)	Mass (kg of U)	Content of Center Cavity	Multiplication Constant		Prompt-Neutron Decay Constant (sec ⁻¹)
Outside Diameter (in.)	Inside Diameter (in.)				Measured ^b	Calculated	
11	7 ^d	7.31	157.9	Air	0.9999	0.9900	$(0.686 \pm 0.008) \times 10^6$
13	7	5.74	165.6	Air	1.0000	1.0014	$(0.883 \pm 0.009) \times 10^6$
13	7	5.26	151.9	C	1.0001	1.0013	-
13	9 ^e	6.08	128.7	C	1.0000	1.0033	-
15	7	4.23	179.0	Air	0.9999	0.9981	$(0.950 \pm 0.010) \times 10^6$
15	7	4.10	173.6	C	0.9998	0.9963	-
15	7	3.98	168.6	Be	1.0000	1.0035	-
15	9	5.93	205.6	Air	0.9998	1.0027	$(0.840 \pm 0.008) \times 10^6$
15	9	5.35	185.3	C	0.9999	0.9980	-

^aThe uncertainty in the height is less than 0.03 in.

^bCorrected for the support structure.

^cError from least-squares analysis of the counter data.

^dA 7-in.-diam, 1.38-in.-thick disc was located at the bottom of the cavity and a 7-in. O.diam, 1.25-in.-thick disc was located with its top surface 0.13 in. below the top of the annulus.

^e1-in. carbon reflector.

ical the prompt-neutron decay constant is inversely proportional to the prompt-neutron lifetime, and the proportionality constant (the effective delayed-neutron fraction) is taken to be 0.0087. The prompt-neutron lifetime for the 11-in.-OD $\times$ 7-in.-ID assembly is 9.8×10^{-9} sec, compared to 6.2×10^{-9} sec for solid uranium metal assemblies.

The effect of inserting graphite (National Carbon type ATL; $\rho = 1.7$ g/cm³) and beryllium ($\rho = 1.83$ g/cm³) into the center cavity has been measured and a comparison of experiment and calculation is also given in Table 1. The calculations of these experiments by the TDC code used the 16-group cross sections of Hansen and Roach³ with U-234 and U-238 treated as above.

In general the agreement between calculation and experiment is good and demonstrates the adequacy of the S_n method in the S_0 approximation for assemblies with large central voids.

1. ORNDOFF, J., "Prompt Neutron Periods of Metal Critical Assemblies," *Nuc. Sci. Eng.* 2, 450 (1957).
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3. CARLSON, B., et al, "The DSN and TDC Neutron Transport Codes," LAMS-2346 (Feb. 17, 1960).
4. HANSEN, G. E. and W. H. ROACH, Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies, LAMS-2543 (1961).
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Clumped Light-Water-Moderated UO₂ Superheat Criticals: Part I, Experiments, F. G. Warzek and H. F. Johnston (GE-Vallecitos).

A series of critical and subcritical measurements was conducted in General Electric's Critical Experiment Facility at its Vallecitos Atomic Laboratory with superheat fuel in a lattice simulating the ESADA-Vallecitos Experimental Superheat Reactor (EVESR). The EVESR, a separate superheat reactor, is being built by General Electric under the joint sponsorship of General Electric and ESADA.* The fuel utilized for the critical experiments was fabricated for a prior AEC-sponsored critical experiment program.¹ Although the EVESR fuel has slightly different dimensions, enrichments, and cladding materials,² the basic fuel geometry is identical to the AEC fuel.

The AEC fuel element is annular, providing two heat-transfer surfaces for superheating steam. It consists of an annular ring of UO₂ (0.632-inch ID, 1.187 OD) enriched to 3.4 wt% U-235, clad internally and externally with 0.028-in. stainless steel 304 tubing. The clad fuel is enclosed in a 0.025-in.-thick, 1.50-in. OD, stainless steel 304 process tube which separates the outer steam coolant passage from the light-water moderator. Nine fuel elements in a 3 $\times$ 3 rectangular array are present in each fuel bundle; 32 such fuel bundles represent the full core loading. Each bundle is enclosed in a 5.76-inch-ID square aluminum channel which has a 0.125-in. wall. The fuel element pitch is 1.82 inches, the bundle pitch 6.60 inches and the active fuel length is 48 inches. The effective UO₂ density is 9.75 grams/cm³, accounting for gaps and pellet chips. The overall water-to-fuel volume ratio is 3.46

*Empire States Atomic Development Associates, an organization of the following New York State electric utility corporations:

Central Hudson Gas and Electric Corporation, Consolidated Edison Company of New York Incorporated, Long Island Lighting Company, New York State Electric and Gas Corporation, Niagara Mohawk Power Corporation, Orange and Rockland Utilities Incorporated, Rochester Gas and Electric Corporation.

when all control rods are withdrawn. The core accepts 12 main and 4 auxiliary control rods. The main control rods are cruciforms of pure stainless steel 304, 10.688 inches in span, and 0.390 inches thick. The auxiliary control rods are boron-stainless steel (2 wt% boron) slabs 5.344 inches in span and 0.390 inches thick.

The measurements performed can be grouped in three categories:

1. Reactivity Measurements with no Control Rods
2. Full Core Reactivity Measurements with Control Rods Present
3. Power-Distribution Measurements in the Full Core with Control Rods Present

For safety, all the critical measurements were made with the reactor core heated to 75-80 C to minimize the reactivity addition to atmospheric boiling in the event of a nuclear excursion. As a consequence, pulsed-neutron subcritical reactivity measurements were utilized in determining temperature reactivity effects.

Void coefficients were measured through the use of thin-walled aluminum tubes. The reactivity differential between the void and no-void condition was evaluated by a measurement of the change in the reactor period between the two conditions with the core fully reflected.

The full core excess, $2.8 \pm 0.5\% \Delta k/k$, was obtained by extrapolating subcritical data obtained by the pulsed-neutron-source technique to the full core size. Flooding effects, fuel worths and control-rod worths were also measured by the pulsed-neutron-source technique.

The results of several of the above key reactivity measurements are given and compared to the analytic predictions in the following paper.³

The power distribution as shown on Figure 1 was determined by circumferential U-235 foil activations on each rod in an octant of the plane located 12 inches below the top active surface of the core. The four central control rods were inserted 24.38 inches from the top active surface of the core during this measurement. The peak element power to the plane average was under-predicted by approximately 5% $\Delta P/P$. The power-distribution measurements were of specific interest due to the large circumferential power variations both predicted and measured around the large superheat fuel elements ($\pm 20\% \Delta P/P$).

1. PETERSEN, G. T. and F. G. WARZEK, "AEC Superheat Criticals—A Comparison of Experiment and Theory on Uniform Lattices," GEAP-3882, (January, 1962).
2. Final Hazards Summary Report for the ESADA Vallecitos Experimental Superheat Reactor, APED-3958, (October, 1962).
3. PETERSEN, G. T., "Clumped Light Water Moderated UO_2 Superheat Criticals—Part II Analyses," These Transactions, p. 220.

Clumped Light-Water-Moderated UO_2 Superheat Criticals: Part II, Analyses, G. T. Petersen (GE-Vallecitos).

Critical and subcritical measurements were conducted as described in the preceding paper.¹ The resulting measurements have been compared to the predictions of a physics design model which has demonstrated good accuracy in treating uniform arrays of the same fuel.²

The physics model relies on several computer codes to provide the necessary individual parameters. It incorporates three energy groups and diffusion theory to predict neutron leakage. The upper two energy-group parameters are determined by the MUFT-IV³ code and then modified to agree with independent calculations of resonance capture, fast fission and enhanced void leakage. The resonance capture in U-238 is predicted by Hellstrand's⁴ surface-to-mass ratio dependence of the U-238 effective resonance integral. The fast fission in U-238 is predicted by the one-dimensional three-region technique prescribed by Roderick.⁵ The enhanced void leakage is predicted by Behrens' prescription.⁶ The resulting modified MUFT parameters are then used for groups one and two of the three-group model.

The thermal energy group is treated by a two-zone spectral technique² which is normalized against one-dimensional multi-group thermal-spectrum calculations^{7,8} to yield the correct spectrally averaged parameters. This technique utilizes the Wilkins heavy-gas model to determine the thermal spectrum and the P-3 flux solution to determine the spatial weighting within the fuel element cell. Finally, the gross flux weighting across a fuel bundle and across the entire lattice is performed using two- or three-dimensional diffusion theory with cross sections for the water and fuel channel between fuel bundles spectrally averaged over a Maxwellian at the temperature of the moderator, and blackness self-shielded spectrally averaged cross sections for the control rods. The detailed power-distribution calculations utilized three-dimensional calculations for local rod power and two-dimensional calculations for the local circumferential power variation around each fuel rod.

Fig. 1. Measured and Calculated Rod Power to Plane Average, EVESR.

TABLE I. EVESR Critical Results Versus Calculations

Core Measurements With All Rods Out	Calculated	Measured
Minimum Critical Size at 77.7 C	23 Bundles	23 $\frac{1}{2}$ Bundles
Void Coefficient for Approximately Uniform Voids at 80 C	$-1.0 \times 10^{-4} \Delta k/\%$	$-2.2 \times 10^{-4} \Delta k/\%$
Temperature Coefficient at 80 C	$+1.9 \times 10^{-4} \Delta k/^{\circ}C$	$+2.1 \times 10^{-4} \Delta k/^{\circ}C$
Full Core Excess Reactivity at 80 C (Extrapolated by differential reactivity measurements)	3.2% $\Delta k/k$	$2.8 \pm 0.5\% \Delta k/k$
Full Core Measurements		
Control-System Worth, Unflooded:		
Worth of Inner Bank of 4 Control Rods	7.2% Δk	$6.0 \pm 0.7\% \Delta k$
Worth of 12 Regular Control Rods	15.1% Δk	$12.5 \pm 0.7\% \Delta k$
Worth of 4 Auxiliary Scram Rods	1.2% Δk	$1.0 \pm 0.2\% \Delta k$
Control-System Worth, Flooded:		
Worth of 12 Regular Control Rods	12.2% Δk	$10.4 \pm 0.7\% \Delta k$
Temperature Effect:		
Temperature Swing 20-80 C Unflooded with Operating Control Distribution*	+1.5% Δk	$+1.3 \pm 0.1\% \Delta k$
Temperature Swing 20-80 C Flooded with Operating Control Distribution*	+1.6% Δk	$+1.8 \pm 0.1\% \Delta k$
Void Coefficient:		
Void Coefficient for Voids Centered in the Moderator in the Unflooded Operating Control Distribution*	---	$+2.0 \pm 0.1 \times 10^{-4} \Delta k/\%$
Void Coefficient for Moderator Voids Adjacent to the Process Tube in the Unflooded Operating Control Distribution*	---	$-13.1 \pm 0.4 \times 10^{-4} \Delta k/\%$
Void Coefficient for Uniform Voids in the Moderator in the Unflooded Operating Control Distribution*	$-2 \times 10^{-4} \Delta k/\%$	---
Flooding:		
Reactivity Change Upon Flooding the Coolant Passages in the Operating Control Distribution at 80 C*	-0.5% Δk	$-0.8 \pm 0.1\% \Delta k$
Maximum Central Bundle Worth:		
Central Fuel Bundle Worth with Central 4 Control Rods Withdrawn and Outer 8 Control Rods Inserted (Core Unflooded)	+5.1% Δk	$+5.3 \pm 0.5\% \Delta k$

*Inner control rods inserted halfway in a bank.

Table I provides a comparison between experiment and theory for several of the key parameters. In general, the multiplication was predicted to $\pm 1\% \Delta k/k$ for all critical loadings. Since the void coefficients and temperature effects change sign under various conditions in the lattice, the agreement between experiment and theory must be considered in the absolute and not the percentage sense. The slight over-prediction of control worth is probably caused by insufficient self-shielding of the unborated stainless steel in the diffusion-theory calculations. In addition to the reactivity parameters of the type illustrated in Table I, detailed power-distribution measurements were performed. The measured power distribution in the horizontal plane is compared to the analytic prediction in the preceding paper.¹ These and other comparisons between experiment and theory which indicated accurate prediction of the doubly heterogeneous lattice led to the acceptance of the physics model described above for the design of the EVESR.

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2. PETERSEN, G. T. and F. G. WARZEK, "AEC Superheat Criticals - A Comparison of Experiment and Theory on Uniform Lattices," GEAP-3882, (January, 1962).
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Reactivity Worth of Transverse Gaps, S. Weinstein and F. Feiner (GE-KAPL).

A series of experiments on the reactivity worth of gaps has been performed in the Solid Homogeneous Assembly (SHA). The objective of the program is to provide data against which calculational models can be checked. An immediate goal is to develop a method by which large void regions can be adequately treated within the framework of diffusion theory. This would enable the nuclear engineer to perform standard design calculations on systems containing such voids.

The experimental procedure consists simply of adding fuel blocks axially to an unreflected SHA core and of determining the halves separation at which the assembly goes critical. The maximum core length attainable is limited by nuclear-safety considerations since ultimately the assembly will be inadequately shut down even with the halves apart.

Two unreflected rectangular critical configurations have been studied: Core 5B has a cross section of 20 in. $\times$ 16 in. and its material length was varied from 24 in. to 34 in.; Core 5BS (5B turned on its side) has a cross section of 20 in. $\times$ 24 in. and its material length was varied from 16 in. to 25½ in. Both cores at their minimum length

Fig. 1. Width of Transverse Gap vs Additional Core Length.

were approximately $\$1$. supercritical so that each series of measurements did not start with zero gap.

The data from the two series are plotted in Figure 1. Also plotted are initial calculations based on the diffusion-theory treatment as presented by Chernick and Kaplan.¹ The data and calculated points are plotted with the abscissa and ordinate appropriately shifted to account for the gap present in the initial length reactor.

Calculations based on other methods presented in Chernick and Kaplan and on the variational method reported by Haggbloom² are currently being performed.

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Effects of Control Blades in Lattices Moderated by D_2O/H_2O Mixtures, R. H. Clark, E. J. DeRoche, G. T. Fairburn, J. W. Hallam, and T. C. Engelder (B&W).

Studies of the nuclear characteristics of D_2O/H_2O -moderated lattices¹ have recently been extended to include measurements of reactivity and flux perturbations. These experiments provided nuclear data on relatively simple configurations for testing theoretical methods. The basic lattices consisted of 1.206-cm-diameter aluminum-clad 2.46%-enriched UO_2 fuel rods on a square pitch of 1.511 cm. Moderator compositions varied from 0 to 72% D_2O to provide a wide range of neutron spectrum. Boric acid was added to keep the core radius constant (61.11 cm).

The perturbations in Table I were introduced by fully inserting the following materials (all 0.090 in. thick) along the core axis: flat blades (10 in. wide) and cruciforms (6 in. wide) of Boral (0.096 gB/cm²), cadmium (0.020 in. thick) clad with stainless steel, and aluminum. The moderator gap was formed by the removal of a partial row of 17 fuel rods, and the gap for the cruciforms was 1 rod thick.

Reactivities were measured two ways to provide a comparison of techniques: by integrating $\partial\rho/\partial h$ vs h measurements between the critical moderator height and a reference moderator height of 145 cm, and by measuring the differential boron worth and the boron concentration required for criticality at the reference height. The

Fig. 1. Perturbed Thermal Flux Distributions (72% D_2O).

TABLE I. Reactivity Worths of Perturbations (Units of $10^{-3}k$)

Configuration No. and Description	Core XVIII - 72% D ₂ O				Core XIX - 50% D ₂ O				Core XX - 0% D ₂ O			
	N _B mg B/l	$\Delta\rho$ ($\partial\rho/\partial h$)	$\Delta\rho$ (boron)	$\Delta\rho$ (calc.)	N _B mg B/l	$\Delta\rho$ ($\partial\rho/\partial h$)	$\Delta\rho$ (boron)	$\Delta\rho$ (calc.)	N _B mg B/l	$\Delta\rho$ ($\partial\rho/\partial h$)	$\Delta\rho$ (boron)	$\Delta\rho$ (calc.)
A - Unperturbed Lattice	85.9	16.7	15.9	17.6	938.6	18.1	18.3	18.0	1838	14.6	14.8	16.7
B - Boral blade in (A)	2.2	0.6	0.6	-	792.1	1.5	1.6	2.0	1705	2.1	2.7	2.0
C - Moderator gap	109.3	21.0	19.8	22.1	940.6	18.5	18.4	18.9	1790	10.4	10.5	11.6
D - Aluminum blade in (C)	104.5	20.0	19.0	19.1	-	-	-	-	-	9.9	-	-
E - Cadmium blade in (C)	24.9	5.9	5.3	5.3	822.5	5.1	5.3	-	1710	2.8	3.2	-
F - Boral blade in (C)	0.9	0.2	0.3	-	783.0	0.4	0.5	-	1686	0.8	1.1	-
G - Boral Cruciform in gap	0.0	0.0	0.0	0.0	779.4	0.0	0.0	0.0	1674	0.0	0.0	0.0
H - Aluminum cruciform in gap	108.4	20.8	19.6	-	937.6	18.2	18.1	-	1786	11.0	10.2	-

N_B is boron concentration in moderator for criticality at 145 cm moderator height.

reactivity worths ($\Delta\rho$) obtained from the two methods, converted to absolute units ($\times 10^{-3}$) using calculated values of β_{eff} , are consistent within their accuracy ($\approx 5\%$, or 2×10^{-3}). The results also agree with two-dimensional PDQ-4 calculations.

Thermal-flux distributions were measured with bare and cadmium-covered gold and dysprosium foils. Thermal fluxes derived from the two detectors agreed, except near the perturbors where the disagreement approached 10% due to spectral effects. Figure 1 shows typical flux distributions at 72% D₂O. Agreement with theoretical calculations was generally satisfactory.

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Results of Initial Experiments on the Cornell Zero-Power-Reactor Cores, H. C. Aderhold, S. S. Berg,* D. D. Clark, E. B. Fehr, and R. A. Shaw (Cornell).

The Cornell Zero Power Reactor cores are generically related to normal water lattices studied earlier at Bettis and Brookhaven.^{1,2,3} The Cornell cores have an hexagonal pattern and nominal water-to-fuel ratios of 1:1, 1.5:1, 2:1, 3:1, and 4:1. The fuel is uranium dioxide, density of 10.38 g/cm³, enriched to 2.094 atom percent. The active column in each fuel rod is 48 inches high and 0.600 inches in diameter. The fuel columns are inside aluminum tubes of 0.666-inch-OD and 0.028-inch wall. Each control rod is constructed as a cluster of 3 fuel-poison control elements. The top half of each control element is boron carbide packed into a 0.666-inch-OD aluminum tube; the bottom half of the control element is a standard fuel-column loading. These control elements may be substituted for fuel rods at any desired locations in the lattice; when they are fully withdrawn, the lattice is completely uniform.

The experimental program included finding the critical buckling from foil traverses, the gold/cadmium ratios in the fuel and the moderator, the manganese/cadmium ratio in the moderator, the peripheral fuel-rod worth, the worth of a water gap left in the reactor by removing the central fuel rod, and the worths of a single control element and of a cluster of three control elements in the center of the reactor. The gold/cadmium ratio in the fuel was measured with 0.002-in. thick gold foils in the center

fuel rod and in the moderator with 0.005 in. foils evenly spaced across the core. The manganese/cadmium ratio in the moderator was found using a cadmium wrapped ribbon of 0.005-in.-thick manganese-copper alloy (P-metal) at a single location near the center of each core. All cadmium covers were 0.020 in. thick.

In analyzing the experiments, the experimental critical bucklings were used to calculate the effective multiplication factors by two computational techniques. The first technique used Ombrellaro's fitted MUFT cross sections⁴ to obtain the group parameters for three fast groups. A spectral distortion factor, derived by Hicks⁵ to adjust the epithermal group cross section of uranium-238, was used. The thermal-group constants were obtained with the use of effective cross sections⁶ calculated with the "r" factors derived⁷ from the gold/cadmium ratio measurements. The effective multiplication factors obtained from this 4-group hand calculation gave satisfactory agreement with experiment.

The second computational technique employed a five-factor method. A uranium-235 resonance escape probability⁸ was calculated; good agreement was found between the overall resonance escape probability calculated by the multigroup method and the product of the multifactor resonance escape probabilities for uranium-235 and -238. The thermal utilization was calculated by averaging the results of both transport⁹ and diffusion-theory calculations. The migration area was found by summing the group ages from the multigroup calculation. The multifactor calculation of the effective multiplication factor was not as accurate as the multigroup calculation. The error may be partially attributed to a low value of the calculated interaction fast fission factor.¹⁰

The values of the reactivity worth of the control elements and control rods were compared with results from a simple two-group calculation. Agreement between experiment and theory was not satisfactory, indicating the need for more sophisticated analysis. No attempt was made to calculate the water-gap reactivity effect.

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TABLE I. Information Summary—Cornell Zero Power Reactor Cores^a

Parameters	1:1 Core	1.5:1 Core	2:1 Core	3:1 Core	4:1 Core	Unit
Critical Mass	19.06±0.05	11.84±0.03	9.49±0.02	8.30±0.02	9.14±0.02	kg U-235
B ²	580±34	806±22	857±23	770±55	616±35	10 ⁻⁵ cm ⁻²
r(Fuel)	0.199	0.154	0.127	0.0981	0.0851	
r(Moderator) ^b	0.224	0.150	0.119	0.086	0.075	
k _{eff} (Multigroup)	0.964	0.970	0.979	0.989	0.991	
k _{eff} (Multifactor)	0.955	0.941	0.954	0.972	0.977	
Peripheral Fuel Rod Worth ^c	563±30	917±7	1126±10	1048±15	866±20	10 ⁻⁶ δ k/k per rod
Worth of Central Control Element						
Experiment	281±15	318±2	371±15	313±6	295±9	10 ⁻⁴ δ k/k
Theory	180	270	290	260	210	
Worth of Central Control Cluster						
Experiment	475±20	530±20	555±20	530±20	500±20	10 ⁻⁴ δ k/k
Theory	340	480	540	510	410	
Worth of Central Water Gap	+228±5	+47±2	-14±3	-515±6	-642±9	10 ⁻⁵ δ k/k

^aExperimental errors are given as estimated probable errors.^bThe observed cadmium ratios were constant out to about 75% of geometrical radius for each core and increased rapidly for larger radii.^cBased on increases in core radius averaging 2.7%.

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Pulsed-Neutron Techniques

Prompt-Neutron Lifetime in Enriched-Uranium Metal Cylinders, J. T. Mihalcz (ORNL).

A series of experiments with unreflected and unmoderated cylinders of enriched-uranium metal (93.15% U-235) has been performed at the ORNL Critical Experiments Facility to determine the dependence of the prompt-neutron lifetime on the cylinder dimensions. Five cylinders ranging in diameter from 17.77 to 38.09 cm and in height-to-diameter ratio from about 0.2 to 0.7 were assembled and their prompt-neutron decay constants measured at delayed critical by the Rossi- α technique¹. Prompt-neutron lifetimes were obtained from the measured decay constants and effective delayed-neutron fractions calculated by transport theory². The average uranium density for each assembly was greater than 18.7 g/cm³.

The experimental procedure consisted in constructing cylinders that were as nearly regular as possible and also as nearly critical as possible. However, the cylinders had irregular top surfaces because of the method of assembly, and in some cases the reactivity was as much as 8.5¢ below delayed critical. To estimate the dimensions of corresponding regular critical cylinders, corrections to the data were applied to compensate for the reflecting effect of the support structure and for the irregularity of the top surface. The former corrections were determined experimentally. The latter were determined by first-order perturbation theory using the adjoint and forward angular fluxes obtained from transport calculations with the TDC code in the S_0 approximation³ and the cross sections described by Hansen and Roach⁴. The resulting dimensions and masses of the regular critical assemblies are given in the table.

For the Rossi- α measurements, two detectors were used to sample leakage neutrons, and the reactivity effect of the detectors plus that of a 50-g aluminum reflector sufficed to make the assemblies critical. In these measurements the time distribution of pulses from detector 2 following a pulse from detector 1 was determined by use

of a Technical Measurement Corporation 256-channel time analyzer equipped with a neutron time-of-flight log unit (Model 211) having a 0.25- μ sec channel width. The initial pulse from detector 1, a proton-recoil-type plastic scintillator (Nuclear Enterprises type NE 102), triggered the time analyzer, and succeeding pulses from detector 2 (a small spiral fission counter) were sorted in time.

The measured decay constant, the calculated β_{eff} value, and the resulting prompt-neutron lifetime for each assembly are given in the table. Within experimental errors the average prompt-neutron lifetime of $(6.25 \pm 0.04) \times 10^{-8}$ sec does not depend on the dimensions of the cylinders and agrees with the value obtained at the Los Alamos Scientific Laboratory⁴ for a uranium sphere (Godiva I).

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Experimental and Calculated Properties of Critical Cylinders of U(93.2) Metal

Diameter (cm)	Corrected Height (cm)	Assembly Density (g of U/cm ³)	Corrected Mass (kg of U)	Correction to Experimental Mass (g of U)	ρ , Prompt-Neutron Decay Constant ^a (10 ⁶ sec ⁻¹)	β_{eff} ^b ($\times 10^3$)	λ , Prompt-Neutron Lifetime ^c (10 ⁻⁸ sec)
17.77	12.63	18.76	58.7	+15	1.082 $\pm$ 0.006	6.79	6.28 $\pm$ 0.03
22.85	9.75	18.73	74.8	-593	1.088 $\pm$ 0.006	6.76 ^d	6.21 $\pm$ 0.03
27.93	8.64	18.71	99.1	-410	1.076 $\pm$ 0.009	6.72	6.25 $\pm$ 0.05
33.01	8.08	18.71	129.4	+1261	1.071 $\pm$ 0.010	6.69 ^d	6.25 $\pm$ 0.06
38.09	7.71	18.70	164.3 ^e	+479	1.063 $\pm$ 0.005	6.67	6.27 $\pm$ 0.03

a. At delayed critical $\rho = \beta_{\text{eff}}/\lambda$; in an unreflected uranium sphere, Godiva I, $\rho = (1.1 \pm 0.02) \times 10^6 \text{ sec}^{-1}$, $\beta_{\text{eff}} = 0.0068$, and $\lambda = 6.2 \pm 0.1 \text{ nsec}$ (Ref. 4).

b. Obtained from S_0 calculations with convergence criterion = 10^{-6} ; experimental and calculated multiplication constants differ by less than 0.005.

c. Error indicated is the standard deviation from a least-squares analysis of the counter data from the Rossi- α measurements.

d. Interpolated value.

e. To be compared with 166.5 $\pm$ 0.6 kg of uranium at a density of 17.9 g of U/cm³ determined at the Los Alamos Scientific Laboratory.

Experimental Reactor Physics I

Neutron-Spectrum Measurements in a (UF₄ + Paraffin) Multiplying Assembly, J. C. Young, J. M. Neill, and J. R. Beyster (GA).

The neutron-moderating properties of polyethylene (CH₂) have been studied experimentally and analytically at General Atomic and other laboratories^{1,2}. Measurements have been made of the neutron spectra with various 1/v absorber concentrations and temperatures as an over-all check of the scattering model for the polyethylene molecule. Since the scattering model for polyethylene should be very similar to that for paraffin, a homogeneous mixture of (UF₄ + paraffin) was obtained from ORNL for quasi-infinite-medium neutron-spectrum measurements. This material has been studied³ extensively in critical experiments.

The geometrical arrangement used for this measurement was a parallelepiped 16 in. by 16 in. by 17 in. with a re-entrant tube in the center. Standard Linac time-of-flight pulsed-source techniques with a 50-meter neutron flight path were utilized. The (UF₄ + paraffin) mixture had a poison concentration of 4.6 barns/hydrogen atom at 2200 meters/sec. neutron energy. No appreciable leakage correction was necessary to calculate the spectrum measured in this geometry. The spectrum was calculated using the Wunderlich⁴ molecular vibration spectrum for polyethylene and a scattering kernel generated from the computer code SUMMIT⁵.

The background of delayed neutrons emanating from the assembly had to be subtracted from the measured flux. To determine the magnitude of this delayed-neutron background (N_d), the complete time behavior of the neutron density was measured at the base of the re-entrant tube with a small fission counter, and the ratio (A) of the total number of delayed neutrons to the total number of prompt neutrons was determined. It can be shown⁶ that the appropriate delayed-neutron background to subtract is given by

$$N_d = \frac{RA}{1+A} \int_0^{1/R} C(t) dt$$

where C(t) is the observed time spectrum and R is the repetition rate of the Linac.

This correction was made to the experimental spectrum shown in Figure 1. For comparison, the experimental data prior to subtracting the delayed-neutron background is also given. The results of the bound-hydrogen and a free-gas calculation are shown in comparison to the experimental data. The agreement between the bound-hydrogen calculation and the measured spectrum appear to be very good, with the neutron-conservation criteria being satisfied to within 2 per cent. From Figure 1, it appears that this scattering kernel for polyethylene is adequate for reproducing the characteristics of the measured neutron spectrum in the (UF₄ + paraffin) mixture. Also, the method for subtracting the delayed-neutron background appears to be quite adequate.

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Fig. 1. Infinite Medium Neutron Spectrum in UF₄ + Paraffin.

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Energy-Dependent Extrapolation-Length Measurements in Polyethylene, R. Fullwood, R. Slovacek, R. Cerbone, and E. Gaertner (RPI).

The extrapolation length (in the diffusion approximation) for polyethylene has been determined directly from asymptotic neutron-spectrum measurements using the pulsed-electron-linac phase-neutron-chopper technique¹. The neutron spectrum is sampled 3.4 mean life times (344 μsec) after the initial burst of neutrons from the linac to insure that the dominant mode is the time asymptotic mode.

Preliminary to these investigations, experiments were performed to study the perturbation on the neutron spectrum due to the reentrant hole used to extract the flux. Slabs of polyethylene 31 cm × 31 cm × 8.02 cm were used in the study with reentrant holes of diameter 3.2 cm to 25.4 cm extending to the center. It was found that the spectrum was remarkably stable against reentrant hole size up to 5.08 cm in agreement with other observers^{2,3}, but in disagreement with the diffusion-theory calculation of Molinari⁴. The time decay constant of the medium is dependent on hole size and agrees to within 10% with a weighted surface-to-volume calculation.

In the experiment to determine the extrapolation-length spectrum measurements were made with a reentrant hole 0.95 cm in diameter at several depths symmetrically spaced about the center in the narrow dimension of the polyethylene slab. Assuming the validity of diffusion theory and space-time separability, the directional flux (scalar and current) in the fundamental spatial mode is

- ... Experimental Data for the Extrapolation Length
- Least Squares fit to Experimental Extrapolation Length Data
- xxx 0.71 of the Experimental Values of Lambda Transport
- 0.71 Lambda Transport as Calculated by L. Esch Using Radkowsky "Prescription."

Fig. 1. Energy Dependence of Extrapolation Length for Polyethylene

$$F(x, E) = F(0, E) \left[\cos \frac{\pi y}{L} + \lambda_{tr} \frac{\pi}{L} \sin \frac{\pi x}{L} \right],$$

therefore
$$\frac{F(-x, E) + F(x, E)}{F(0, E)} = \cos \frac{\pi x}{L}$$

$L = 2(x_m + \epsilon)$, $F(0, E)$ is the flux measured at the center of the slab, x_m is the slab half thickness, and ϵ is the extrapolation distance. Spectrum measurements were made at $\pm 3.70, \pm 3.07, \pm 1.80$ cm and at the center of the slab. A least-squares fit to a cosine function by iteration gives L for each value of neutron energy. The corresponding values of ϵ are given in Figure 1.

It is apparent that values for the transport mean free path $\lambda_{tr}(E)$ may also be obtained in a similar way. The crosses in Figure 1 are 0.71 times the experimental values for $\lambda_{tr}(E)$ obtained from the ratio of surface to center flux using the values of L already determined; namely,

$$\frac{F(x_m, E)}{F(0, E)} = \cos \frac{\pi x_m}{L} + \frac{\pi}{L} \lambda_{tr} \sin \frac{\pi x_m}{L}$$

The solid curve is due to L . Each^s calculated for polyethylene using the Radkowsky prescription.

If the curve ($\lambda_{tr}(E)$) is numerically integrated over a Maxwellian distribution (30 C), a value of the diffusion constant $D_0 = 24800 \pm 3600$ cm²/sec is obtained. This value may be compared with Sjöstrand's⁶ value of 26500 ± 600 cm²/sec measured by a die-away experiment.

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Experimental Reactor Physics II

Homogeneous Critical Experiments with Plutonium/Plastic Mixtures in the Intermediate Energy Range, C. R. Richey, J. D. White, R. C. Lloyd, and E. D. Clayton (GE-HAPO).

Plutonium mixtures in the form of wet powders, precipitates, slurries, polymers, etc. constitute a potential criticality hazard in chemical plants processing irradiated reactor fuels. Critical experiments are being conducted at the Hanford Critical Mass Laboratory to obtain criticality data for plutonium mixtures characteristic of the above forms. Criticality data have thus far been obtained with PuO_2 -polystyrene compacts having a plutonium concentration of 1.12 g/cm^3 2.2% Pu-240 and a hydrogen-to-plutonium atomic ratio of 15. The experimental assemblies were reflected prisms ranging from near-cubic geometry to thin slabs; the principal reflector material was 6-inch-thick Lucite, $\text{C}_5\text{H}_8\text{O}_2$. Criticality was also obtained for a non-reflected array of near-cubic geometry.

The fuel compacts were in the form of 2.01-in. cubes. Each cube was spray-painted with one-mil-thick coat of aluminum paint and 11-mil-thick rubberized plastic. The approach-to-critical technique was used to assemble the critical arrays utilizing a remotely operated split-table device¹. A summary of the experimental data is given in Table I. Also given are the criticality data for an equivalent Pu/water mixture fully reflected by water and having an H/Pu atomic ratio of 15 and a plutonium density of 1.62 g/cm^3 . The conversion factors were obtained from a 12-group diffusion-theory calculation.

The critical dimensions of the various experimental arrays may be reproduced using a constant buckling of

TABLE I. Critical Experiments with PuO_2 /Polystyrene Compacts (Pu Density = 1.12 g/cm^3 , H/Pu = 15, Pu-240 = 2.2%)

Base Dimensions		PuO_2 /Polystyrene Arrays (Lucite Reflector)		Equivalent Pu/ H_2O Systems (H_2O Reflected) ^a		
Length (Inches)	Width (Inches)	Height (Inches)	Mass (Kg Pu)	λ_{eff} (cm)	Height (Inches)	Mass (Kg Pu)
20.35	18.16	3.56 ± 0.02	24.16 ± 0.14	7.07	2.88	28.28
16.28	15.14	4.07 ± 0.02	18.41 ± 0.09	7.04	3.18	20.82
12.21	12.21	5.17 ± 0.01	14.14 ± 0.03	6.93	3.74	14.80
10.18	10.18	6.47 ± 0.02	12.30 ± 0.04	6.90	4.39	12.07
9.16	9.16	7.79 ± 0.02	12.00 ± 0.04	6.81	4.93	10.97
8.14	8.14	9.71 ± 0.02	11.81 ± 0.02	6.87	5.60	9.85
12.21 ^b	12.21 ^b	12.93 ± 0.06^b	34.67 ± 0.32^b	2.2 ^b	--	--

^a Pu concentration is 1.62 g/cm^3 ; H/Pu atomic ratio is 15.

^b Non-reflected array.

Fig. 1. Reflector Savings as a Function of Lucite Thickness.

$233.44 \times 10^{-4} \text{ cm}^{-2}$ and the effective extrapolation lengths recorded in Table I. It is of interest to note the dependence of λ_{eff} on the base dimensions. A least-squares analysis of this variation leads to an infinite slab thickness of $6.24 \pm .09 \text{ cm}$ with $\lambda_{\text{eff}} = 7.16 \pm .09 \text{ cm}$ when reflected by Lucite and a non-reflected slab thickness of $16.16 \pm 20 \text{ cm}$ with $\lambda_{\text{eff}} = 2.2 \text{ cm}$. The corresponding infinite slab thickness for the Pu/water mixture (infinite water reflected) is 5.56 cm . This slab thickness is approximately 60 percent larger than previously reported for a Pu-239/water mixture².

The variation of the reflector savings with Lucite thickness is given in Figure 1. Note that the reflector savings for six inches of Lucite is nearly as effective as that for an infinitely thick layer. Inserting a 30-mil cadmium sheet between the fuel core and reflector reduced the reflector savings to an equivalent $1.0 \pm .2$ -in. of Lucite. Although the density of hydrogen is less in Lucite than in water, the reflector savings for eight inches of Lucite was approximately 20 percent larger than that for eight inches of water. This difference was expected since the higher concentration of oxygen and carbon in Lucite reduces the leakage for higher-energy neutrons ($E > 0.01 \text{ MeV}$) from the arrays.

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Lattice Calculations and Measurements

Preliminary Analysis of Some $U^{233}O_2$ -ThO₂-H₂O Lattices, H. C. Honeck and B. R. Sehgal (BNL)

A preliminary theoretical analysis of a series of exponential experiments with U^{233} -Th oxide rods in water has been completed. The fuel rods are 0.0345 in. thick Zircalloy tubes with an O.D. of 0.499 in. filled with vibratory compacted Thorium-Uranium-oxide powder. The fuel contains 3.0% of $U^{233}O_2$ by weight and has an average density of 8.96 gm/cm³. The fuel also contains about 9 parts/10⁶ of boron as an impurity and trace amounts of U^{234} , U^{235} , and U^{238} .

The theoretical methods have been previously reported¹ and involve the computer codes ZUT and TUZ (resonance capture rates), THERMOS (thermal-neutron reaction rates), and MUFT (epithermal smooth reaction rates). Homogenization of the lattice was by flux-volume weighting in the fast and thermal-energy regions, and by volume weighting in the intermediate energy region.

The cross-section data are from the Westinghouse MUFT-4 library using decks 450 for U^{233} and 350 for thorium. Resonance parameters for thorium are from a recent BNL Sigma Center compilation² with a $\Gamma_\gamma = 30$ mV. The 2200 m/sec value of η is taken as 2.286.

The results are given in Table I. It is apparent that the eigenvalues are high by about 1% and show a trend with V_m/V_f . The epithermal α used here was 0.172 and the total fission integral was 831 barns. Both values are based on measurements by Halperin et al.³. A more recent measurement by Bigham⁴ gives a value of 743 barns. We have reduced the fission integral to 760 barns ($\alpha = 0.188$) which fall within the uncertainty of both the fission integral and α . The corrected eigenvalues are given in Table I and are about unity for the dilute lattices but some 2-3% low for the tight-packed lattices.

Reflector savings were computed and are shown in Fig. 1. Experimental values are also shown. The discrepancy at large V_m/V_f is not understood.

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Fig. 1. U-Th Lattices: Reflector Savings at $B_L^2 = 0$.

TABLE I
Results of the Th- U^{233} Oxide Rod Analysis

V_m/V_f^a	ϵ^b	η	I	U^{233} R_a^c	U^{233} R_f^c	Th 232 R_a^c	Computed B_L^2 (m ⁻²)	Measured B_L^2 (m ⁻²)	λ^d	Corrected λ
0.997	1.0101	1.5930	0.9433	404	321	9.92	82.30	-	-	-
1.384	1.0079	1.5913	0.9237	411	326	10.45	92.42	91.14	1.004	0.977
1.713	1.0067	1.5904	0.9074	415	329	10.73	96.29	92.29	1.012	0.988
2.194	1.0053	1.5896	0.8839	418	331	11.02	97.94	94.21	1.010	0.990
3.004	1.0040	1.5887	0.8456	422	334	11.30	92.75	86.75	1.016	1.000
4.272	1.0027	1.5881	0.7904	425	336	11.56	76.47	70.07	1.017	1.005
6.845	1.0012	1.5874	0.6951	426	337	11.69	37.01	-	-	-
9.276	1.0005	1.5871	0.6222	426	338	11.71	1.57	-1.61	1.010	1.002

^aVolume of moderator/volume of fuel.

^bFast effect to 0.825 MeV.

^cHeterogeneous resonance integral for absorption (R_a) and fission (R_f).

^dEigenvalue or k_{eff} .

Analysis of BNL $\text{UO}_2\text{-H}_2\text{O}$ Lattice Experiments, B. R. Sehgal (BNL)

At Brookhaven a series of 20 $\text{UO}_2\text{-H}_2\text{O}$ exponential-lattice experiments has been performed. The UO_2 fuel is 3% U^{235} enriched and has a density of 9.57 g/cm^3 . The particulars of the experiments are reported in a review paper¹. Measurements were made for critical buckling, reflector savings, δ^{28} , the ratio of U^{235} fissions to U^{238} fissions and ρ^{28} , the ratio of epicadmium to subcadmium absorptions in U^{238} .

A theoretical analysis of these experiments was performed. The methods are in general the same as previously reported². Briefly, the Dancoff corrections are calculated with the Monte Carlo code KITTY HAWK, the resonance integrals are obtained from ZUT and TUZ codes; the epithermal reaction rates are calculated with the MUFT code and the thermal-reaction rates are obtained through the THERMOS code. The homogenization of the lattice is by flux-volume weighting in the fast and thermal-neutron groups and by volume weighting in the intermediate energy range. Improvements in the calculational methods include a precise treatment³ of the stainless-steel clad in the calculation of the Dancoff corrections and the resonance integrals and a multigroup collision-probability calculation of the fast fission effect.

Analysis of the reflector savings was done with the FOG code, utilizing the 4-group averaged cross sections obtained by the above procedure.

Results of the calculations for the five clean lattices are presented in Table I. It is seen that the calculations and experiments are in excellent agreement for the values

TABLE II
A Comparison of the Calculated and Measured
Values of δ^{28} and ρ^{28}

V_m/V_f	δ^{28}		ρ^{28}	
	Calculated	Measured	Calculated	Measured
1.319	0.0648	0.0646 ± 0.0026	2.973	2.92 ± 0.09
1.632	0.0561	0.0562 ± 0.0022	2.463	2.41 ± 0.04
2.091	0.0476	0.0480 ± 0.0019	1.977	1.81 ± 0.06
2.863	0.0387	0.0373 ± 0.0014	1.497	1.41 ± 0.02
4.071	0.0312	0.0307 ± 0.0012	1.120	1.04 ± 0.02

of critical buckling and k_{eff} . There is also no evident trend with moderator-to-fuel volume ratio. There is, however, some discrepancy between the calculated and the measured values of the reflector savings. I have not yet accounted for this discrepancy.

In Table II a comparison of the calculated and measured values of δ^{28} and ρ^{28} is presented. The values of δ^{28} are in excellent agreement but those of ρ^{28} are about 2-6% too high. This may be because of a too-low value chosen for the Cd cutoff energy.

The calculations for the poisoned lattices were also performed and they are also in excellent agreement with the measurements, except for the reflector savings.

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TABLE I
Results for the Five Clean Lattices

V_m/V_f	ϵ	ρ	η	f	Critical Buckling		Reflector Savings at $B_0^2 = 0$		k_{eff}
					Calculated (m^{-2})	Measured (m^{-2})	Calculated (cm^{-2})	Measured (cm^{-2})	
1.319	1.0407	0.6641	1.8256	0.8007	54.77	55.23	7.14	7.29	1.00155
1.632	1.0347	0.7094	1.8274	0.7881	63.41	63.17	6.85	6.74	0.99923
2.091	1.0288	0.7577	1.8291	0.7698	69.18	68.90	6.53	6.71	0.99917
2.863	1.0226	0.8106	1.8310	0.7397	68.22	68.99	6.27	6.67	1.00216
4.071	1.0176	0.8577	1.8325	0.6952	61.30	60.89	5.85	6.30	0.99883

Critical Masses and Bucklings of $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Systems, L. C. Schmid, R. C. Liikala, W. P. Stinson, and J. R. Worden (GE-HAPO)

The results of subcritical experiments and analytical correlations are described for $\text{PuO}_2\text{-UO}_2$ rods moderated with H_2O . The correlation is the first reported with this type of fuel, and the results are important as a check on the analytical techniques used in calculating such systems.

The rods, 0.372 ± 0.004 in. diam and 48.50 ± 0.07 in. long, were loaded at spacings of 0.55, 0.60, 0.71, 0.80, 0.90, or 0.93 in. in hexagonal lattices which were completely reflected with H_2O . The rods contain vibrationally-compacted 1.50 wt% PuO_2 ($\approx 24 \mu\text{m}$) in UO_2 and are clad with 0.027 ± 0.002 in. thick Zircaloy-2. Each rod contains an average of 826.6 g of $\text{PuO}_2\text{-UO}_2$ distributed with $< 2\%$ variation in density along the length of the rod. Both the uranium and plutonium are $88 \pm 2\%$ of the density of the corresponding oxide. The uranium is depleted to 0.22% U^{235} and the plutonium contains 91.41% Pu^{239} , 7.83% Pu^{240} , 0.73% Pu^{241} , and 0.03% Pu^{242} by weight. The top and bot-

tom end caps contain 7.6 g and 12.8 g of Zircaloy-2, respectively. A 1.9-g ZrO_2 disc and a 6.3-g stainless-steel compression spring is contained in that order in a 5-3/4-in. gap between the top of the core and the top end cap.

The number of rods for a critical loading was determined by the extrapolation of neutron multiplication data obtained with as much as 96% of the critical mass. Count rates were also obtained along the central axis of each loading when 50% of the critical mass was loaded. From these data the material buckling was computed and equated to the geometrical buckling of the critical loading. A value of the reflector savings was calculated, from the equality, for each lattice with the assumption that the savings are equal in the vertical and horizontal directions. The results are summarized in Table I for a moderator temperature of 25°C .

The numbers of rods required for the critical loadings are compared in Table I to the numbers obtained from a one-dimensional diffusion-theory calculation¹ assuming a four-energy-group homogeneous-cell model. The cross sections for the two upper-energy groups (10 MeV to 11.7 keV and 11.7 keV to 2.38 eV) were obtained from GAM-1². The cell-averaged absorption and fission cross sections for the two low-energy groups labeled 3 and 4 (2.38 to

TABLE I
Analytical and Experimental Results of $\text{PuO}_2\text{-UO}_2$ Rods in Light Water

Lattice Spacing (in.)	H/Pu Atom Ratio	No. of Rods for Critical		Reflector Savings (cm)	Buckling (m^{-2})
		Calculated	Experimental		
0.55	230	1522	1487 ± 3	8.40 ± 0.06	48.0 ± 0.2
0.60	326	844	829 ± 7	7.99 ± 0.10	65.1 ± 0.8
0.71	567	492	484 ± 1	7.27 ± 0.03	78.5 ± 0.3
0.80	794	435	420 ± 1	6.93 ± 0.03	74.9 ± 0.3
0.90	1077	474	452 ± 2	6.75 ± 0.03	60.9 ± 0.2
0.93	1169	509	488 ± 2	6.64 ± 0.05	55.2 ± 0.2

0.683 eV and 0.683 to 0.0 eV) were obtained from THERBAR³. The free-gas scattering model for hydrogen was employed in group 3 and the Nelkin scattering model in group 4. The transport and transfer cross sections for groups 3 and 4 were obtained from GAM-1 and TEMPEST⁴ and weighted by the corresponding moderator-to-cell flux ratio obtained from THERMOS⁵. All programs use the R.B.U. cross-section library^{5,6}. The thermal cross sections used for Pu^{239} are those recommended by Sher⁷. The methods used in these calculations result in critical masses which are between 2 and 5% high. This corresponds to effective multiplications which are about 0.5% low. When lumping the PuO_2 into 24 micron particles is considered, the calculated value for the multiplication at a lattice spacing of 0.71 inches is lowered by 0.2%.

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Reactor Methods

Monte Carlo Calculations for Enriched Uranium Metal Assemblies, D. C. Irving and John T. Mihalcz (ORNL)

The capability of the general geometry subroutine GEOM as used in the 05R Code¹ for handling complicated three-dimensional geometries exactly has been tested in a series of calculations of unmoderated and unreflected critical assemblies of 93.2% U²³⁵-enriched uranium metal ($\rho = 18.69 \text{ g/cm}^3$). Descriptions of the assemblies are given in Fig. 1 and Table I. The upper section of each assembly was built on a 30-in. diam stainless-steel diaphragm, and the lower section was supported by a low-mass frame that could be raised remotely.

GEOM will treat all geometries in which material interfaces are describable by functions that are at most quadratic (e.g., planes, cylinders, cones, spheres, etc.). The quadratic material boundaries may intersect each other and be oriented in any possible manner. The com-

plexity of the geometries that can be handled is limited only by the amount of computer storage available for input data.

Relatively good efficiency in the handling of detailed geometries is achieved by successively breaking the geometry up into simpler pieces. The entire system, enclosed in a rectangular parallelepiped, is divided into an array of smaller parallelepipeds, which are likewise subdivided into an array of still smaller parallelepipeds, called blocks. The quadric surfaces describing the material interfaces divide each block into volumes called sectors which contain a single medium. In treating a flight path which originates in a certain sector, GEOM need consider only those surfaces which bound that sector. Once a boundary is crossed, then the new sector is determined and a new set of boundaries is considered.

Multiplication constants calculated with GEOM for the experimental assemblies are shown in Table I. Although the calculated values are 1.4 to 2.1% larger than those

Table I. Critical Masses and Calculated Multiplication Constants^a of
93.2 wt % U²³⁵-Enriched Uranium Metal

Assembly				Calculated Multiplication Constants ^b	CDC-1604 Computational Time (min)
Upper Section	Lower Section	Height (in.)	Critical Mass (kg of U)
Same as upper section.	3.98	147.6	1.021	65
Same as upper section.	5.11	166.8	1.014	64
6.37, annulus 3.00, No. 1 4.40, No. 2	215.4	1.020	64
Eight-unit ring and centerpiece (see Fig. 1) ^c			206.5	1.017	82

^aA similar calculation for Godiva I² required 58 min and yielded a multiplication constant of 1.015.

^bNumber of histories, 20 000; standard deviation, ± 0.007 . The neutron cross sections used in the calculations are those given in BNL-325,³ with an energy variation of $\nu^{235} (E \text{ MeV}) = 2.423 + 0.088E + 0.0088E^2$ (Ref. 4). The angular distributions for elastic scattering were taken from Smith⁵ for energies below 1.5 MeV and from ANL-400⁶ for higher energies. The total inelastic cross sections were those of Smith⁷ below 1.5 MeV, those of Howerton⁸ from 1.5 to 9.0 MeV, and those obtained from optical model calculations⁹ of direct neutron interactions above 9.0 MeV. The angular distribution of neutrons inelastically scattered from the low energy levels which contribute at high energies to the measurement of $\sigma_{el}(0)$ was assumed to be the same as that for elastic scattering. Inelastic scattering was treated by evaporation theory except where level excitation cross sections were available. For U²³⁸ the level excitation cross sections were those measured by Smith up to 1.5 MeV; for U²³⁵ they were those calculated by Parker¹⁰ and modified slightly by optical-model results.

^cThe running time for GEOM to treat random paths in this geometry was 26.8 msec per path compared to 11 msec for a single sphere.

Fig. 1. Critical 93.2% U^{235} -Enriched Uranium Metal Assembly with Eight-Unit Upper Section and an Irregularly Shaped Centerpiece (Inset). The unit at the top of the photograph is an approximate parallelepiped whose base is 5 by 5 in. and whose height varies in three steps (5.14, 5.27, and 4.39 in., front to back); the opposite unit is a parallelepiped with a 3 by 5 in. base and a 3.51-in. height topped by a 3.60-in. diam by 1.70-in. cylinder; the four small cylinders are 3.59-in. diam and 5.11-in. high; and the two large cylinders are 4.5-in. diam and 5.3-in. high. The centerpiece, which penetrated the hole in the support diaphragm, consists of a 4.5-in. diam by 1.11 in. cylinder topped by a parallelepiped with a 5.00 by 5.00-in. base and a 2.25-in. height and by a hemisphere with a 2.39-in. radius.

measured, they are well within the error introduced by the $\pm 5\%$ uncertainty in σ_f^{235} in the MeV range (for these assemblies $\Delta\sigma_f^{235}/\Delta k = 0.8$).

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Critical and Exponential Measurements

Material Bucklings of Uranium-Carbide-Fueled Graphite Assemblies - Measurements,
O. R. Hillig and D. W. Latham (AI)

Critical and subcritical experiments, in support of the Advanced Sodium Graphite Reactor (ASGR) program¹, have been performed to determine values of material buckling for graphite assemblies fueled with slightly-enriched (3.02 wt%), uranium carbide-fuel elements. These experiments provide the first data available for comparison with results of calculations.

A fuel element is shown in Fig. 1. Lattice arrangements have been studied which include control-channel mockups for triangular-lattice pitches of 9, 11 and 12 in. The following arrangements were studied:

- 1) 'Closed Lattice' - all channels filled with fuel elements;
- 2) 'Open Lattice with Sodium' - every fourth channel containing a sodium-filled stainless-steel tube, others filled with UC-fuel elements;
- 3) 'Open Lattice with SS' - every fourth channel containing an empty, stainless-steel tube, others filled with UC-fuel elements;
- 4) 'Open Lattice with Graphite' - every fourth channel containing a graphite plug, others filled with UC-fuel elements.

Arrangements 2 and 3 mockup the 'wet' (sodium-filled process channels) and 'dry' (empty process channels) control-rod concepts. Arrangement 4 was studied to check the validity of calculational techniques for unpoisoned, non-uniformly-loaded assemblies.

Measurements were carried out in subcritical assemblies of varying sizes (assemblies containing 19, 23 or 31 channels) to study two effects: 1) diffusion anisotropy and 2) the finite assembly-size dependence. The diffusion anisotropy was caused by void streaming and assembly heterogeneities. The size dependence was due to the inability to measure the radial component of buckling. Instead, an assumed extrapolation distance was used. Subcritical measurements were also made using 2 wt% U-metal, hollow-cylinder fuel elements. Assemblies with this fuel loading had a relatively small amount of diffusion anisotropy, thereby providing information relating to the dependence of the radial buckling upon assembly size. After these two effects are separated, the measured results may be corrected. The theoretical interpretation of

Fig. 1. Uranium Carbide Fuel Element.

TABLE I
 B_m^2 - Uncorrected Subcritical and
Inferred from Critical

Lattice	Material Buckling (m^{-2}) ^a			
	31 channel	23 channel	19 channel	Inferred from Critical
9-in.				
'Closed,' UC Fuel	---	5.66 ± 0.05	6.78 ± 0.03	4.14
'Closed,' U-Metal Fuel	---	8.52 ± 0.04	9.10 ± 0.04	7.53
'Open with Sodium'	4.08 ± 0.04	4.76 ± 0.04	---	3.28
'Open with SS'	5.41 ± 0.03	6.42 ± 0.01	7.84 ± 0.02	4.09
'Open with Graphite'	5.98 ± 0.02	6.41 ± 0.02	---	5.57
11-in.				
'Closed,' UC Fuel	---	6.34 ± 0.02	---	5.74
'Closed,' U-Metal Fuel	---	7.16 ± 0.03	---	6.93
'Open with Sodium'	4.04 ± 0.06	4.06 ± 0.03	---	3.75
'Open with SS'	5.18 ± 0.05	5.38 ± 0.02	---	4.55
'Open with Graphite'	5.75 ± 0.06	5.85 ± 0.01	---	5.59
12-in.				
'Closed,' UC Fuel	---	6.14 ± 0.03	---	5.92
'Closed,' U-Metal Fuel	---	6.46 ± 0.01	---	6.30
'Open with Sodium'	3.79 ± 0.02	3.81 ± 0.05	---	3.55
'Open with SS'	4.83 ± 0.02	4.85 ± 0.02	---	4.41
'Open with Graphite'	---	5.50 ± 0.02	---	5.24

^aErrors associated with the error in the measurement of the axial component of buckling only.

the measured results to provide these corrected values of material buckling is presented in a companion paper².

As a basis for comparison, material-buckling values were inferred from two-region critical mass measurements using four-group theory. Two-region critical mass measurements were carried out because only a limited number of UC-fuel elements (24) were available for critical mass studies. The dependence of the uncorrected measured values upon size and diffusion anisotropy can be seen in Table I, which also includes the inferred from critical values.

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Material Bucklings of Uranium-Carbide-Fueled Graphite Assemblies - Theoretical Interpretation, Eugene R. Specht (AI)

Material bucklings for lattices fueled with uranium carbide enriched to 3 wt% U^{235} were obtained from critical and subcritical experiments¹. When the buckling measurements from the subcritical assemblies were corrected for finite-size effects on the radial buckling and diffusion anisotropy, good agreement was obtained with material bucklings from critical assemblies. These measurements appear to be the first experimental verification of the validity of D. S. Selengut's formulae² for diffusion anisotropy, which arises because of assembly heterogeneities when no voids are present. These formulae are used on assemblies which have dissimilar but finite diffusion coefficients. Several experiments have been

conducted^{3,4} by others which indicate the existence of diffusion anisotropy in assemblies without voids, but no theoretical verification of these measurements has been demonstrated.

The radial bucklings for the subcritical measurements¹ were calculated using an effective assembly diameter and a constant extrapolation length. As noted by others⁵, this procedure is inadequate for small subcriticals. Corrections were obtained for the radial bucklings of the UC subcriticals by performing critical and subcritical experiments with U-metal-fueled lattices which had negligible diffusion anisotropy. The radial buckling corrections for the UC subcriticals were assumed to have the same size dependency as had been obtained from the uranium metal data.

Some of the experiments had large voided channels (4 in. I.D.) which mocked up control-rod locations. The diffusion anisotropy because of these voids was quite high (about 12%). Good correlation between the critical and subcritical measurements was obtained for different assembly sizes by using Behrens' theory for voided channels^{6,7}.

The subcritical-material bucklings were first modified by the radial-buckling corrections obtained from the uranium-metal experiments. Next, the subcritical-material bucklings were corrected for the effects of diffusion anisotropy by employing the following equation:

$$\Delta B_m^2 \text{ (subcritical)} = \frac{M_z^2 - M_r^2}{M_z^2} \left[\frac{B_m^2 \text{ (critical)}}{2} - B_r^2 \text{ (subcritical)} \right],$$

where the values of B_m^2 (critical) had been obtained for a configuration in which $B_r^2 = B_z^2$.

Finite-size corrections (to the subcritical values of B_r^2) and diffusion-anisotropy corrections were applied to all of the configurations studied¹ (lattice pitches of 9 in., 11 in., and 12 in. and 4 types of mockup control cells). Subcriticals also were studied which contained various numbers of cells (19, 23, and 31). Table I shows the data for the assemblies which contained sodium-filled SS channels (4 in. O.D.) in every fourth cell (the other three cells contained UC-fuel elements) and for the assemblies with empty SS tubes in every fourth cell.

As seen from this table the amount of diffusion anisotropy in these lattices is substantial even when no voids are present. The radial-buckling corrections and diffusion-anisotropy corrections have reduced the differences between the critical and subcritical experiments to a level which is about the same as the estimated error in the material-buckling measurements.

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TABLE I

Corrected and Uncorrected Material Bucklings for Critical and Subcritical Assemblies

Configuration	$\frac{M_z^2 - M_r^2}{M_z^2}$	B_m^2 Critical	B_m^2 Uncorrected Subcritical	B_r^2 Correction for Subcritical	Anisotropy Correction	B_m^2 Corrected Subcritical
<u>31 cell Subcriticals with Na Control Cells</u>						
9 in. pitch	0.065	3.28	4.08	-0.35	-0.68	3.05
11 in. pitch	0.049	3.75	4.06	-0.11	-0.30	3.65
12 in. pitch	0.042	3.55	3.81	-0.07	-0.22	3.52
<u>23 cell Subcriticals with Na Control Cells</u>						
9 in. pitch	0.065	3.28	4.76	-0.89	-0.95	2.92
11 in. pitch	0.049	3.75	4.06	-0.23	-0.44	3.39
12 in. pitch	0.042	3.55	3.81	-0.14	-0.30	3.37
<u>31 cell Subcriticals with void in Control Cells</u>						
9 in. pitch	0.113	4.09	5.41	-0.35	-1.14	3.92
11 in. pitch	0.082	4.55	5.16	-0.11	-0.47	4.58
12 in. pitch	0.070	4.40	4.84	-0.07	-0.33	4.44
<u>23 cell Subcriticals with void in Control Cells</u>						
9 in. pitch	0.113	4.09	6.42	-0.89	-1.60	3.93
11 in. pitch	0.082	4.55	5.38	-0.23	-0.71	4.44
12 in. pitch	0.070	4.40	4.85	-0.14	-0.47	4.24
<u>19 cell Subcritical with void in Control Cells</u>						
9 in. pitch	0.113	4.09	7.84	-1.44	-2.09	4.31

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Critical Dimensions of Aqueous UO_2F_2 Solutions Containing 4.9% Uranium-235-Enriched Uranium, E. B. Johnson and D. F. Cronin,* (ORNL)

The critical dimensions of aqueous solutions of $\text{U}(4.9)\text{O}_2\text{F}_2$ ($\text{U}(4.9)$ designates uranium enriched to 4.9% in U^{235}) have been measured as a part of the Oak Ridge

*Now at National Lead Company of Ohio, Cincinnati.

National Laboratory program of critical experiments providing data useful in authenticating nuclear reactor calculations and in establishing specifications for chemical processes and for transport and storage of fissile materials. Many earlier results from this program are included in Ref. 1.

The high solubility of uranyl fluoride (≈ 890 g of uranium per liter) allowed experiments with solutions having

Fig. 1. Critical Planar Arrays of Cylindrical Units of $\text{U}(4.9)\text{O}_2\text{F}_2$ Aqueous Solution at an H^{235}U of 490.

TABLE I
Critical Dimensions of $\text{U}(4.9)\text{O}_2\text{F}_2^a$ Aqueous Solutions in Single Units

Solution Concentration		Container	Cylindrical Critical Dimensions			
H/U ²³⁵	g of Uranium per Liter	(0.16-cm-thick wall)	Diameter (cm)	Height (cm)	Volume (liters)	U ²³⁵ Mass (kg)
Water-Reflected Units						
495	890	Aluminum	38.1	41.7	47.6	2.11
		Aluminum	33.0	72.4	62.1	2.75
		Aluminum	31.1	139.6	106.0	4.70
		Aluminum	30.7	173.2	128.1	5.67
524	870	Stainless Steel	50.8	29.3	59.4	2.53
		Aluminum	38.1	44.8	51.1	2.17
		Stainless Steel	38.1	50.4	57.4	2.44
		Stainless Steel	30.5	>153	>111	>4.73
643	728	Aluminum	76.2	23.9	109.2	3.89
		Stainless Steel	50.8	34.7	70.3	2.50
		Stainless Steel	38.1	75.5	86.1	3.06
735	650	Aluminum	76.2	24.2	110.4	3.50
		Stainless Steel	50.8	40.1	81.1	2.58
		Stainless Steel	38.1	153.0	174.4	5.54
994	496	Aluminum	76.2	37.9	173.1	4.20
		Stainless Steel	50.8	85.7	173.7	4.21
1099	452	Aluminum	69.3 ^b		170.5	3.77
Unreflected Units						
524	870	Stainless Steel	50.8	38.7	78.3	3.33
		Stainless Steel	38.1	>147	>168	>7.13
643	728	Aluminum	76.2	28.9	132	4.69
		Stainless Steel	50.8	45.7	92.6	3.30
		Stainless Steel	38.1	>166	>189	>6.73
735	650	Aluminum	76.2	31.5	144	4.58
		Stainless Steel	50.8	54.3	110	3.50
994	496	Aluminum	76.1	44.8	204	4.96
		Stainless Steel	50.8	>140	>283	>6.84
1002	492	Aluminum	69.3 ^c	-	172	4.14

^a $\text{U}(4.9)$ designates uranium enriched to 4.9 wt% in U^{235} .

^b This was a spherical container 98% filled when critical.

^c This was a spherical container 99% filled when critical.

an H/U^{235} atomic ratio as low as 500, approximately that for minimum critical mass and near that for minimum critical volume at this enrichment. The solution was contained in spherical or cylindrical aluminum or stainless steel vessels.

The critical dimensions of a number of single units, both unreflected and surrounded by an effectively infinite water reflector, are summarized in Table I.

In another series of experiments, as many as 25 cylinders of solution 24.1 cm in diameter have been arranged in planar arrays with the individual units in a square lattice. The number of units in a critical array as a function of the separation of their axes is shown in Fig. 1. The arrays were located sufficiently far from massive structural members so as to be considered unreflected and contained no moderator except the solvent itself. The results may be compared with those from similar experiments with a solution of highly enriched uranium².

From the results of these experiments it is possible to

establish some limiting dimensions useful in process designs. The following configurations will be subcritical: a water-reflected cylinder of $U(4.9)O_2F_2$ aqueous solution 30.5 cm in diameter, provided its height to diameter ratio (h/d) is less than 7; an unreflected cylinder 38 cm in diameter, provided h/d is less than 5; three 24.1-cm-diam cylinders 142.5 cm high, separated 25 cm between centers in a planar array, triangular pattern, reflected by water on one face; a linear array of nine 24.1-cm-diam cylinders 61 cm high, spaced 25 cm on centers and reflected on one side by water; and four of these cylinders in a planar array, square pattern, unreflected and separated 25 cm.

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Reactor Calculations—I

Effect of Distribution of Fissile Material on Critical Mass, H. K. Clark, (du Pont-SRL)

Experiments¹ have established the minimum critical masses of fissile nuclides in homogeneous aqueous solutions. Goertzel² has shown, however, that with an optimum distribution of fissile material, lower masses are critical. The present work calculates minimum critical masses of U-235, Pu-239, and U-233 as a function of distribution.

Five concentration distributions (Fig. 1) of fissile material within a water-reflected sphere were chosen, together with Goertzel's optimum distribution for U-235 (dashed curve). The sphere was subdivided into five regions having equal ratios of surface-to-volume within which the average relative concentration was calculated by integrating the concentration curve. (The shading in Fig. 1 shows the resulting discrete concentrations for distribution V.) Two-group diffusion-theory calculations were performed to obtain the average critical concentration and hence the critical mass for spheres of various sizes. The results are given in Table I and for the Goertzel distribution may be compared with his calculated minimum critical mass of 690 g at a radius of 16.7 cm. Subdividing into four regions or into five regions having equal volumes was found not to affect the results appreciably.

The two-group parameters for each concentration were obtained by normalizing two-group diffusion-theory parameters obtained in multigroup-transport calculations³ to experiment. The normalization consisted principally of adjusting the number of secondary neutrons-per-fast-interaction remaining in the fast group so that the critical size of a water-reflected sphere, calculated with the adjusted parameters, agreed with experiment⁴. Calculations made with the adjusted parameters for an experiment⁴ in which a non-uniform loading of a cylinder subdivided into five regions and reflected laterally by water had a critical mass of 1066 g U-235 compared with 1163 g for a uniform loading gave respectively 1082 g and 1163 g, the error corresponding to only 0.002 in k_{eff} .

Fig. 1. Concentration Distributions.

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TABLE I
Minimum Critical Masses as Function of Distribution
of Fissile Nuclide in Water

Distribution	Nuclide					
	U-235		Pu-239		U-233	
	Mass, g	Rad, cm	Mass, g	Rad, cm	Mass, g	Rad, cm
Uniform	815	14.9	512	15.5	598	13.6
I	770	15.9	492	16.6	575	14.6
II	768	16.7	493	17.4	571	15.2
III	768	17.7	495	18.4	572	16.0
IV	772	18.1	504	19.0	578	16.7
V	803	21.3	524	22.3	600	19.3
Goertzel	764	16.7				

The Criticality of Plutonium-238, Eric H. Ottewitte, Vahe Keshishian (AI)

Plutonium-238 promises to be an excellent radioisotope heat source. The question of criticality arises for large quantities of Pu-238.

The criticality of Pu-238 was calculated with the one-dimensional DTF-II Code¹, embodying Carlson's S_4 transport-theory approximation. Two compositions were evaluated, 1) 100% pure Pu-238, 2) 80.94% Pu-238, 15.26% Pu-239, 2.88% Pu-240, 0.79% Pu-241, and 0.118% Pu-242. The latter is the composition of the material available from Savannah River.

Sixteen-group cross sections were prepared for Pu-238, using the Hansen-Roach energy-group structure². Neutron fission and capture-group constants from 10 MeV down to 550 eV were based on the Hanford cross-section library³. These cross sections do not include the recent limited experimental data⁴; but they are the only relatively complete published set of cross sections. Below 550 eV, the fission and capture cross sections were based on Pu-240 cross sections normalized to Pu-238 cross sections at thermal energies⁵. Plutonium-240 scattering matrix was used for all energy groups². The Hansen-Roach sixteen-group library provided the group constants for

Fig. 1. Critical Mass of Unreflected Plutonium Sphere.

Pu-239 and Pu-240 (Ref. 2). The cross sections of the Pu-241 and Pu-242 isotopes were taken as those of Pu-239 and Pu-240, respectively.

The calculations showed that Pu-238 systems are about as fast in neutron energy (the median fission energy is about 1.3 MeV) as those of Pu-239. Figure 1 presents the calculated critical masses of bare Pu-238 spheres. These results agree well with Hanford Monte Carlo calculations⁶ which used the Butler cross-section measurements⁴. In addition, subcritical and critical material-replacement experiments, and fission cross-section measurements have been performed on the Jezebel assembly at LASL. Their results indicate that the reactivity worth of Pu-238 is at least as much as that of Pu-239 (Ref. 7). Figure 1 shows that the Pu-238 critical mass is roughly one-half that for Pu-239 (Ref. 8).

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One-Velocity Monte Carlo Calculations of Uranium-Metal Critical Geometries, J. T. Mihalcz (ORNL)

In an earlier paper¹ it was shown that geometrically complicated critical configurations²⁻⁴ of 93.2% U-235-enriched uranium metal ($\rho = 18.75 \text{ g/cm}^3$) can be analyzed by a Monte Carlo method in which the geometry is handled exactly. In the calculations reported the neutron energy was treated in as much detail as the cross-section information allowed. In this paper it is shown that when only multiplication constants are to be calculated, a simpler one-velocity treatment that assumes isotropic scattering is adequate.

The one-velocity calculations require as input a collision cross section (Σ_t) and a neutron production per collision. Both were determined from solutions of the one-velocity transport equation for two delayed-critical experiments: a 7-in.-diam (17.78 cm) by 4.971-in.-high (12.63 cm) cylinder of the U(93) metal and a 15-in. (38.1 cm) diam by 3.03-in.-high (7.69 cm) cylinder. The solutions, which were obtained from S_{12} transport calculations with the DDK code⁷, using a 16×16 grid of space points, yielded values of $\nu\Sigma_f/\Sigma_t$ as a function of Σ_t . The pair of values common to both experiments was then used as input to the Monte Carlo calculations for all the geometries.

This Monte Carlo method⁸ calculates the fission distribution from a batch of 400 neutrons uniformly distributed in space. The resulting distribution is then assigned to the succeeding batch of 400 neutrons, and a new distribution is calculated. This procedure was repeated for a total of 30 batches. The multiplication constant was then obtained both by calculating the ratio of the number of neutrons produced to the number of source neutrons in each batch and averaging over the batches (method 1); and by dividing the assembly into a number of regions, computing the probability that a neutron born in each region will produce a fission in any other region, forming a matrix of these probabilities, and calculating the multiplication constant which is the largest eigenvalue of this matrix (method 2). In the batch method (1), the matrix of fission probabilities was used to determine when the effects of the assumed initial source distribution had disappeared. Only subsequent batches were used in computing the multiplication constant.

The results are given in Table I. The batch-method multiplication constants vary from 0.971 to 1.028, reflecting deviations from experiment that are either statistical or result from the use of the single pair of one-velocity constants for all geometries. If the deviations are statistical, the average value should be 1.000. It is actually 0.9993 ± 0.016 , with 70% of the results in the range 1.000 ± 0.014 . This good agreement indicates that the one-velocity method is reliable for predicting the multiplication constant of a single material in an unreflected, complicated geometry.

TABLE I
One-Velocity Multiplication Constants for Uranium Metal Critical Geometries

Geometry	Multiplication Constant ^a	
	Method 1 ^b	Method 2
Cylinder, ^c 38.10-cm diam x 7.65-cm high	1.028	1.023
Two Coaxial Cylinders ^c		
Each 38.10-cm diam x 6.04-cm high, flat faces 12.27 cm apart	1.002	
Each 17.78-cm diam x 7.31-cm high, flat faces 0.86 cm apart	0.986	
Cylindrical annulus, ^c 38.10-cm OD, 22.86-cm ID, 14.98-cm high	1.020	1.022
	1.014 ^d	
Parallelepiped, 12.70 x 12.70 x 23.19 cm	0.996	0.994
Two Parallelepipeds ^e		
Each 20.32 x 25.40 x 7.94 cm, large flat faces 12.45 cm apart	1.014	1.008
Each 20.32 x 25.40 x 5.08 cm, large flat faces 0.97 cm apart	0.997	0.994
Cylinder and Cylindrical Annulus, Each 10.11-cm high ^f	1.007	0.999

Annulus: 38.1-cm OD; 27.94-cm ID
Cylinder: 17.78-cm diam

Cylindrical Annulus and Parallelepiped, Each 12.98 cm high

1.011 1.010

Annulus: 38.10-cm OD; 27.94-cm ID
Parallelepiped: 12.70-cm side length

Cylindrical Annulus and Two Parallelepipeds^f

1.012 1.008

Annulus (two sections combined):
38.10-cm OD, 27.94-cm ID, 12.98-cm high

Upper Section

Parallelepipeds: 12.70-cm side length;
No. 1, 7.62-cm high; No. 2, 11.18-cm high

Lower Section

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TABLE I (cont.)

Geometry				Multiplication Constant ^a	
				Method 1 ^b	Method 2
Three-Dimensional Arrays of Cylindrical Units ^c					
Array	Unit Diameter (cm)	Unit Height (cm)	Surface Separation ^d (cm)	Unit Mass (kg U ⁻¹)	
2 x 2 x 2	11.49	8.08	0.90	15.7	0.975
	11.51	10.77	2.25	21.9	0.997
2 x 3 x 3	11.51	5.38	2.01	10.5	1.004
	11.48	10.77	4.34	20.9	0.995
4 x 4 x 4	11.51	5.38	3.95	10.5	0.971
4 x 4 x 1	11.49	10.77	1.52	20.9	0.993
2 x 4 x 2	11.49	10.77	3.89	20.9	0.997
2 x 2 x 2	Each unit a 11.45-cm-diam x 5.38-cm-high cylinder between two 9.10-cm-diam x 4.32-cm-high cylinders			15.7	1.010
2 x 2 x 2	Each unit a 9.12-cm-diam x 4.32-cm-high cylinder between two 11.49-cm-diam x 3.09-cm-high cylinders			20.9	0.977
Eight units of various shapes arranged in a circle and an irregularly shaped composite. ^e				1.022	1.017
				0.997 ^d	

^a12 000 neutron histories in 30 batches; $k = 0.2271 \text{ cm}^{-1}$, $\sigma = k_p/k = 0.2974$.

^bStandard deviations vary from ± 0.009 to ± 0.014 ; the average of the batch multiplication constants for all the different geometries is 0.9993 ± 0.016 ; the average of the matrix multiplication constants is 0.9992.

^cSee Ref. 3.

^dRepeat of calculation with different initial random number.

^eSee Ref. 4.

^fSee Ref. 1.

^gSee Ref. 6.

^hEqual in these dimensions.

Measurement of Reactor Parameters

Experimental Thermal-Disadvantage Factors in Some Very Undermoderated, 3%-Enriched Water-Uranium Oxide Lattices, Allen R. Boynton (ANL)

As a part of the High Conversion Critical Experiment (Hi-C)^{1,2} at Argonne National Laboratory, the thermal disadvantage factor (ζ) has been measured in five lattices that had water-to-uranium oxide-volume ratios of from 0.96 down to 0.16. The data obtained in these cores indicate that ζ does not decrease monotonically with decreasing water-to-uranium oxide volume ratio but possesses a minimum and increases as the lattice pitch decreases beyond a certain value.

The fuel in these cores was 3.04 wt% enriched uranium in the form of 10.17 g/cm³ UO₂ pellets of 0.935-cm diameter loaded to 122-cm length in 1.057-cm OD cladding tubes of 0.0496-cm-thick stainless steel. Both square and triangular lattice pitches were used.

The disadvantage factor was obtained by the so-called integral technique. Foils of a uranium-aluminum alloy of 17.5 wt% uranium that was enriched to 92.75 wt% U-235 were shaped to measure the U-235 fission rate in representative portions of the fuel and moderator volumes of a unit cell. Cadmium ratios in both fuel and moderator sections were determined with 0.051-cm cadmium covers. Various thicknesses (0.0051 to 0.066 cm) of foils were used. The foil activities were corrected for thermal self-shielding. In the cadmium-covered irradiations, various masses of cadmium were used and the cadmium ratios were corrected to zero cadmium mass.

The results of these measurements are given in Table I. Also given are the results of two computations. First are 27 energy-group Nelkin-kernel THERMOS results (the upper energy is 0.415 eV). The unit cells used had an extra scattering region to approximate an isotropic return at the boundary. Both neutron density and U-235 fission ζ 's are given. The disadvantage factor was also calculated using a collision probability technique³ and the

flat-flux approximation. The collision probabilities were computed by the B692/RP code⁴ which considered a three-region cell (fuel, clad and moderator). The cross sections used for this one-group calculation were the averaged ones obtained from the THERMOS outputs. This computation results in flux ζ 's.

Two disagreements are noted in the comparison of the calculations with the experimental results. First, except for the tightest lattice, THERMOS values are larger than experimental values. Honeck⁵ has observed this trend for water-uranium oxide lattices with water-to-fuel volume ratios greater than 1. The second difficulty is that while the experimental results indicate a minimum of ζ at a water-to-uranium oxide volume ratio of about 0.5, calculations show ζ to be monotonically decreasing with decreasing lattice pitch.

It is not clear whether these discrepancies between experiment and theory result from errors of experimental or computational origin. The small-cell problem is a severe test of calculational techniques; it is an equally severe test of experimental techniques.

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TABLE I
Thermal Disadvantage Factors and U-235 Fission-to-Cadmium Ratios in High Conversion Critical Experiment Lattices

Lattice		Experimental Results				Calculated ζ		
Pitch (cm)	Water-to-Oxide Volume Ratio	Fuel-to-Cadmium Ratio	Moderator to-Cadmium Ratio	Bare-Moderator to-Bare-Fuel Ratios	ζ	Thermos Number Density	Thermos U-235 Fission	B692/RP Flux
1.24 (square)	0.962	4.49 $\pm$ 0.11	5.19 $\pm$ 0.15	1.118 $\pm$ 0.009	1.160 $\pm$ 0.015	1.238	1.248	1.163
1.27 (triangular)	0.757	3.75 $\pm$ 0.08	4.14 $\pm$ 0.10	1.075 $\pm$ 0.009	1.112 $\pm$ 0.015	1.225	1.234	1.154
1.166 (triangular)	0.437	2.55 $\pm$ 0.06	2.72 $\pm$ 0.08	1.054 $\pm$ 0.004	1.096 $\pm$ 0.026	1.190	1.198	1.140
1.127 (triangular)	0.324	2.15 $\pm$ 0.05	2.33 $\pm$ 0.08	1.050 $\pm$ 0.008	1.120 $\pm$ 0.038	1.186	1.193	1.136
1.069 (triangular)	0.164	1.59 $\pm$ 0.03	1.69 $\pm$ 0.09	1.081 $\pm$ 0.041	1.189 $\pm$ 0.109	1.157	1.162	1.133

A Measurement of the Fast Disadvantage Factor in a Light-Water-Moderated, Slightly Enriched Uranium Lattice, Henry E. Bliss, (MIT), Glenn A. Price (BNL)

A measurement of the fast disadvantage factor, which will be defined as the ratio of the average activation in the moderator to the average activation in the fuel, is of interest in studies of heterogeneous systems, because it provides a means for direct comparison between theory and experiment. The value of the fast disadvantage factor reported here will be denoted by the quantity ζ_f^{oz} , where the superscript 'oz' designates the material being activated (thorium in this measurement) and the subscript 'f' designates the activity being detected (fission-product activity). In the past, values of the fast disadvantage factor have been inferred from measurements of the U-238 to U-235 fission ratio (δ^{28}), and it is only recently that attempts have been made to measure the fast disadvantage factor directly¹. Previous direct measurements have utilized intra-cell fast-flux traverses to obtain average activations, whereas the present value was obtained from activations of foils having the same shape and size as the unit cell. Hence, the present value should have a significantly smaller statistical uncertainty than previously reported values.

In the case of slightly enriched uranium water-moderated systems, high-energy effects, e.g. fast fission, are commonly treated by a volume-weighted homogenization of the lattice with corrections for heterogeneity subsequently applied. Both multigroup and Monte Carlo calculations² indicate that heterogeneous effects may be adequately described by a one-group treatment of fast neutrons above the U-238 fission threshold. Such a treatment yields the following representation for the fast disadvantage factor:

$$\zeta_f^{oz} = \left[1 + \left(\frac{V_1}{V_0} \right) \beta \Sigma_1 \ell_0 \right]^{-1}, \quad (1)$$

where

$\left(\frac{V_1}{V_0} \right)$ is the moderator-to-fuel volume ratio

β is a numerical factor related to the chord-length distribution in fuel and moderator for which the constant value 0.45 is quite accurate in typical light-water systems

Σ_1 is the moderator removal and absorption cross section

ℓ_0 is the average chord length of the fuel rod.

To test the validity of Eq. (1), a series of experiments was performed in a lattice of 0.600-in. (1.52-cm) diameter, 1.3%-enriched uranium rods immersed in water with a moderator-to-fuel volume ratio of 4:1. The average

activation in the fuel was obtained from circular Th-232 foils while that in the moderator was obtained from triangular-shaped thorium-sector foils. The foils were placed in a central three-rod cluster at identical axial locations and irradiated in the BNL Miniature Lattice Facility. After irradiation the foils were integral gamma counted to obtain the induced fission-product activity.

The value of ζ_f^{oz} derived from these experiments is given in Table I along with two measurements performed by Klein³. Also listed are the theoretical values calculated from Eq. (1).

The close agreement between experimental and theoretical values in Table I appears to substantiate the claim that heterogeneous fast effects may be adequately described on the basis of a one-group theory.

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Experimental Lattice Parameters for 28-Rod Clusters of Natural UO₂ in D₂O,* R. E. Green, K. J. Serdula, C. B. Bigham (AECL)

Reactor-physics parameters have been measured for D₂O-moderated, natural-UO₂ lattices similar to those planned for the 500-MWe CANDU-type reactors at Pickering, Ontario¹.

The fuel assemblies studied were composed of 28-pin fuel bundles mounted in two concentric tubes simulating the pressure and calandria tubes of the power reactor. Each fuel pin consisted of a 47.7-cm length of sintered UO₂ pellets (1.42-cm diam) of density 10.45 g/cm³ sheathed in Zircaloy-2 (1.52-cm OD x 0.45-mm wall). Symmetric rings of 4, 8 and 16 pins on diameters of 2.33, 5.30 and 8.41 cm respectively formed the 28-pin bundle. Aluminum mounting plates and Zircaloy end caps increased the over-all bundle length to 49.7 cm. Five bundles mounted in a 65S-Al 'pressure tube' (12.78-cm OD x 2.96-mm wall) and a 50S-Al 'calandria tube' (12.74-cm OD x 1.39-mm wall) formed a fuel assembly.

The following parameters were determined in the ZED-2 critical facility² using 55 fuel assemblies in triangular arrays from 24 to 40 cm in pitch with both D₂O and air in the coolant space:

- 1) Buckling was derived from flux distributions measured with Mn (10% Ni) foils at cell-boundary positions. An improved two-group method of analysis was used for the radial distribution³.
- 2) Values of the epithermal spectrum index, r , and the effective neutron temperature, T_n , were deduced from In-to-Mn and Lu-177-to-Mn activation ratios using the technique of Chidley et al.⁴ modified to allow for the variation of the g -value of In with neutron temperature.
- 3) Moderator-to-fuel neutron-density ratios were determined by irradiating Cu foils between fuel pellets and Mn-Ni wires in the moderator. Corrections were made for resonance activation using the measured spectrum parameters and also for the macroscopic flux variation across the cell.

TABLE I
Values of the Fast Disadvantage Factor

V_1/V_0	Fuel-Rod Diameter	Detector Material	Exptl Value	Theor Value
4:1	0.600 in. (1.52 cm)	Th-232 Fission Product	0.780 ± 0.006	0.769
4:1	0.387 in. (0.983 cm)	Al-27 Na-24	0.84 ^a	0.84
4:1	0.387 in. (0.983 cm)	U-238 Fission Product	0.83 ^a	0.84

* See Ref. 3.

- 4) Fast-fission ratio, δ , (number of U-238 fissions per U-235 fission) was measured using natural UO_2 and depleted uranium-metal foils. The fission rates in a representative cross section of the fuel were compared to the rates in a fast-neutron facility with a known fast-fission ratio, as determined with a back-to-back fission chamber.
- 5) Initial conversion ratio, γ_0 (number of Pu-239 atoms produced per U-235 atom destroyed) was determined by irradiating 0.010-cm thick wafers of UO_2 in representative fuel pins and in a pure thermal flux (a tank of D_2O attached to a thermal column of the NRX reactor). The neptunium was counted using a coincidence method as described by Tunncliffe et al.⁵.

The results are summarized in Table I. Bucklings have been corrected to moderator conditions of 99.72 at% D_2O and 25°C using calculated purity and temperature coefficients. The nonsystematic errors in the bucklings are given in the table. We estimate the errors in the other quantities to be $\pm 3\%$ in r , $\pm 3^\circ\text{C}$ in ΔT , $\pm 3\%$ in δ , $\pm 0.5\%$ in γ_0 and $\pm 1.5\%$ in n_m/n_f .

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TABLE I
Lattice Parameters for 28-Rod Clusters of Natural UO_2 in D_2O

Pitch cm	B^2 m^{-2}	r_f (Footnote a)	r_s	r_{cb}	ΔT_f °C (Footnote b)	ΔT_s °C	ΔT_{cb} °C	γ_0	δ	n_m/n_f
D_2O Coolant										
24	2.858 ± 0.034	0.0715	0.0558	0.0459	61	45	25	0.9118	0.0580	1.895
26	3.570 ± 0.026	-	-	-	-	-	-	-	-	-
28	3.826 ± 0.023	0.0522	0.0396	0.0320	55	30	16	0.8339	0.0582	2.049
30	3.830 ± 0.013	-	-	-	-	-	-	-	-	-
32	3.731 ± 0.014	0.0403	0.0295	0.0229	51	20	7	0.7949	0.0554	2.180
34	3.528 ± 0.008	-	-	-	-	-	-	-	-	-
36	3.299 ± 0.005	-	-	-	-	-	-	-	-	-
40	2.774 ± 0.004	0.0277	0.0203	0.0127	48	18	3	0.7623	0.0545	2.430
Air Coolant										
24	2.965 ± 0.036	0.0798	0.0614	0.0508	69	47	29	0.8964	0.0691	1.859
26	3.765 ± 0.026	-	-	-	-	-	-	-	-	-
28	4.033 ± 0.021	0.0568	0.0450	0.0322	55	40	10	0.8075	0.0632	1.990
30	4.059 ± 0.012	-	-	-	-	-	-	-	-	-
32	3.967 ± 0.011	0.0443	0.0335	0.0245	52	28	7	0.7548	0.0619	2.121
34	3.803 ± 0.008	-	-	-	-	-	-	-	-	-
36	3.607 ± 0.015	-	-	-	-	-	-	-	-	-
40	3.096 ± 0.005	0.0284	0.0222	0.0128	50	26	1	0.7143	0.0624	2.356

^a Subscripts f, s, m, cb refer to fuel average, surface of calandria tube, moderator average and cell boundary (halfway between neighboring rods), respectively.

^b $\Delta T = T_n - T_p$, i.e. the difference between the neutron temperature and the physical temperature.

Critical Experiments

A Comparison of Analysis with Experiment for the Saxton Plutonium Critical Experiments/W. L. Orr, H. I. Sternberg (W-APD), P. Deramix (Belgo-Nucleaire & W-APD)

The purpose of the Saxton Plutonium Program is to develop information concerning the use of plutonium-enriched fuel in pressurized-water reactor systems through the design, fabrication and operation of a partial core of $\text{PuO}_2\text{-UO}_2$ fuel in the Saxton Reactor. As a part of this program, a series of design-oriented critical experiments was carried out at the Westinghouse Reactor Evaluation Center using UO_2 fuels enriched to 5.74 wt% U-235 and $\text{PuO}_2\text{-UO}_2$ fuels containing 6.6 wt% PuO_2 in natural UO_2 .

The following parameters of particular importance to the Saxton design were investigated in separate single-region criticals with both plutonium and uranium fuels and in multi-zone criticals consisting of distinct plutonium and uranium regions:

- 1) criticality
- 2) power peaking effects
- 3) power sharing between the two fuels
- 4) control-rod and soluble-poison worth
- 5) kinetic characteristics.

A supporting program of analysis was carried out and detailed comparisons of analysis with experiment are available in each of these areas. The following statements briefly summarize some of the results in three of the areas of interest:

- 1) The measured buckling for five critical lattices with the $\text{PuO}_2\text{-UO}_2$ fuel and two critical lattices with the UO_2 fuel were used in the LEOPARD¹ code with cross sections reported by Westcott² at the 1964 Geneva Conference. The calculated values of k_{eff} are listed in Table I for comparison with an experimental k_{eff} of unity. Fuel-rod specifications are also contained in Table I. Additional comparisons for variations in the method of analysis and in the cross sections used are included in a separate paper³.

Fig. 1. Comparison of Calculated and Measured Power Distributions for a Two-Region Assembly.

- 2) Power peaking effects near water slots were investigated in single-region cores and at the fuel interface in multi-region cores. Since conventional methods of analysis, LEOPARD-PDQ-3, resulted in

TABLE I

Calculated k_{eff} for the WREC $\text{PuO}_2\text{-UO}_2$ and UO_2 Critical Experiments Using the LEOPARD Code and the 1964 Geneva Conference Westcott Cross Sections

Fuel	Lattice Pitch (in.)	Measured Buckling (cm ⁻²) × 10 ³	Calculated k _{eff}				
PuO ₂ -UO ₂	0.52 ^a	10.88 ± 0.13	0.9890				
PuO ₂ -UO ₂	0.56	12.15 ± 0.08	1.0103				
PuO ₂ -UO ₂	0.56 (337 parts/10 ⁶ Boron)	11.23 ± 0.10	1.0148				
PuO ₂ -UO ₂	0.735	15.96 ± 0.19	1.0128				
PuO ₂ -UO ₂	0.792	15.93 ± 0.22	1.0175				
PuO ₂ -UO ₂	1.040	12.84 ± 0.14	1.0167				
UO ₂	0.56	12.71 ± 0.14	1.0059				
UO ₂	0.792	13.68 ± 0.19	1.0021				
Specifications of Fuel Rods Used in Buckling Measurements							
Fuel				Clad			
Material	OD, (in.)	Height, (in.)	Weight, g/rod	Enrichment	Material	OD, (in.)	ID, (in.)
PuO ₂ -UO ₂	0.3374	36.6	546.6	6.6 Wt% PuO ₂ ^b	Zircaloy-4	0.391	0.3445
UO ₂	0.357	36.6	604.3	5.74 Wt% U-235	304 SS	0.391	0.361

^a Data being evaluated
^b Isotopic Composition

Pu-239 = 90.43 at%
 Pu-240 = 8.60 at%
 Pu-241 = 0.92 at%
 Pu-242 = 0.05 at%

an overprediction of power peaking for both fuels, the pre-critical calculated power-peaking factors for the Saxton design are conservative by approximately 4%.

- 3) The measurement of relative power by fuel-rod gamma scan in cores composed of different types of fuel is subject to error when the gamma source and decay characteristics of the two fuels are different. Temperature measurements⁴ in thermally insulated rods of both fuels and foil irradiation experiments using foils of both fuel types were used to determine time-dependent factors to relate the measured gamma activity after shutdown to rod power. These factors were then used to determine the power sharing in the multi-region cores. Comparisons of the calculated and measured power sharing as illustrated in Fig. 1 show a discrepancy of approximately 5% using factors based on the heat-rate experiments and approximately 1% using the foil factors.

As a result of the experimental information obtained in the WREC critical experiments, only small revisions in the expected reactivity and lifetime predictions for the Saxton design are necessary and no adverse effects on the performance of the design core are expected as a result of the small differences in calculated and measured power.

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Neutronic Isolation Characteristics of Concrete, Lead, Wood, Polyethylene and Beryllium / Thomas G. McCreless (USAEC), David R. Smith, George A. Jarvis (LASL), Dick Duffey (U of Md.)

Critical-approach experiments have been conducted to determine the neutron isolating effectiveness of concrete, lead, wood, polyethylene and beryllium. The first three are widely used in the construction of storage vaults and shipping containers. Polyethylene has moderating features similar to water but has the advantage of a solid form to facilitate handling. Beryllium, which is a well-known reactor moderating material with excellent reflective characteristics, may provide a basis for comparison with neutron physics measurements.

The Los Alamos Comet assembly machine was used to determine the change in critical thickness of a reference slab of enriched (93.2 wt% U-235) uranium with and without an additional enriched-uranium slab beyond an isolator 8-in. thick. This change in critical thickness is a

measure of the neutron interaction between the uranium units. The lesser the effective neutron interaction, the better the isolator.

The critical thickness of enriched-uranium slabs as reflected on one side by different thicknesses of each of the isolating materials is shown in Fig. 1.

Fig. 1. Composite Plot of Critical Thickness of Uranium Slabs vs Thickness of Reflectors.

The reflector saving is used to determine the fraction of critical slab thickness, F , of the additional uranium slab placed beyond the isolator. These values and the measured interaction effects are given in Table I. The difference between the interaction effects of the isolating material 8-in. thick and of an 8-in. air space, both with the same value of F , is defined as the isolator saving. The ratio of this saving to the interaction effect of air,

TABLE I
Measured Interaction Effects for Values of Fraction of Critical Slab Thickness (F)

Isolator	T_M^a (in.)	R.S. ^b (in.)	$\frac{T_M + R.S.}{T_{UC}^c}$	Interaction Effect (in.)
Concrete	0.945	1.011	0.695	0.059
	1.063	1.011	0.737	0.070
Lead	0.827	0.867	0.602	0.018
	0.945	0.867	0.644	0.030
Wood	1.063	0.867	0.686	0.041
	0.945	0.847	0.637	0.099
Polyethylene	1.063	0.847	0.679	0.116
	0.945	0.927	0.666	0.001
Beryllium	0.472	1.490	0.697	0.051
	0.709	1.490	0.782	0.072
Air	0.945	1.490	0.866	0.089
	0.945	0.276	0.434	0.124
	1.417	0.276	0.602	0.182
	1.535	0.276	0.644	0.193
	1.654	0.276	0.686	0.216
	1.772	0.276	0.728	0.244
	1.890	0.276	0.770	0.287

^a T_M = measured thickness of additional uranium slab placed beyond isolator.

^b R.S. = reflector saving as provided by the isolator to the additional uranium slab.

^c T_{UC} = critical thickness of unreflected enriched-uranium slab (2.813-in.)

expressed as a percentage, is the relative isolating ability of the material. For $F = 0.675$, the relative isolating abilities are: polyethylene, 99.6%; lead, 82%; beryllium, 79%; concrete, 74%; and wood, 45%.

While these percentages are not constant for all values of F , the general trend remains unchanged. For example, For $F = 0.7$, the relative isolating abilities are: lead, 80.4%; beryllium, 76.8%; and concrete, 72.8%.

Neutron Interaction Between Multiplying Media Separated by Various Materials/ J. D. White, C. R. Richey (BNW)

The quantity of material necessary to effectively isolate multiplying media is a primary concern to the safe and efficient design of shipping containers and the storage regulations for fissionable isotopes. In a series of critical experiments conducted at the Battelle-Northwest Critical Mass Laboratory, the effective isolation thickness was determined for several materials. Those materials tested were inserted at the interface of two multiplying media that comprised the fuel core of a critical assembly. The critical length of the fuel core was then determined as a function of test-material thickness, i.e. the effective isolation thickness being defined as that thickness for which the critical core length approaches an asymptotic value. Figure 1 shows the resulting curves for polyethylene, borated polyethylene (10 wt% B), and compressed wood ($\rho = 1.341 \text{ g/cm}^3$).

The critical assemblies, in the form of rectangular prisms, were assembled by a remote split-table device¹. Plutonium dioxide-polystyrene compacts (1.12 g Pu/cm^3 , Pu-240 = 2.2%, and H/Pu atomic ratio = 15) in 2-in. cubes were the basic fuel material. These were alternated with 2-in. cubes of Plexiglas ($\text{C}_5\text{H}_8\text{O}_2$) to form a three-dimensional checkerboard array of fuel and Plexiglas. The fuel core had cross-sectional dimensions of $12.16 \times 12.06 \text{ in.}$ While the length of one section of the fuel core remained fixed at 2.03 in., the other section located on the opposite side of the test interface was varied to achieve criticality.

Fig. 1. Variation of Critical Size with Isolator Thickness for Polyethylene, Borated Polyethylene and Compressed Wood.

It is evident from the results summarized in Table I that a considerable reduction in the effective isolation thickness is obtained by the addition of highly neutron-absorbing isotopes. Another advantage to using the highly absorbing isolators is evident from Fig. 1; the critical mass of the isolated unit is considerably increased, permitting larger quantities of fissile materials to be stored in the array.

1. RICHEY, C. R., et al., Hazards Summary Report for the Hanford Plutonium Critical Mass Laboratory, Supplement No. 1, "The Remote Split-Table Machine," HW-66266 Sup. 1 Rev. (October 1963).

TABLE I

Material	Effective Isolation Thickness (in.)
Polyethylene	6.5 ± 0.5
Polyethylene ^a	3.5 ± 0.25
Polyethylene ^b	3.0 ± 0.25
Borated Polyethylene	3.5 ± 0.25
Compressed Wood	8.0 ± 0.5
Concrete	9.0 ± 1.0
Borated Concrete	5.0 ± 0.5
Lead	8.0 ± 0.5

^a 0.02-in. cadmium sheet inserted between variable core and polyethylene.

^b 0.02-in. cadmium sheet inserted at both core-polyethylene surfaces.

Critical Uranium-Metal Annuli Reflected with Polyethylene and Graphite / John T. Mihalcz (ORNL)

The previously reported work¹ with annular cylinders of uranium metal ($\rho = 18.76 \text{ g/cm}^3$) enriched to 93.2% in U-235 has been extended to include the effects of reflecting materials. Annuli with outside diameters as large as 15 in. and inside diameters as small as 7 in. were assembled to delayed critical with the central cavity filled with HLM-type graphite ($\rho = 1.73 \text{ g/cm}^3$), beryllium ($\rho = 1.83 \text{ g/cm}^3$), or Plexiglas ($\text{C}_5\text{H}_8\text{O}_2$, $\rho = 1.18 \text{ g/cm}^3$). In some cases the outer surfaces of the annuli were reflected with up to 18 in. of graphite or at least 5.75 in. of polyethylene (CH_2 , $\rho = 0.92 \text{ g/cm}^3$), both with and without a reflecting material in the center of the annuli. Some of the results for the graphite- and polyethylene-reflected experiments are shown in Fig. 1.

Multiplication constants of some of the unreflected and graphite-reflected experiments were calculated by multi-group transport-theory methods using the S_n method in the S_2 approximation and the 16-group cross sections of Hansen and Roach² for the U-235 and U-238 isotopes. Fission of U-234 and U-236 was included and the absorption and scattering of these isotopes were assumed to be the same as those of U-238. Hansen and Roach cross sections were used for the cross sections of the reflecting materials². The TDC and DDK codes^{3,4} plus a modifica-

tion of the DDK code⁵ were used in the calculations. Table I gives the results of some of the experiments and the calculated multiplication factors. The agreement between calculations and experiments is good and verifies that the S_n transport-theory methods, together with the Hansen and Roach 16-group cross sections, are adequate for the calculation of carbon-reflected uranium metal annuli.

1. MIHALCZO, J. T., *Nucl. Sci. Eng.* 20, 60 (1964).

Fig. 1. Critical Mass of Cylindrical Annuli of U(93.2) Metal Reflected by Polyethylene and Graphite.

2. HANSEN, G. E., and W. H. ROACH, "Six and Sixteen-Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543 (1961).
3. CARLSON, B. G., C. E. LEE, and J. WORLTON, "The DSN and TDC Neutron Transport Code," LAMS-2346 (1960).
4. CARLSON, B. G., and J. WORLTON, private communication.
5. WHITESIDES, E., private communication.

Recent Exponential and Critical-Approach Experiments with Slightly Enriched Uranium Metal Tubes in Light Water / C. L. Brown, R. C. Lloyd (BNW)

Nearly all of the criticality measurements performed to date on reactor fuels in light water have been on solid fuel rods; very few data are available on fuel tubes. Because of this paucity of data, calculational methods for estimating critical parameters for fuel tubes are difficult to verify. The experiments described below extend the data in this area.

Exponential and critical-approach experiments were performed on several slightly enriched uranium metal tubes and tube-in-tube combinations in light water. Exponential measurements were performed on 1) 1.002 wt% U-235 enriched uranium tubes; 2) 1.002 wt% U-235-enriched tubes containing an inner tube of 1.002 wt% U-235; 3) 1.25 wt% U-235-enriched tubes, and 4) 1.25 wt% U-235-enriched tubes containing an inner tube of 0.95 wt% U-235. Critical-approach measurements were performed on 1.95 wt% enriched-uranium tubes. The dimensions of the tubes and the results of the experiments are presented in Tables I and II.

In the exponential measurements, extrapolation lengths were determined by measuring relaxation length in two assemblies of different diameter. The fuel columns were 42-in. long for the 1.002 wt% fuel and 52.4-in. long for the 0.95 wt% and 1.25 wt% fuels, being comprised of two fuel elements per column. In each case, material bucklings were measured for hexagonal lattices of 2.1, 2.8, 3.1, and 3.4 in. For the 1.95 wt% U-235 fuel tubes, critical-approach experiments were performed on cylindrical arrays of fuel columns of two different heights (27.3 in. and 40.3 in.). Hexagonal lattices of 2.8, 3.1, and 3.4 in. were measured at each height. The number of tubes for criticality was determined from extrapolation of the inverse multiplication curves for each of the lattices, with the array containing from 90 to 95% of the critical number of tubes.

The maximum bucklings determined from these experiments are as follows:

1.002 Wt% U-235 Assemblies:	$2300 \times 10^{-6} \text{ cm}^{-2}$
1.002 Wt% U-235 Outer Tubes:	$2500 \times 10^{-6} \text{ cm}^{-2}$
0.95/1.25 Wt% U-235 Assemblies:	$3900 \times 10^{-6} \text{ cm}^{-2}$
1.25 Wt% U-235 Outer Tubes:	$4700 \times 10^{-6} \text{ cm}^{-2}$
1.95 Wt% U-235 Tubes:	$8450 \times 10^{-6} \text{ cm}^{-2}$

TABLE I
Reflected and Moderated Uranium Metal Annuli

OD (in.)	ID (in.)	Carbon Reflector Thickness (in.)	Material in Core	Height of Fuel (in.) for Radial Increments of				Uranium Density (g/cm ³)	Reactivity (cents)	Calculated k_{eff}
				3.5-4.5 in.	4.5-5.5 in.	5.5-6.5 in.	6.5-7.5 in.			
13	7	—	C	5.252	5.255	5.268	—	18.705	+15.6	1.0013 ^a
15	7	—	C	4.131	4.072	4.074	4.134	18.746	-24.7	0.9963 ^b
15	7	—	Be	4.008	4.013	3.949	3.947	18.749	-6.3	1.0035 ^b
15	9	—	C	—	5.317	5.328	5.391	18.752	-11.5	0.9980 ^b
15	9	—	Plexiglas	—	5.076	5.085	5.075	18.693	+8.8	1.0023 ^c
13	9	1	C	—	—	6.139	6.120	18.714	-25.0	1.0035 ^b
15	11	3	Air	—	—	4.330	4.332	18.686	-28.6	1.0047 ^c
11	7	7	Air	2.791	2.755	—	—	18.702	+3.0	0.9975 ^d
11	7	7	C	2.697	2.726	—	—	18.695	-4.2	0.9989 ^d
15	7	10	Air	1.188	1.219	1.188	1.219	18.636	-13.0	1.0042 ^d
9	7	13	Air	5.219	—	—	—	18.759	+24.0	1.0024 ^d

^a As indicated by the subheadings, the uranium metal assemblies were constructed from concentric 1-in.-thick cylindrical annuli having inside radius of 3.5, 4.5, 5.5, and 6.5 in., respectively.

^b TDC, S_n , $\epsilon = 10^{-5}$

^c Modified DDK, $\epsilon = 10^{-3}$

^d DDK, S_n , $\epsilon = 10^{-4}$

TABLE I
Results of Exponential Measurements with Slightly Enriched
Uranium Metal Tubes in Light Water

Lattice Spacing, in.	Water-to-Uranium Volume Ratio	Measured Extrapolation Distance, cm	Material Buckling, 10^{-6} cm^{-2}
Tube-in-Tube Assemblies			
Inner Tube: 1.002 Wt% U-235 (0.494-in. ID; 1.181-in. OD - 0.020-in. Al Clad) Outer Tube: 1.002 Wt% U-235 (1.794-in. ID; 2.337-in. OD - 0.020-in. Al Clad)			
2.40	0.68	9.5	-943
2.80	1.35	7.5	2049
3.10	1.93	7.7	2255
3.40	2.56	7.5 ^a	1276
Outer Tubes Only			
1.002 Wt% U-235 (1.794-in. ID; 2.337-in. OD - 0.20-in. Al Clad)			
2.40	1.60	8.51	1931
2.80	2.64	7.34	2434
3.10	3.51	7.70	1411
3.40	4.47	7.0 ^a	14
Tube-in-Tube Assemblies			
Inner Tube: 0.95 Wt% U-235 (0.476-in. ID; 1.179-in. OD - 0.019 0.34-in. Zr Clad) Outer Tube: 1.25 Wt% U-235 (1.802-in. ID; 2.368-in. OD - 0.019-in. Zr Clad)			
2.40	0.66	8.40	1504
2.80	1.30	7.92	3244
3.10	1.86	7.13	3884
3.40	2.47	6.83	2785
Outer Tubes Only			
1.25 Wt% U-235 (1.802-in. ID; 2.368-in. OD - 0.019-in. Zr Clad)			
2.40	1.57	8.20	4436
2.80	2.53	7.82	4624
3.10	3.36	7.58	3789
3.40	4.27	7.32	2180

^a Estimated from plot of measured λ versus H_2O/U volume ratio.

TABLE II
Results of Critical-Approach Measurements with Slightly Enriched
Uranium Tubes in Light Water

1.95 Wt% U-235 (1.414-in. ID; 2.275-in. OD - 0.030-in. Zr Clad)

Lattice Spacing, in.	Vw/Vu	Uranium Height in.	No. of Tubes for Critical ^a	Extrapolation Distance cm ^b	Material Buckling 10^{-6} cm^{-2} ^c
2.80	1.58	27.3	30.96	7.81	8442
		40.3	28.43		
3.10	2.19	27.3	26.48	7.62	8309
		40.3	21.36		
3.40	2.87	27.3	27.88	7.39	7287
		40.3	24.44		

^a Estimated from least-squares fit of inverse multiplication measurements.

^b Extrapolation lengths were taken from the measurements on the 1.002 Wt% U-235 enriched tubes.

^c Bucklings computed from data on the 27.3-in. fuel height; the 40.3-in. height requires correction for the effect of end caps.

Calculational Checks of Plutonium Cross Sections by Comparison with Critical-Assembly Results /D. A. Meneley, J. R. White (ANL)

A computational study has been carried out on several plutonium-fueled and mixed-fueled critical assemblies of small to medium size to verify the plutonium cross-sections of the twenty-six group set recently produced at Argonne National Laboratory. Plutonium worths in some larger uranium-fueled reactors have been computed to ascertain the effect of softer spectra on the comparisons. Other reactor parameters included in the study were detector-response ratios and U-235, U-238 and B-10 central danger coefficients, as well as effective delayed-neutron fraction and prompt-neutron lifetime.

Critical masses of cylindrical cores were determined using one-dimensional transport theory, S4 approximation, in finite cylindrical geometry. Checks on the validity of this procedure were made with a two-dimensional transport code. Elastic scattering for light elements in all cores were determined from ultra-fine-group normal-mode calculations.

Central reactivity worths were calculated in spherical geometry by void replacement. Effective delayed-neutron fractions were determined both by subtraction of delayed-group spectra from the fission-source distribution and by the standard diffusion-theory perturbation method. Prompt-neutron lifetimes were calculated by 1/v poison and by perturbation methods.

Table I lists results for some of the assemblies studied. Critical masses are in good agreement except for the smallest assembly, number 3-44. This was a small, highly reflected cylindrical core, whose calculation was made in spherical geometry. Detector ratios are in good agreement throughout. Data for 3-46B and 3-46C are not yet available.

A definite trend appears in the U-235 and Pu-239 central worth figures. For hard spectrum cores the calculated values fall below the measured, while for soft spectrum cores the calculated worth is considerably higher. The discrepancy is consistent for U-235 and Pu-239.

Criticality Studies with Plutonium Solutions /R. C. Lloyd, C. R. Richey, E. D. Clayton, D. R. Skeen (BNW)

A series of criticality experiments has been performed at the Hanford Critical Mass Laboratory¹ with plutonium (4.6 wt% Pu-240) nitrate solutions in thin stainless-steel spheres of 11.5-, 14- and 15.2-in. diam. Various types of neutron reflectors used in the experiments were water, concrete, paraffin and stainless steel. Criticality data were also obtained with the 15.2-in. sphere unreflected. The effect of partial reflection by concrete and the effect of an air gap between the core and its reflector were also examined. Critical plutonium concentrations were measured in the range from 24 to 435 g Pu/l, depending on the nitric acid molarity (0.2 to 7.7 molar) and the condition of neutron reflection.

Results of the measurements with the 11.5- and 15.2-in.-diam spheres are given in Table I. Critical experiments with the 14-in.-diam sphere were previously reported². Minimum critical volumes for the plutonium nitrate solution containing 4.6 wt% Pu-240, when bare and water reflected, were determined from interpolation of the data to be about 22 and 11 l, respectively, for a Pu concentration of 175 ± 25 g Pu/l. A concrete reflector was found to be somewhat more effective than water on the 14-in. sphere, but slightly less effective on the 11.5-in. sphere where the concentration of Pu was greater. Neutron-flux measurements made in the 11.5-in. and the 15.2-in. sphere were used to determine extrapolation lengths and, hence, critical bucklings for the solutions. A typical flux plot is shown in Fig. 1. The effect of a stainless-steel shell on criticality was also evaluated in several experiments by the addition of extra shells that permitted a subsequent extrapolation to zero shell thickness.

Multigroup diffusion theory was used to compute criticality for some of the $\text{Pu}(\text{NO}_3)_4$ solutions in the experiments. The results are shown in Table I where the

Fig. 1. Gold-Foil Transverse in 15.2-in. diam. Concrete Reflected Sphere with Air Gap.

in Diffusion and Transport Theory Calculations," BNWL-35, (March, 1965).

TABLE I
Criticality of Plutonium Nitrate Solutions

Reference	Sp. Gr.	(g Pu/l)	Molarity	Total Nitrate (g NO ₃ /l)	H ₂ O (g/l)	H ⁺ /Pu Atomic Ratio	Crit. Mass (kg Pu)	Calc. Exp.
15.2-in. Stainless-Steel Sphere - 12.95" (Vessel Wall Thickness 0.040 in.)								
Water	1.120	75	0.2	86	971	354	0.95	1.0034
	1.143	74.5	0.4	105	963	344	0.96	1.0023
	1.230	100	1.0	230	889	243	1.30	0.9971
	1.282	126	2.3	383	802	182	1.63	1.0057
Water	1.084	88	1.1	548	864	87	3.48	1.0039
	1.482	385 ^b	0.8	303	883	81	3.82	1.0150
	1.880	425 ^b	0.8	372	879	84	8.83	1.0154
	1.445	75	0.4	188	881	341	0.97	
Concrete (11.25 in.)	1.432	324	1.2	318	878	188	3.85	
15.2-in. Stainless-Steel Sphere - 30.2" (Vessel Wall Thickness 0.040 in.)								
Unreflected	1.081	29.0	0.4	64	979	968	1.18	1.0010
	1.429	172.3	4.0	606	766	125	5.30	0.9963
Stainless Steel (0.26 in.)	1.076	84.3	0.5	66	975	758	1.04	
Concrete Shell With Air Gap (0-in. Concrete; 2.83-in. Air Gap)	1.071	29.0	0.6	64	977	900	0.94	
Water	1.000	24.6	0.5	56	977	1000	0.74	1.0004
Water and 0.000-in. Stainless Steel	1.000	28.7	1.7	517	977	553	1.17	1.0078
Water Only (Corrected for Effect of Vessel Wall & Supports)		33.6	0.5	55	979	1107	0.71	1.0036
Unreflected (Corrected for Effect of Vessel Wall & Supports)		41.8	0.4	67	976	834	1.24	0.9991

^a Includes 4.8 wt% Pu-240; experimental values not corrected for vessel walls or neck supports.
^b Contains Pu(VI) and Pu polymer.

computed values of k_{eff} should be unity to be in agreement with the experiment. The GAMTEC-II computer code³ was used to calculate the eighteen-group constants utilized in the multigroup calculation. The criticality calculations are in good agreement with experimental data.

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3. CARTER, L. L., et al., "GAMTEC-II: A Code for Generating Consistent Multigroup Constants Utilized

TABLE I
Preliminary Results of Critical Comparison

Assembly ^a	Critical Mass ^b		$\sigma_t^{238}/\sigma_t^{235}$		$\sigma_t^{238}/\sigma_t^{235}$		Pu-239	Central Worth ^c		U-235	Central Worth ^c	
	Meas.	Calc.	Mean	Calc.	Mean	Calc.		Meas.	Calc.		Meas.	Calc.
3-44	99.88	89.38	1.19	1.23	0.0816	0.0786	924	867	514	477		
3-46B	134.4	129.0		1.21		0.0703		718		398		
3-46C	151.7	150.8		1.20		0.0696		569		315		
3-38	423.9	409.5	1.17	1.18	0.0305	0.0327	289	259	164	155		
3-39	426.5	397.9	1.18	1.18	0.0475	0.0450	197	191	108	108		
3-35	457.0	470.9	1.09	1.09	0.0328	0.0323	162	195	96	112		
6-2	544.0	501.7	1.05	1.07	0.0351	0.0354	167	191	104	118		

^a 3- indicates ZPR-III; 6- indicates ZPR-VI. Following are the assembly numbers.

^b Total U-235 and Pu-239 where applicable (kg). Calculated values are corrected.

^c In hours/kg.

Measurement of Reactor Parameters

Analysis and Measurement of PuAl-Polyethylene Systems in the PCTR / D. G. Albertson, G. J. Busselman (BNW)

A series of experiments with hydrogen-moderated plutonium systems has been performed in the Physical Constants Testing Reactor (PCTR)¹. The boron concentration required to reduce the infinite multiplication factor, k_{∞} , to unity was measured for various Pu-240 concentrations, fuel plate thicknesses, and moderator-to-fuel ratios. The measured values of the boron concentration were compared to theoretical predictions to check the accuracy of calculational methods and parameters. The principal components for these experiments were Pu-Al (20 wt% Pu), pure polyethylene, and borated polyethylene (0.90 wt% natural boron). All three materials were in the form of disks, 1.960-in. in diam and 0.020-in. thick.

The disks were stacked into aluminum cans (12.0-in. inside length, 2.0-in. ID, and $\frac{1}{8}$ -in. wall thickness). The loading arrangement shown in Fig. 1 was surrounded by the graphite and fuel of the PCTR. The fuel, moderator and poison disks were stacked in layers in the cans so that continuous slabs of each material were formed. For each system the boron concentration was varied until $k_{\infty} = 1.00$. This condition was detected by the standard method of reactivity comparisons^{1,2}. Table I gives the

* These cans form a region where the thermal spectrum of the PCTR is hardened to more closely match the characteristic spectrum of the PuAl lattice.

Fig. 1. Fuel Loading Arrangement.

TABLE I

Plutonium Compositions

Type	at% Pu-239	at% Pu-240	at% Pu-241	at% Pu-242
I	90.48	8.19	1.25	0.08
II	77.80	18.89	3.01	0.30
III	67.42	26.86	4.76	0.96

plutonium compositions. The results of the experiments and calculations are summarized in Figure 2.

In the calculations the multiplying medium was represented by one-dimensional slabs of fuel and moderator. The calculated values agree well with the observed experimental trends. The discrepancy between measured and calculated values of the B-10/Pu ratios range from 10 to 20%, which corresponds to about 2.5 to 5% in k_{∞} . The discrepancy has not yet been fully resolved.

The calculations used the SPECTRUM code³ with a polyethylene kernel⁴ to obtain thermal cross sections and the GAM⁵ slowing down code to generate epithermal group constants. These cross sections were then used in the S-XIII (Ref. 6) transport theory code to carry out a 4-energy group, 13 double-angle calculation. The microscopic cross sections used are based on the work of Liikala⁷, except that the Pu-239 cross sections are normalized to those of Sher⁸ for 2200 m/sec.

Various methods were compared for obtaining the Pu-240 absorption cross section near the 1 eV resonance. For most calculations the NR1A approximation was found to be adequate.

As shown in Fig. 2, the discrepancy between calculated and measured B-10/Pu ratios is greater for the higher

Fig. 2. Characteristics of PuAl Borated Polyethylene Systems ($k_{\infty} = 1.00$ in all cases.).

H/Pu systems. The analysis of these very thermal systems should benefit from a more exact representation of the thermalization process. For this purpose the space-dependent thermalization code, THERMOS⁹, is being used. Preliminary results indicate that the discrepancy in k_{∞} can be reduced to less than one percent.

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Physics Measurements

Studies of Subcritical Lattices with Distributed Neutron Absorber by Means of the Pulsed-Neutron Source Technique / Henry E. Bliss, D. D. Lanning, Joseph Harrington, I. Kaplan, B. K. Malaviya, T. J. Thompson (MIT)

Pulsed neutron measurements have been made on subcritical uranium-heavy-water lattices with and without distributed neutron absorbers leading to a determination of the amount of absorber necessary to reduce the prompt infinite multiplication factor to unity. A value of k_{∞} has also been obtained for each lattice studied.

The prompt-neutron fundamental-mode decay constant for a bare, homogeneous, multiplying medium can be expressed in terms of the medium's nuclear parameters¹

$$\lambda = \bar{v}\Sigma_a + \bar{v}DB^2 - \bar{v}\Sigma_a k_{\infty}(1-\beta)P(B^2), \quad (1)$$

where β is the effective delayed-neutron fraction and $P(B^2)$ is the fast non-leakage probability. If a Fermi slowing down kernel is used for the fast non-leakage probability ($P(B^2) = e^{-B^2\tau} \approx 1 - B^2\tau + \frac{1}{2}B^4\tau^2$), Eq. (1) takes the form

$$\lambda = m + nB^2 - qB^4, \quad (2)$$

where

$$m = \bar{v}\Sigma_a [1 - k_{\infty}(1-\beta)] \quad (3a)$$

$$n = \bar{v}D + \bar{v}\Sigma_a k_{\infty}(1-\beta)\tau \quad (3b)$$

$$q = \frac{1}{2}\bar{v}\Sigma_a k_{\infty}(1-\beta)\tau^2. \quad (3c)$$

The product $\bar{v}\Sigma_a k_{\infty}(1-\beta)$ may be written as

$$\bar{v}\Sigma_a k_{\infty}(1-\beta) = (1-\beta)\eta\rho_e \frac{\bar{v}\Sigma_a^{\text{fuel}}}{1 + \sum_i \frac{V_i}{V_{\text{fuel}}} \frac{\phi_i}{\phi_{\text{fuel}}}}, \quad (4)$$

where the summation subscript 'i' represents moderator and any other material (such as neutron absorber) present in the lattice. Thus, $\bar{v}\Sigma_a k_{\infty}(1-\beta)$ changes only due to changes in volume fractions and flux disadvantage factors and the coefficients n and q in Eqs. (3b, c) exhibit relatively small fluctuations following the addition of thermal absorber. The coefficient m reflects the effect of adding absorber and decreases with increasing absorber concentration. In addition, Eqs. (3a, b) may be solved (using calculated values of τ and β and a Maxwellian averaged value of $\bar{v}D$) to obtain values of k_{∞} and $\bar{v}\Sigma_a$ for each lattice studied.

TABLE I

Values of k_{∞} and the Least-Squares Coefficients m , n and q in Subcritical Lattices With and Without Absorber

Lattice	m (sec ⁻¹)	n (10 ⁵ cm ² sec ⁻¹)	q (10 ⁹ cm ⁴ sec ⁻¹)	k_{∞}
(1)	-280.0 ± 29.5	3.288 ± 0.082	10.51 ± 22.92	1.352 ± 0.058
(2)	-63.9 ± 18.9	2.936 ± 0.047	5.07 ± 8.64	1.094 ± 0.033
(3)	+36.1 ± 29.4	3.279 ± 0.074	7.87 ± 13.59	0.976 ± 0.025

Fig. 1. Values of the Prompt Multiplication Factor as a Function of the Percent of Moderator Displaced.

An initial set of experiments has been performed in the MIT Lattice Facility² on a lattice consisting of 0.25-in.-diam, 1.143% enriched-uranium metal rods located on a 2.50-in. triangular pitch and immersed in 99.50 mole% heavy water. Copper in the form of thin rods served as a thermal absorber. Three lattices were studied, 1) the clean core with no copper, 2) a lattice with one 0.144-in.-diam copper rod per fuel rod, and 3) a lattice with two such rods per fuel rod.

Values for coefficients m , n and q obtained from least-squares fitting of the λ vs B^2 data for the three lattices studied are given in Table I along with the determinations of k_{∞} . The variation of the prompt infinite multiplication factor ($k_{\infty}(1-\beta)$) with percent of moderator displaced is shown in Fig. 1. A null state ($k_{\infty}(1-\beta) = 1$) is obtained for a percent of moderator displaced equal to $0.550 \pm 0.013\%$. Extensive steady-state measurements, permitting comparison with this value, have also been made and will be reported in the near future.

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Aerospace Nuclear Safety—Snap

The Criticality of Curium-244/ Vahê Keshishian, Eric H. Ottewille, Charles L. Dunford (AI)

Curium-244 promises to be an excellent radioisotope heat source. The question of criticality arises for large quantities of Cm-244.

Since there is a lack of experimental data on neutron cross sections for Cm-244, fast-neutron cross sections were generated by means of nuclear systematics and nuclear models. The NEARREX¹ and 2-PLUS² codes were used to generate the neutron cross sections that were then used to calculate the critical mass of Cm-244 as a function of density. Fig. 1.

Neutronic calculations were made using the one-dimensional DTF-II code, embodying Carlson's S₄ transport-theory approximation. Twenty-six group cross sections were prepared for Cm-244 using Bondarenko³ energy group structure. This group structure was selected since it has eight energy groups above 0.1 MeV. Plutonium-240 neutron-scattering matrix³ was used for all energy groups. The number of neutrons emitted in the induced fission of Cm-244 was calculated using nuclear systematics^{4,5} and is equal to $\nu(E) = 3.23 + 0.13 E(\text{MeV})$.

Figure 1 shows the unreflected critical mass of Cm-244 spheres as a function of density. The calculated critical mass of Pu-238 and the critical mass of Pu-239 (Jezebel) are shown for comparison. At a density of 15.66 g/cm³ the critical mass of Pu-238 is about equal to that of Pu-239, and the critical mass of Cm-244 is approximately one half that of Pu-238. The above shows the importance of the critical mass of Cm-244.

Fig. 1. Critical Mass of Unreflected Pu-238 and Cm-244 Spheres.

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Lattice Calculations and Measurements

Experimental Bucklings for Seven-Rod Clusters of Natural UC in Heavy Water Moderator* / R. E. Green, R. E. Kay, A. Okazaki (AECL) G. Casini, H. J. Metzendorf, M. Paillon (Euratom, Ispra)

Bucklings for D₂O-moderated lattice of seven-rod natural UC clusters have been measured in the ZED-2 critical facility at Chalk River¹. Twenty-five complete fuel assemblies borrowed from EURATOM at Ispra (being studied there as part of the ORGEL project) were used in the experiments. Each cluster contained seven 3-m long rods of 2.52-cm diam UC of density 13.54 g/cm³ sheathed in Al (2.55-cm i.d., 1.0-mm wall). The cluster arrangement was a central pencil plus six on a 5.90-cm diam circle; each cluster was contained in an Al "pressure tube" (9.10-cm i.d., 2.0-mm wall) and "calandria tube" (10.10-cm i.d., 1.5-mm wall), also Al.

The bucklings were determined using the substitution method, in which up to 25 UC assemblies successively replaced reference fuel at the center of a large core of the latter. The difference in buckling between test and reference regions was derived from the observed changes

in critical moderator level. We have used 19-rod UO₂ CANDU-type bundles for our reference fuel².

Measurements were made at room temperature in square lattices at several pitches in the range 20 to 32 cm, with the UC clusters having four different "coolants": D₂O; air; HB-40 (Green and Bigham³); and Dowtherm A. The reference fuel always had D₂O coolant.

The critical height measurements were analyzed by four different methods

1) a simplified Persson method³ using only 5 and 9 cluster substitutions

2) a one-group statistical weight method that uses a Co wire absorber² to determine the change in radial buckling with critical height for our reflected system

3) a two-group, two-region homogeneous diffusion-theory method developed at EURATOM and incorporated in the computer program BERTHE⁴

4) a heterogeneous source-sink method, the MICRETE program described in another paper at this conference⁵.

The results are listed in Table I as the difference between the bucklings of the test and reference fuels for the actual experimental conditions, to compare the various methods of analysis. Also given in this table are

*Sponsor: N. L. Snider

TABLE I
Bucklings for D₂O-Moderated Lattices of Natural UC Clusters

Square Pitch cm	N ^a	Moderator Purity at.% D ₂ O	Moderator Temperature °C	B ² _{Ref} m ⁻²	ΔB ² = B ² _{UC} - B ² _{Ref} (m ⁻²)				B ² _{UC} (25°C, 99.63 at.%) m ⁻²
					Persson	Co Wire	BERTHE	MICRETE	
D ₂ O Coolant									
20	121	99.531	24.76	3.498	0.34	0.39	0.33	0.39	3.93
24	121	99.532	24.76	3.995	1.07	1.03	0.89	0.91	4.99
28	97	99.534	24.47	3.560	1.21	1.13	0.99	1.00	4.68
32	69	99.534	24.55	2.834	1.01	1.00	1.12	1.00	3.97
Air Coolant									
20	121	99.618	26.13	3.527	0.27	0.34	0.15	0.05	3.59
24	121	99.622	25.49	4.071	1.36	1.25	1.00	0.88	4.96
28	97	99.601	27.08	3.630	1.74	1.63	1.30	1.17	4.85
32	69	99.593	26.75	3.913	1.53	1.72	1.28	1.25	4.22
Dowtherm A Coolant									
20	121	99.554	24.73	3.508	-0.91	-0.97	-0.86	-0.91	2.63
22	121	99.557	24.98	3.935	-	-	-0.87	-0.80	3.18
24	121	99.563	26.35	4.013	-0.98	-0.98	-0.94	-0.82	3.26
28	97	99.575	26.41	3.601	-0.92	-0.93	-0.81	-0.75	2.93
32	69	99.578	26.44	2.892	-0.77	-0.79	-0.83	-0.75	2.22
HB-40 Coolant									
20	121	99.529	24.31	3.500	-1.56	-1.62	-1.69	-1.64	1.90
24	121	99.543	25.06	4.003	-1.77	-1.76	-1.77	-1.64	2.44
28	97	99.545	25.30	3.570	-1.67	-1.66	-1.63	-1.57	2.11

^aN is the total number of fuel assemblies in each lattice.

^bB_{ref}² is the buckling for the CANDU reference fuel assemblies.

^cThese values are obtained from the reference bucklings and the ΔB² values given by MICRETE.

bucklings for the reference fuel as determined by flux mapping. It is seen that all methods give the same results within about $\pm 0.10 \text{ m}^{-2}$ for all but the air-cooled clusters, where it is felt that the one-group methods of analysis as used here do not properly allow for the changes in diffusion coefficients.

The various methods of analysis were tested by doing substitution experiments with fuel assemblies for which bucklings had been determined from flux-mapping experiments in large cores². The results of these test cases showed that the heterogeneous method gave the best agreement with the flux-mapping experiments and, therefore, the buckling values listed in the last column of Table I are derived from this method of analysis. These final values have been corrected to moderator conditions of 25°C and 99.63 at.% D₂O. We believe the errors to be $\leq \pm 0.10 \text{ m}^{-2}$.

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Experimental and Analytical Results for PuO₂-UO₂-H₂O Lattices/R. C. Liikala, W. P. Stinson (BNW)

The results of subcritical experiments and analytical correlations for PuO₂-UO₂ light-water lattices are described. The fuel rods are 0.030-in. thick Zircaloy tubes with a 0.508-in. i.d. filled with vibratory compacted 2 wt% PuO₂-UO₂ to a length of 36.0 in. Each rod contains an average of 1128 g of PuO₂-UO₂. The U in UO₂ is natural, and the isotopic compositions of the plutonium are, by weight, 91.62% Pu-239, 7.65% Pu-240, 0.70% Pu-241, 0.03% Pu-242, and 81.11% Pu-239, 16.54% Pu-240, 2.15% Pu-241, 0.20% Pu-242, and 71.76% Pu-239, 23.50% Pu-240, 4.08% Pu-241, 0.66% Pu-242. The rods were loaded in hexagonal lattices at spacings covering the range of moderator-to-PuO₂-UO₂ volume ratios of 0.61 to 9.7. All lattices were completely reflected with H₂O.

The experimental procedures are identical to those utilized previously¹ in the determination of critical masses and bucklings of 1.5 wt% PuO₂-UO₂-H₂O lattices.

The results of experimental and theoretical analysis are shown in Table I. Variations with lattice spacing and Pu-240 concentration are apparent in calculated multiplications. The multiplications and numbers of rods for critical were obtained from a one-dimensional four-energy group-diffusion theory calculation² assuming a two-region reactor consisting of a homogeneous core and an H₂O reflector. Axial leakage effects were accounted for with transverse bucklings. Cross sections for the homogenized lattice were obtained from unit cell calculations utilizing the HRG code³ for nonthermal effects, and the THERMOS code⁴ for thermal effects. Dancoff factors

TABLE I
Experimental and Analytical Results of 2 wt% PuO₂-UO₂ Rods in H₂O

Lattice Spacing (in.)	Pu-240 Isotopic Composition (wt%)	No. of Rods For Critical Experimental	Temperature (°C)	Calculated Multiplications k_{eff}	Reflector Savings (cm)	Bucklings (m^{-2})
0.66	8	1503. $\pm$ 14	25.2	0.982	7.30 $\pm$ 0.18	42.5 $\pm$ 0.6
	8	319.0 $\pm$ 0.3	23.9	0.998	8.11 $\pm$ 0.04	86.9 $\pm$ 0.3
0.80	24	524.2 $\pm$ 1.3	24.4	1.011	8.07 $\pm$ 0.06	63.3 $\pm$ 0.3
	8	193.0 $\pm$ 0.1	20.2	0.998	7.65 $\pm$ 0.02	102.1 $\pm$ 0.2
0.93	16	241.2 $\pm$ 0.3	20.0	1.004	8.22 $\pm$ 0.04	85.1 $\pm$ 0.3
	24	286.7 $\pm$ 0.4	24.7	1.005	7.65 $\pm$ 0.03	79.1 $\pm$ 0.2
	8	149.0 $\pm$ 0.4	19.8	0.988	7.95 $\pm$ 0.06	100.8 $\pm$ 0.6
	16	188.1 $\pm$ 0.3	22.6	0.994	8.07 $\pm$ 0.04	86.2 $\pm$ 0.3
1.05	24	234.2 $\pm$ 0.3	23.5	1.002	7.92 $\pm$ 0.05	75.7 $\pm$ 0.3
1.143	8	146.8 $\pm$ 0.3	27.8	0.994	7.42 $\pm$ 0.05	95.0 $\pm$ 0.4
1.32	8	165.8 $\pm$ 0.4	23.6	0.993	7.02 $\pm$ 0.05	74.5 $\pm$ 0.3
	8	180.9 $\pm$ 0.4	26.2	0.986	6.86 $\pm$ 0.07	66.4 $\pm$ 0.3
	16	254.9 $\pm$ 0.6	24.4	0.986	5.40 $\pm$ 0.09	59.6 $\pm$ 0.3
	24	366.7 $\pm$ 0.4	23.6	0.989	5.29 $\pm$ 0.03	44.4 $\pm$ 0.1
1.386						
1.60	8	474 $\pm$ 18	25.3	0.978	4.8 $\pm$ 0.8	31.7 $\pm$ 1.8
1.61	8	472 $\pm$ 3	19.4	0.965	3.4 $\pm$ 0.2	33.3 $\pm$ 0.3

taken from published tables⁵ were used for the resonance shadowing corrections in the HRG code. The Nelkin scattering model for hydrogen was used in the thermal calculations. All microscopic cross sections are taken from the BNW Master Library⁶. The thermal cross sections for Pu-239 are the earlier recommended set of Sher⁷ in which $\eta_{\text{Pu-239}} = 2.093$.

More recent evaluations^{8,9} result in a value for $\eta_{\text{Pu-239}} = 2.115$. If the latter sets^{8,9}, and probably the more correct cross-section data for Pu-239 were used, the multiplications would all be $\approx 1\%$ higher. Thus, any calculated multiplications near and/or at unity probably are the result of compensating errors in the theoretical methods.

Results of calculations for the 1.5 wt% PuO₂-UO₂-H₂O experiments¹⁰, and for the Saxton critical experiments¹¹ using 6.6 wt% PuO₂-UO₂ rods, confirm this conclusion.

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Critical Experiments and Calculations in Some Very Undermoderated, Slightly Enriched, Uranium-Oxide/Water Lattices/A. R. Boynton, E. M. Pennington (ANL)

The High Conversion Critical Experiment (Hi-C) at Argonne National Laboratory was initiated to extend the range of investigations with light-water-moderated, slightly enriched, oxide cores. Lattices have been assembled that had H-to-U-238 atom ratios of from 4.16 to 0.50. The fuel was 3.04 wt% enriched uranium in the form of 10.17 g/cm³ UO₂ pellets of 0.935-cm diam loaded to 122-cm length in cladding tubes of either 1.058-cm o.d., 0.048-cm thick aluminum, or 1.057-cm o.d., 0.0496-cm

thick stainless steel. Both square and triangular lattice pitches were used.

The modified conversion ratio, U-238 to U-235 fission ratio (δ^{28}), U-238 capture cadmium ratio, and thermal disadvantage factor have been measured in the full range of lattices, while critical bucklings and reflector savings have been measured in the looser assemblies. The measurements of lattice and cell parameters were carried out as previously described¹⁻⁴.

Calculations of four-group constants, material buckling, reflector savings, and various cell parameters were performed. Constants for three fast groups with lower energy boundaries at 1.35 MeV, 1.23 KeV, and 0.414 eV were obtained with the GAM-I code⁵, while those for the thermal group were found using THERMOS⁶. The P-1 option was used in GAM-I, while the Nelkin kernel for hydrogen in H₂O was used in THERMOS. An extra scattering region was used in THERMOS to mock-up the isotropic return condition at the unit-cell outer boundary. Both experimental and calculated results are given in Table I.

Calculated critical bucklings obtained by fundamental mode calculations of four-group diffusion theory differ only slightly from experimental bucklings, in contrast to preliminary calculations⁷ which yielded values much higher than experiment. This improved agreement is attributed to the replacement of U-235 and U-238 cross sections in the original GAM-I library by versions obtained from Hanford⁸. Radial reflector savings were obtained using the Argonne multigroup diffusion-theory code REX. Agreement with experiment is good. The U-238-to-U-235 fission ratio was calculated using the four-group constants and fluxes. Agreement with experiment is poor. The explanation for the discrepancy is not known; it may be caused by inadequacies in cross sections or by some systematic experimental error. The calculated values of the modified conversion ratio are in reasonable agreement with experiment.

Improvements could be made in the theoretical methods used in the study, although those used here should be reasonably accurate. However, even with very good methods, the theoretical study of tightly packed, light-water-moderated lattices is difficult. This is because nonthermal events are of considerable importance, and inaccuracies in cross-section data produce large uncertainties in the calculations.

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TABLE I
Hi-C Parameter Values

Lattice Pitch, cm	Clad	H to U-238 Atom Ratio	Critical Buckling $\text{cm}^{-2} \times 10^4$		Radial Reflector Savings, cm		δ^{28}		Modified Conversion Ratio	
			Calc.	Experimental	Calc.	Experimental	Calc.	Experimental	Calc.	Experimental
1.349 sq	Al	4.15	89.94	91.82 ± 0.80	7.78	7.38 ± 0.12	0.0642	---	0.5141	---
1.349 sq	SS	4.16	66.35	---	---	---	0.0661	0.0742 ± 0.0011	0.5472	0.635 ± 0.007
1.24 sq	Al	2.91	71.11	70.76 ± 0.71	8.45	8.13 ± 0.16	0.0823	---	0.6218	---
1.24 sq	SS	2.92	49.68	47.47 ± 0.27	7.76	7.69 ± 0.08	0.0838	0.1011 ± 0.0004	0.6679	0.622 ± 0.006
1.27 tri	Al	2.29	57.77	55.38 ± 0.24	9.02	8.77 ± 0.17	0.0965	---	0.7067	---
1.27 tri	SS	2.29	37.64	---	---	---	0.0974	0.1160 ± 0.0017	0.7631	0.730 ± 0.021
1.166 tri	Al	1.32	26.49	24.36 ± 0.10	10.86	10.73 ± 0.14	0.1349	---	0.9790	---
1.166 tri	SS	1.33	7.52	---	---	---	0.1337	0.1539 ± 0.0039	1.078	1.082 ± 0.006
1.127 tri	Al	0.98	13.80	---	---	---	0.1566	---	1.137	---
1.127 tri	SS	0.98	-4.80	---	---	---	0.1537	0.1877 ± 0.0020	1.265	1.203 ± 0.025
1.069 tri	SS	0.50	-27.73	---	---	---	0.1979	0.2622 ± 0.0036	1.834	2.023 ± 0.047

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Reactor Calculations

Criticality of Heterogeneous Arrays Undergoing Dissolution/C. R. Richey (BNW)

A perplexing problem in chemical plants reprocessing reactor fuel is the paucity of experimental data on the criticality of slightly enriched uranium fuel elements undergoing dissolution. The problem is somewhat tedious to treat theoretically, because, during dissolution, the resonance absorbing U-235 becomes admixed to the moderator. Chernick¹ proposed a semianalytical approach which, with subsequent development^{2,3}, provides an accurate method for computing the resonance integral for a lattice array. In this paper, this semianalytical method is extended to include resonance absorber admixed in the moderator. Utilizing the method in conjunction with a two-group calculation similar to that of Hellens et al., for the water-moderated lattice experiments of Kouts et al.⁴, critical bucklings were computed for slightly enriched uranium rods (1 to 5 wt% U-235) latticed in uranium-water mixtures.

The average collision density in an absorbing lump imbedded in an absorbing moderator expressed in the notation of Nordheim³ is

$$F_0(u) = (1 - P_0) \left[\frac{1}{\alpha_0} \int_{u-\Delta_0}^u F_0(u') \frac{\sigma_{s_0}}{\sigma_{t_0} + \sigma_m} du' + \frac{1}{\alpha_m} \int_{u-\Delta_m}^u F_0(u') \frac{\sigma_m}{\sigma_{t_0} + \sigma_m} du' \right] + P_1 \left[\frac{1}{\alpha_1} \int_{u-\Delta_1}^u F_1(u') \frac{\sigma_{s_1}}{\sigma_{t_1}} du' + \frac{1}{\alpha_0} \int_{u-\Delta_0}^u F_1(u') \frac{C\sigma_{s_0}}{\sigma_{t_1}} du' \right], \quad (1)$$

where $C = (N'_0 V_1 / N_0 V_0)$ with N_0 and N'_0 being the atom densities of the moderator in the lumped volume V_0 and the moderator volume V_1 , respectively. In the NR approximation, the resonance integral for the absorbing lump becomes

$$I_{NR}^A = \int \frac{\sigma_{a_0} [\delta p_0 + \sigma_m + P_0 (\sigma_{t_0} - \sigma_{p_0})]}{\sigma_{t_0} + \sigma_m} du - \int \sigma_{a_0} \frac{C\sigma_{r_0}}{\sigma_{t_1}} P_0 du, \quad (2)$$

where $\sigma_{r_0} = \sigma_{t_0} - \sigma_{p_0}$. The first term on the right-hand side of Eq. (2) is the usual resonance integral expression for a lumped absorber in the NR approximation³, and the second term modifies the surface absorption for resonance shielding by the absorbing moderator.

The average collision density in the moderator is

$$F_1(u) = (1 - P_1) \left[\frac{1}{\alpha_1} \int_{u-\Delta_1}^u F_1(u') \frac{\sigma'_{s_1}}{\sigma_{t_0} + \sigma_{s_1}} du' + \frac{1}{\alpha_0} \int_{u-\Delta_0}^u F_1(u') \frac{\sigma_{s_0}}{\sigma_{t_0} + \sigma'_{s_1}} du' \right] + P_0 \left[\frac{1}{\alpha_0} \int_{u-\Delta_0}^u F_0(u') \frac{C^{-1} \sigma_{s_0}}{C^{-1} \sigma_{t_0} + \sigma'_m} du' + \frac{1}{\alpha_m} \int_{u-\Delta_m}^u F_0(u') \frac{\sigma'_m}{C^{-1} \sigma_{t_0} + \sigma_m} du' \right], \quad (3)$$

TABLE I

Comparison of Experimental and Calculated Bucklings for Water and Uranyl-Nitrate Moderated Arrays

Method of Determination	B_m^2, m^{-2}		
	Water Moderated	Uranyl Nitrate Moderated	$\Delta B_m^2, m^{-2}$
Experiment	33.84	12.52	21.32
Monte Carlo	22.68	- 3.68	26.36
Calculated (this paper)	36.98	15.64	21.34

where all cross sections are now in reference to one absorber atom in the moderator. In the NR approximation the resonance integral for the moderator is

$$I_{NR}^M = \int \sigma_{a_0} \frac{\sigma'_{s_1} + \sigma_{p_0}}{\sigma'_{s_1} + \sigma_{t_0}} du - (C^{-1} \sigma_{s_m} - \sigma'_m) \times \int \frac{\sigma_{a_0} \sigma_{r_0}}{(\sigma'_{s_1} + \sigma_{t_0})^2} P_0 du. \quad (4)$$

The first term on the right side of Eq. (4) is the usual expression for a homogeneous mixture³. The second term contains two effects due to the presence of a lumped absorber: 1) capture in the lump of neutrons that would otherwise be captured in the moderator absorber; and 2) capture in the moderator absorber of neutrons transferred into resonance by the admixture in the absorbing lump.

Following Adler-Nordheim², the GAMTEC-II code⁵ was modified to evaluate the resonance integral expressions of Eqs. (2) and (4). For the energy dependence of the cross sections, the usual Doppler-broadened Briet-Wigner line shape was taken with interference scattering ignored. Resonances whose absorption is large over the energy decrement greater than $\alpha_0 E$ were treated as narrow resonances with the scattering parameters within the resonance region equated to zero, the NR1A approximation.

The change in material buckling on replacing the water moderator of a heterogeneous array (0.925-in. diam rods at a 1.5-in. spacing) with a uranyl-nitrate solution (431 g U/l) was measured by Bennett and Block⁶. Both the rods and solution consisted of uranium enriched to 1.007 wt% U-235. A comparison of the experimental results and those calculated by the above methods is given in Table I. Also given are corresponding data computed by Carter using Monte Carlo techniques⁷.

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The Accuracy of Reactor Codes for Criticality Calculations on Arrays of Fuel Elements in Water with Various Water Gaps/*J. K. Fox (PPC)*

Recently, an experimental study has been made at ORNL of the critical numbers of SPERT-D Fuel Elements in water having various water gaps between elements¹. The experiments were performed in order to extend basic criticality data in this area. Critical parameters were obtained for uniformly spaced rectangular arrays as a function of water-gap width and on a series of three- and four-slab arrays in which the water gaps between each varied. Because of the wide variations in water-gap thickness, it was felt that a comparison with experiment would be a good test of the accuracy of a set of reactor codes. For the calculated k_{eff} values, PDQ² and FOG³ few-group diffusion-theory codes were used. Four-group constants were obtained by PHROG⁴ and TAMP⁵. PHROG is a revision of GAM-1⁶, and TAMP is a thermal-range code that uses the Wigner-Wilkins equation for the energy spectrum, and a transport code for disadvantage factors. In obtaining all constants, input bucklings corresponding to overall-array geometric bucklings were used. The upper limit of the thermal range was at 0.683 eV.

For these calculations, the fuel plates were homogenized with the water separating them. The aluminum boxes were made into a separate region. In some of the early calculations, the homogenized fuel region extended to the inside of the box. However, for the more widely separated elements, this gave k_{eff} values as much as three percent high. For these cases, the homogenized region was later reduced to correspond to the fuel meat width and a third region of aluminum and water inserted between fuel and box. Thus, the true gaps between fuel were defined and the calculated results agreed better with experiment. For the rectangular arrays, PDQ in XY geometry was used. For the slab arrays where two dimensions are large, one-dimensional FOG was used in the direction perpendicular to them. However, for the double-slab case, the result was checked with PDQ. To estimate transverse bucklings, a single-channel flux synthesis utilizing both FOG and PDQ was used. From these results, water reflector savings were obtained. These varied from about 6.5 to 7.5 cm. Since each slab layer consisted of fuel elements separated 0.5-in. in the row, it was necessary to homogenize the fuel with the water gaps when using FOG. Flux-weighted average thermal constants were obtained using TAMP. In one case, a calculation was made on a rectangular-array experiment where the moderator and reflector was a dilute solution containing 4.0 g/liter U-235 and 0.891 g/liter natural boron. Here, the poison was sufficient to increase the critical number from about 12 to 33 at an element separation of 0.5 in.

TABLE I

A Comparison of Calculated k_{eff} Values With Experiments Using Spaced Arrays of SPERT-D Fuel in Water

Description of Array	Crit No	Calc k_{eff}
Rect 4 × 3.77 sq lattice array, Sep 0.0 in.	15.09 ± 0.10	1.005
Rect 4 × 3.09 sq lattice array, Sep 0.50 in.	12.36 ± 0.17	1.002
Rect 4 × 3.7 sq lattice array, Sep 1.00 in.	14.82 ± 0.10	1.008
Rect 7 × 6.88 sq lattice array, Sep 1.60 in.	46.77 ± 0.31	1.018
Rect 6 × 5.55 sq lattice array, Sep 0.50 in.	33.12 ± 0.10 ^a	1.006
Slabs: 3 rows × 16 Ele wide Sep 2.19 ± 0.03 in. ^b	48.0	1.008
4 rows × 16 Ele wide Sep 2.56 ± 0.02 in. ^b	64.0	1.010
Two double slabs 16 Ele wide Sep 6.37 ± 0.01 in. ^c	64.0	0.996 ^d

^aSolution contains 4.0 g/liter U-235 and 0.871 g/liter natural boron.

^bElements separated 0.50-in. in the rows.

^cElements separated 0.50-in. in the rows and each layer of the double slab separated by 0.50 in.

^dResult with PDQ: $k_{eff} = 1.004$.

The results of the comparisons are shown in Table I. In general, the agreement between calculated k_{eff} values and experiment is quite good. Since no flux profiles were made experimentally, no flux comparisons could be made. However, the agreement in eigenvalue indicates that flux-peaking effects are well represented for these rather wide water gaps by diffusion theory. Thus, the use of transport theory may be limited to local effects between fuel plates. The results also indicate that the methods of obtaining few-group average constants are accurate for these near-thermal systems.

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Few-Group Monte Carlo Criticality Calculations/*G. E. Whitesides, G. W. Morrison, E. C. Crume (UCC-OR)*

The accurate determination of the criticality condition of individual fissile units and of arrays of interacting fissile units is a continuing need in the criticality safety analysis of the handling of fissile material. Extensive experimental critical data have been taken for a large variety of fissile systems^{1,2}. Often, the determination of the criticality condition of a system of interest can be made or adequately estimated by a comparison with such data. Alternatively, recourse may be made to computational programs, e.g., the multigroup transport-theory codes such as DTF-II³ for one-dimensional geometry, and DDK⁴ for two-dimensional geometry. However, compli-

cated multiregional geometries, such as arrays of fissile units or individual fissile units without appropriate symmetry, cannot be handled by such calculational methods without further approximations. Such geometries, however, can be handled by some Monte Carlo neutron-transport codes. It is the purpose of this paper to give evidence that the multiplication constants for such geometries can be adequately predicted using appropriate few-group cross sections, in particular, the widely used Hansen-Roach 16-group cross-section set⁵. In addition, it is indicated that sufficient accuracy in the calculated multiplication constants for use in criticality analyses can likely be attained for a reasonable computational cost.

A general-purpose Monte Carlo neutron-transport code⁶ was altered to accept the 16-group energy structure of Hansen and Roach. The transport, scattering, and fission cross sections for the groups are input for each element in a medium. Neutron slowing down is treated by sampling from the transfer matrix cross sections $\sigma_g \rightarrow g'$. With the exception of hydrogen, all scattering is treated as being isotropic in the laboratory system. Neutrons that undergo an elastic collision with hydrogen are scattered in the forward direction only, i.e., $0 \leq \mu \leq 1$. The scattering angle for hydrogen is determined by sampling from the analytic expression for the angular distribution.

Life histories of neutrons are run in batches. A sufficient number of batches is run to obtain the statistical accuracy required. The fission-source distribution in each batch is obtained from the fission distribution generated by the previous batch. The initial batch is started from a selected spatial distribution such as a cosine. The multiplication constant for a batch is taken to be the ratio of the number of neutrons born in the fissions produced by that batch to the number of source neutrons for that batch. This multiplication constant is averaged over all batches run for a system, excluding the

first few which are still affected by the initial source distribution, to obtain the multiplication constant of the system.

Multiplication constants of experimental systems that were calculated to check the capabilities of this method are given in Table I. The systems chosen are considered to provide a fairly severe test of the capabilities. Histories for at least 4000 neutrons were used in the calculations for each system. The neutrons were started in batches of 100 or 200. It can be seen from Table I that a statistical accuracy of the order of a few percent in the multiplication constant is obtained from a relatively small number of neutrons. It is also apparent that greater statistical accuracy can be obtained simply by increasing the number of histories.

TABLE I

Calculated Multiplication Constants (k_{eff}) for Experimentally Critical Systems

	Ref	k_{eff}	Number of Neutron Histories
1. Godiva I—8.718-cm radius Oy(93.8) ^a metal sphere, unreflected.	7	1.010 ± 0.014 1.005 ± 0.011	3500 8000
2. Unreflected annular cylinder of 93.2%-enriched uranium-molybdenum alloy, 5.08-cm i.d., 10.16-cm o.d., 19.74-cm high.	8	1.004 ± 0.016 0.978 ± 0.011	1700 4000
3. Graphite-reflected Oy(93.2) metal annulus, 8.89-cm i.d., 13.97-cm o.d., 6.88-cm high 17.78-cm reflector. Graphite completely surrounds metal.	9	0.982 ± 0.021 0.988 ± 0.015	1700 4000
4. Water-reflected Oy(93.5) metal hemispherical shell, 25.4-cm i.d., 30.48-cm o.d., > 15-cm reflector. Water completely surrounds metal.	2	0.998 ± 0.014 1.010 ± 0.010	1500 4000
5. Unmoderated, unreflected 8-unit cubic array of 15.7 kg Oy(93.2) metal cylinders. Each unit is a 9.12-cm diam $\times$ 4.32-cm high cylinder between two 11.49-cm diam $\times$ 2.69-cm high cylinders. Surface separation of units = 2.61 cm.	10	1.017 ± 0.015 1.011 ± 0.013	1700 4000
6. Plexiglas-moderated, paraffin-reflected 8-unit cubic array of 21.0 kg Oy(93.2) metal cylinders. Each unit is 11.57-cm diam $\times$ 10.76-cm high. The moderator is 1.27-cm thick. Surface separation of units = 8.61 cm.	10	1.011 ± 0.021 0.998 ± 0.017	1700 4000

^aOy(x) indicates uranium enriched to x percent in U-235.

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Calculated Nuclear-Safety Parameters of Low-Enriched Uranium/ Kenneth R. Ridgway (Isochem)

To provide data for scoping, designing, and modifying chemical-operation facilities at Hanford, a computational survey was made to obtain K_{∞} and material buckling at varying H/U atomic ratios of uranyl-nitrate (UNH) and uranium-trioxide (UO₃) water mixtures enriched between 1.5 and 6 wt% U-235. Multigroup diffusion theory was utilized with the multigroup constants obtained from the GAMTEC-2 computer code¹. Fast-group constants were averaged over a 64-group slowing down spectrum computed from a B₁ approximation to the Boltzmann equation using U-235 fission-spectrum weighted source term. The one-thermal-group constants were averaged over a Wigner-Wilkins spectrum using cross-section data from the Battelle-Northwest Master Library². Resonance absorption by U-238 was treated by the Adler-Nordheim method³.

A special averaging technique in GAMTEC-2 computes two-group constants (one broad fast group) consistent

TABLE I
Homogeneous Uranium Trioxide-Water

Weight % U-235	H/U	Uranium Density g/liter	K_{∞}	Material Buckling cm^{-2}
2.0	5.2	3 130	1.2045	0.006399
	7.0	2 577	1.2069	0.006670
	10.0	1 994	1.1857	0.006103
	20.0	1 137	1.0693	0.002295
4.0	8.0	2 347	1.3901	0.013051
	10.0	1 994	1.3928	0.013252
	20.0	1 137	1.3382	0.011446
	100.0	256	0.8363	-0.005388
6.0	8.0	2 347	1.4650	0.015825
	10.0	1 994	1.4772	0.016376
	20.0	1 137	1.4612	0.015881
	100.0	256	1.0326	0.001073

with multigroup constants. As a typical example, the K_{eff} for a large 2 wt% U-235-enriched UNH system was computed to be 0.977 and 0.980 utilizing 18- and 2-group theory, respectively. The 2-group calculations also compare quite well with experimental data. Computed critical bucklings for uranium tetrafluoride (2 wt% U-235 in paraffin) were within 1.0% of the experimental values⁴ at optimum moderation, with the largest difference being only 4% for the undermoderated region. Also, the K_{∞} for 1.034 wt% U-235 $\text{UO}_3\text{-H}_2\text{O}$ was computed to be 0.997 compared to 1.000 ± 0.003 determined experimentally⁵. For 3.04 wt% U-235 uranyl nitrate in polyethylene at an H/U atom ratio of 10, k_{∞} was calculated to be 1.138 as compared to the experimental value⁶ of 1.145 ± 0.010 .

From the calculated parameters it can be seen that 2 wt% uranium in the form of uranyl nitrate cannot support a chain reaction as long as the nitrate ion is present. The methods possible by which the nitrate ion and uranyl ion can be separated are precipitation and calcination. Calcination can be prevented by limiting process temperature. A laboratory study was undertaken by C. A. Colvin, Isochem Inc., Research and Engineering Section, to determine the maximum uranium concentration that could be obtained in the precipitates from uranyl nitrate.

TABLE II
Homogeneous Uranyl Nitrate-Water

Weight % U-235	H/U	Uranium Density g/liter	K_{∞}	Material Buckling cm^{-2}
2.0	4.0	1 880	0.9354	-0.000738
	8.0	1 600	0.9810	-0.000320
	10.0	1 475	0.9765	-0.000440
	14.0	1 190	0.9487	-0.001138
4.0	8.0	1 600	1.2161	0.003702
	10.0	1 475	1.2262	0.004340
	14.0	1 190	1.2203	0.004953
	70.0	327	0.9105	-0.002804
6.0	10.0	1 475	1.3397	0.006610
	12.0	1 330	1.3483	0.007443
	14.0	1 190	1.3487	0.007943
	34.0	608	1.2792	0.008404
	70.0	327	1.1033	0.003246

Reagents common to the Hanford aqueous separation process such as NaOH, KOH, NaCO_3 , and zirconium-coating removal wastes (NH_4ZrF_6 , NH_4F , and NH_4NO_3) were used in the studies. Precipitates formed under varying conditions were analyzed for uranium content and overall density. The maximum uranium density in precipitates was 905 g/liter obtained by centrifuging a sodium polyuranate precipitate ($\text{Na}_2\text{U}_2\text{O}_7$) formed at a temperature of 70° after rapid addition of 50% NaOH to a 2M uranium solution.

Two percent enriched uranium in the $\text{UO}_3\text{-H}_2\text{O}$ system at a uranium concentration of 905 g/liter has a calculated K_{∞} of 1.008.

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Critical Experiments

Critical Experiments with Highly Burned-up Plutonium in Heavy Water/R. L. Olson, W. E. Graves (duPont-SRL)

Plutonium with large fractions of Pu-240, Pu-241, and Pu-242 was prepared as an intermediate step in a Savannah River program to produce 3 kg of Cm-244. This material afforded an opportunity to investigate the reactor physics of the higher Pu isotopes by comparing plutonium with a wide range of isotope compositions in D₂O-moderated criticals in the Process Development Pile (PDP).

Two plutonium compositions were used, as listed in Fig. 1, which also shows the designs of the coextruded aluminum fuel assemblies. The experimental lattices consisted of 19 of these fuel assemblies in hexagonal arrays at 12.12- or 14.00-in. pitches. To provide an easily calculated radial boundary, the fuel lattices were extended for two additional rings with nonfissioning iron-lithium assemblies having very nearly the same absorption cross section as the fuel.

The experiments consisted primarily of critical moderator height measurements and flux traverses to determine the lattice bucklings, B_m^2 at various uniform temperatures between 20 and 50°C. The lattices had a bottom reflector composed of D₂O and the aluminum extensions of the fuel tubes. This reflector could be ignored in determining the axial buckling from the flux traverses by choosing data sufficiently far from the boundary. This procedure was checked by calculations of reflector savings using the FLOG four-group code¹ with cell parameters calculated with the HAMMER code¹.

The determination of the isothermal temperature coefficients required the estimation of temperature effects on radial buckling, and on bottom extrapolation distance. The effect on B_r^2 was calculated with the SRL-HERESY code². The change in the bottom extrapolation distance was calculated with the FLOG code. These corrections are listed with the experimental results in Table I.

The primary purpose of the experiments was to permit comparisons with parameters calculated by the HAMMER code. These comparisons, shown in Table I, indicate only fair agreement in the estimates of B_m^2 and an overestimate of the temperature coefficients in the heavy isotope lattices. Plutonium cross sections used in the calculations are being reexamined to determine if changes within the cross-section uncertainties can match the experimental results.

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Type I Fuel

Dimension, inches		Isotope	Per cent of Total Pu	Content, g/lb
Outer housing, OD	3.420			
Outer housing, ID	3.300			
Clad outer fuel, OD	3.020	239Pu	22.9 ± 1.0	39.6
Bare outer fuel, OD	2.960	240Pu	49.6 ± 0.5	85.9
Bare outer fuel, ID	2.796	241Pu	14.8 ± 0.4	25.6
Clad outer fuel, ID	2.736	242Pu	12.7 ± 0.4	21.9
Clad inner fuel, OD	2.354	Total		173.0 ± 2.0
Bare inner fuel, OD	2.294			
Bare inner fuel, ID	2.108			
Clad inner fuel, ID	2.048			
Inner housing, OD	1.740			
Inner housing, ID	1.640

Type II Fuel

Dimension, inches		Isotope	Per cent of Total Pu	Content, g/lb
Outer housing, OD	3.420			
Outer housing, ID	3.300			
Clad outer fuel, OD	3.020	239Pu	91.0 ± 0.05	27.21
Bare outer fuel, OD	2.960	240Pu	8.2 ± 0.05	2.45
Bare outer fuel, ID	2.796	241Pu	0.7 ± 0.05	0.21
Clad outer fuel, ID	2.736	242Pu	0.1 ± 0.05	0.03
		Total		29.9 ± 0.3

Fig. 1. Fuel Dimensions and Specifications.

TABLE I
Experimental Results

Fuel Type	I	I	II
D ₂ O Purity, mole % D ₂ O	99.60	99.58	99.55
Pitch, in.	14.00	12.12	14.00
B_m^2 , m ⁻²	3.39 ± 0.05	3.93 ± 0.06	4.51 ± 0.05
B_r^2 , m ⁻²	5.35 ± 0.16	7.22 ± 0.22	5.62 ± 0.20
B_m^2 , m ⁻² (at 20°C)	8.74 ± 0.17	11.15 ± 0.23	10.13 ± 0.21
dB_m^2/dT , m ⁻² × 10 ² /deg C	-0.131 ± 0.006	-0.266 ± 0.013	-0.876 ± 0.045
Radial Reflector Correction, m ⁻² × 10 ² /deg C	-0.083	-0.121	-0.122
Bottom Reflector Correction, m ⁻² × 10 ² /deg C	-0.050	-0.033	-0.048
dB_m^2/dT , m ⁻² × 10 ² /deg C	-0.26 ± 0.05	-0.42 ± 0.04	-1.04 ± 0.07
B_m^2 , m ⁻² , calculated ^a	9.64 ± 0.07	11.55 ± 0.07	10.87 ± 0.06
dB_m^2/dT , m ⁻² × 10 ² , calculated	-0.539	-0.658	-1.140

^aIncludes only effect of change in pile top boundary.

^bError flag accounts for uncertainty in experimental fuel content.

Critical Parameters of Uranium (1.95) Metal Cylindrical Annuli/E. B. Johnson (ORNL)

Experiments have been performed at the ORNL Critical Experiments Facility to determine the critical parameters of lattices of cylindrical annuli of uranium enriched to 1.95% in U-235.

In the first series of experiments, the dimensions of the units were 7.2-in. o.d., 2.6-in. i.d., and 40-in. long, with an average mass of 442 kg, of which 8.6 kg was U-235. A total of 25 units was available. For the next series, the o.d. was reduced to 6.2 in., which decreased the mass of a unit to 308 kg (6.0 kg of U-235); the other dimensions were unchanged.

Fig. 1. Critical Lattices of U(1.95) Metal Annular Cylinders Moderated and Reflected by Water.

The units were latticed in water in both square and triangular patterns to determine the critical number as a function of spacing. With the units that were 7.2-in. o.d., it was found that the minimum for each pattern occurred at a water-to-uranium volume ratio of 0.65, and that the critical number of units arranged in a triangular pattern was half that required for a square pattern, as shown in Fig. 1. For the 6.2-in.-o.d. units, the minimum was found at a water-to-uranium volume ratio of 0.8, and the difference between the critical number of units arranged in square and triangular patterns was appreciably reduced.

In several critical lattices, water was excluded from the 2.6-in.-diam central region of each annulus by inserting coaxial Styrofoam cylinders (density = 0.024 g/cm³) of three different diameters. In all cases, exclusion of water from the central region decreased the reactivity. For example, the presence of a 1.55-in.-diam Styrofoam cylinder in each annulus of a lattice increased the critical number of units from 10 to 12.5, and shifted the position of the minimum volume ratio from 0.65 to 0.60 at a surface separation of 1 in. in triangular pattern.

Fast Reactor Physics

Time-of-Flight Measurement of the Neutron Spectrum from a Spherical Uranium-235 Fast Reactor/A. E. Profio, H. M. Antunez, K. L. Crosbie, J. L. Russell, Jr. (GA)

The time-of-flight method was used to measure the neutron spectrum from the Accelerator-Pulsed Fast Assembly (APFA), which is a 7-in.-diam sphere of 93%-enriched uranium similar to the GODOVA reactor. The reactor was subcritical with a prompt multiplication of 18, giving a prompt-neutron dieaway time constant of about 112 nsec. The reactor is controlled by separation into three segments; for this experiment, a 0.5-in. gap was introduced between the upper third and the remainder of the core. Neutrons were produced near the center of the sphere by (γ, n) reactions resulting from 30-MeV electrons striking the uranium at the bottom of a 1-in.-diam by 3-in.-deep re-entrant hole. The electron beam, from the General Atomic Linear Accelerator, was pulsed at 20 ns and 360 pulses/sec. The detectors used were a 2×2 -in. liquid scintillator for energies above 1 MeV, and a B_4C-CH_2 capture slab and gamma-ray sensor. A 1-in.-diam collimated beam from the surface was allowed to drift down a 50-m evacuated flight path. The flight distance and dieaway time results in a 15% energy resolution at 10 MeV, and somewhat better resolution at lower energies. The spectrum was obtained from 30 keV to 10 MeV with an estimated error of $\pm 10\%$.

Measurements were made at different positions and angles on the surface. It was found that the angular distribution of the leakage flux was cosine, and the spectrum was essentially independent of angle. The spectrum is plotted in Fig. 1. For comparison, a GAPLSN S_N transport-theory calculation, normalized at a few MeV, is shown. The calculation was carried out for a spherical critical reactor. The calculated spectrum has fewer low-energy neutrons than the measured one. Also shown for comparison is the photoplate measurements of Frye et al.¹, on the critical GODOVA reactor. The photoplate spectrum has an even greater excess of low-energy neutrons.

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Fig. 1. Neutron Spectrum of a Spherical U-235 Fast Reactor.

Power and Propulsion Systems Technology

Cavity Reactor Critical Experiment, J. H. Lofthouse, G. D. Pincock, J. F. Kunze, R. E. Wood (GE-Idaho), R. E. Hyland (NASA-Lewis)

A critical experiment is being performed on a reactor consisting of a cylindrical moderator tank with a central cavity containing dilute uranium. Critical masses, power distributions, reactivity worths of mockup heat shields, holes in the moderator tank, and the effects of varying fuel distribution are being studied. The reflector thickness and the cavity size are substantially larger than those in early experiments of this type^{1,2}, and the fuel, instead of lining the cavity wall, has been dispersed throughout the specified volume of the cavity in the form of 0.001-in.-thick sheet material.

A preliminary experiment was performed in a small (7- $\frac{3}{4}$ liter) cavity container placed in a relatively isotropic thermal flux of a conventional reactor. A gaseous (UF_6) cavity was compared with a uniform dispersion of the above-mentioned fuel sheets (93.2% U-235). For equivalent reactivities, the U-235 mass in the fuel sheets was 6% greater than in the gas, and this value is approximately the same as the thermal-flux self-shielding effect for such fuel sheets.

The cavity reactor has 35-in.-thick radial and end reflectors of D_2O . The cavity wall is $\frac{3}{8}$ -in. Al on the radius and $\frac{1}{2}$ -in. Al on the ends. Inside dimensions of the cavity are 4-ft. long by 6-ft. diam. The fuel sheets are supported on corrugated aluminum screens within the cavity. A total of 75 kg of aluminum and 273 g of manganese impurity form the structure in the cavity. There are 764 kg (1.0% volume fraction) of structural aluminum inside the D_2O region. The heavy water contains 0.22% H_2O . One end reflector contains 36 holes (void), $\frac{3}{8}$ -in. diameter, for control rods. This void represents 1.0% of the volume of an end reflector.

Measurements have been made with several different fuel-region diameters, concentric within the cavity and of full cavity axial length. Figure 1 shows the fuel mass and fuel density vs fuel radius for the configuration described above, with an adjusted multiplication factor of 1.02. The actual configurations that were measured precisely had a loading that gave zero excess reactivity. However, a gap in the reflector was formed by the meeting of the split-table assembly on which the experiment was mounted. This gap accounted for a 2% Δk difference in multiplication between the actual and the ideal configurations. Experiments were also conducted to evaluate the effect of fuel arrangement and fuel sheet orientation within the fuel volume, and it is estimated that the optimum arrangement of fuel sheets would have increased the reactivity approximately 6% Δk on the configuration with the largest fuel radius.

The worth of a one-foot-diameter central hole in one end reflector was measured to be 0.76% Δk . In the most dilute core (largest fuel radius) the peak power (at the "corner" of the fuel) was a factor of 1.3 above the core-center power. In the core with a fuel radius of 19.1 in. (0.53 of the cavity radius), the peak power was 3.4 above the core center power. The fraction of fissions occurring above the cadmium cutoff for 0.020-in.-thick cadmium disks (nominally 0.45 eV) was 4.2% and 7.1%, respectively, for these two configurations.

1. C. B. MILLS, *Nucl. Sci. Eng.*, 13, 301 (August 1962).
2. G. JARVIS and C. BYERS, AIAA Propulsion Joint Specialist Conf., Paper 65-555, Colorado Springs (June 14, 1965).

Fig. 1. Fuel Radius Effects on Fuel Mass and Density for an Adjusted Multiplication Factor of 1.02.

Fast Reactor Measurements

Fast-Spectrum Refractory-Metal Critical-Experiment Measurements, F. L. Sims, J. F. Kunze, W. P. Walsh (GE-Idaho), W. B. Henderson (GE-NMPO)

A fast-spectrum refractory-metal critical experiment with a moderating reflector has been operated in Idaho to determine the basic nuclear characteristics of this type of reactor. This paper describes the measurements that have been performed with this critical experiment.

The configuration on which the experiments were performed is shown in Fig. 1. The core was composed of 0.156-in.- and 0.170-in.-diam rods in an hexagonal aluminum structure giving a composition of 26.8% metallic uranium (93.2% U-235), 26.2% tungsten, 10.2% tantalum, and 6.9% aluminum. The radial reflector was 88% beryllium and 6.9% aluminum. The front reflector was a mixture of aluminum, tantalum, and beryllium, with a total volume fraction of materials of nominally 83%. The

inner ring of beryllium was worth 4.2% Δk in reactivity. Replacing the beryllium with beryllium oxide increased the system reactivity by 2.2% Δk .

Neutron lifetime measurements were made using both the Rossi-alpha technique and the static $1/v$ -poison technique. Three separate Rossi-alpha decay components were detected. When combined into an equivalent $1/v$ lifetime, the result is very close to that measured by incrementally poisoning typical regions throughout the reactor with boron-10. With equivalent uniform poison density, the radial reflector contributes 97% of the reactivity effect, the front reflector 1%, and the core the remainder. The resulting lifetime is $4.6 \pm 0.4 \mu\text{sec}$. The effective lifetime deduced from Rossi-alpha results was $4.7 \pm 0.8 \mu\text{sec}$. A two-dimensional r - z S_4 transport calculation with dilute $1/v$ poisoning (0.4% Δk for entire reactor) gave a neutron lifetime of $4.0 \mu\text{sec}$ while a one-dimensional radial S_8 transport calculation gave $4.65 \mu\text{sec}$ by perturbation theory.

Reactivity coefficient measurements were made with core-length samples of various materials. The results are listed in Table I, along with the calculated values. The latter were obtained by perturbation theory from a one-dimensional axial S_8 transport calculation. All materials measured, except Ta, Re, and B had positive longitudinally averaged worths near the core axis. This is indicative of the overwhelming scatter-leakage effect

Fig. 1. 710 Basic Critical Experiment Configuration Dimensions in Centimeters.

median fission energy in the core was calculated to be 0.4 MeV.

Experiments on the worth of the radial reflector showed that it was nearly infinite, i.e., an increase in thickness by 8 cm (50% increase) increased the system reactivity by only 1.8% Δk . However, reactivity control by removal of reflector bars from the inner ring adjacent to the core was quite effective. Removal of 45% of the

TABLE I
Worth of Core-Length Materials Near the Core Axis

Material	Worth, Percent $\Delta k/k$ per g	Axial S_8 Calculation	Error, Percent
Oralloy	7.00-4	7.00-4 ^a	---
Hydrogen	2.06-2	2.39-2 ^b	+16
Beryllium	1.02-3	1.26-3 ^b	+24
Aluminum	1.94-4	2.16-4 ^b	+11
Iron	9.82-5	1.12-4 ^b	+14
Nickel	4.86-5	5.41-5 ^b	+11
Manganese	1.01-4	1.18-4 ^b	+17
304 SS	8.58-5	1.02-4 ^b	+19
Tantalum	-4.67-5	-4.21-5 ^b	+10
B-10	-9.0-3	-6.73-3 ^c -9.07-3 ^d	+25 -1
W	1.37-5	2.84-5 ^b 1.26-5 ^e	+107 -9
Mo	6.32-5	4.17-5 ^b 6.62-5 ^e	-34 +5
Nb	4.65-5	4.11-5 ^b 6.41-5 ^e	-12 +38

^aNormalized to measured value. Normalization factor = 1.423 vs measured local-to-average Oralloy worth of 1.35.

^bNMPO 16-group cross-section set¹ was used. A small positive bias is expected due to the two-dimensional effect of the radial reflector extension.

^cNMPO 16-group B-10 was used.

^dSame as c except α_n values above 80 keV, where correction suggested by D. Bogart² was used.

^eBondarenko cross-section set³ was used.

with respect to absorption for most materials. The measured core-average alloy worth per fractional fuel mass ($\Delta k/k$) / ($\Delta m/m$) was 0.58, approximately that which would be expected for a reactor with virtually no other absorbing material, a consequence of the fast spectrum over much of the core. The calculated value was 0.52.

Other measurements included fission ratios which, at the core center with respect to U-235, were U-238 (0.083), Np-237 (0.56), Pu-239 (1.20), Pu-240 (0.52), Th-232 (0.022), and U-233 (1.56) using 1-in. o.d. absolute fission chambers of the gas flow type. A noticeable softening of the spectrum was observed near the front reflector, where the U-238/U-235 ratio drops to one-half of its core-center value. A minor spectral hardening increased the ratio 10% near the rear of the core. Satisfactory correlation of the calculated fission ratios requires inelastic-scatter-transfer cross sections based on individual-level excitation cross sections below about 2 MeV in W and Ta.

Cadmium ratio measurements using Au, In, Mn, Ta, and U-235 were made at several radial locations in the radial reflector. Conversion to infinitely dilute values and fitting of the results to a multigroup flux format gave a reflector energy spectrum 4 to 5 cm from the core interface of $\phi(E) = \text{Constant} \times E^{-\frac{1}{2}}$. At this location, the infinitely dilute U-235 cadmium ratio (bare/Cd-covered) was 5.5; for gold, 1.23. The effect of the thermal (subcadmium) flux component on the reactor multiplication was measured by placing a 0.025 in.-thick cadmium shield between the reflector and core; its worth was $-2.7\% \Delta k$. Satisfactory correlation of cadmium ratios was obtained using fluxes from one-dimensional radial S_4 calculations, although, for manganese, an epicadmium absorption integral of 11.3 b is implied by the calculated flux and the measured cadmium ratios.

1. H. L. BERMANIS, W. B. HENDERSON, and C. S. ROBERTSON, Jr., "Nine and Sixteen Group Cross Sections for Reactor Analysis," GE-NMPO, GEMP-173, Supplement No.1, General Electric Company (September 29, 1965).
2. D. BOGART, "Boron Cross Sections as a Source of Discrepancy for Capture Cross Section in the Kev Region," NASA TM X-52162, National Aeronautics and Space Administration (1966).

3. I. I. BONDARENKO, et al., *Group Constants for Nuclear Reactor Calculations*, Consultants Bureau, New York (1964).

Sensitivity of Neutron-Spectrum-Dependent Quantities in Uranium (93.2% U-235) Metal Assemblies to Inelastic Scattering Models for U-235, J. T. Mihalcz (ORNL)

The existing discrepancy between the measured and calculated neutron lifetime in fast critical assemblies has motivated a study of the sensitivity of this and other quantities dependent on the neutron energy spectrum to the inelastic scattering properties of U-235. The energy distributions of neutrons after one, two, and three successive inelastic collisions with the U-235 nucleus have been determined by using the evaporation model at all energies and by using a combination of the evaporation model and the level excitation model. In one variation of the latter, the calculated cross sections of Yiftah, Okrent, and Moldauer¹ (YOM) were used in the excitation level model up to 2.3 MeV; in another variation the cross sections of Parker² were used in level excitation up to 2.0 MeV. The evaporation model was used for higher energies in both cases. The source neutrons for these scattering studies were assumed to have a Maxwellian spectrum with a mean temperature corresponding to that resulting from fissions produced by neutrons having an energy of 1 MeV. The results of these three calculations are summarized in Table I.

The effect of these variations on a number of characteristics of the Godiva I critical assembly have been calculated by a Monte Carlo code and are summarized in Table II. The effect of a 30% decrease in the nuclear temperature used in the evaporation model was also investigated. The results are expressed as differences, in percent, between the values obtained from using the evaporation model at all energies at the higher temperature and the values obtained by each of the other variations.

1. S. YIFTAH, D. OKRENT, and P. A. MOLDAUER, *Fast Reactor Cross Sections*, Pergamon Press, London, England (1960).

TABLE I
Average Energy of Neutrons Inelastically Scattered from U-235

Method	Energy After			Energy of Source Neutrons (MeV)
	First Collision (MeV)	Second ^a Collision (MeV)	Third ^a Collision (MeV)	
Evaporation model at all energies	0.461	0.184	0.089	1.975
Level excitation model with YOM ¹ cross sections below 2.3 MeV; evaporation model above	0.689	0.486	0.357	1.968
Level excitation model with Parker ² cross sections below 2.0 MeV; evaporation model above	1.418	1.068	0.810	1.964

a) The values here refer to the distribution after two or three successive inelastic collisions.

TABLE II
Sensitivity of Neutron Spectrum-Dependent Quantities in
Godiva I to Inelastic Scattering Models for U-235

Quantity	Percentage Change from Value Calculated by the Evaporation Model ^a		
	YOM Cross Sections ^b	Parker Cross Sections ^b	Evaporation Theory with Temperature Reduced 30%
Time-to-loss by absorption and leakage	-5.7	-10.2	+ 6.7
Time-to-loss minus escape flight time	-7.4	-12.3	+ 8.2
Generation time	-9.0	-13.6	+10.6
Number of collisions before loss	-1.9	- 4.2	+ 1.8
Average energy of neutrons in core	+7.9	+12.2	-14.0
Average energy of the leakage flux distribution	+1.4	0.3	~ 0
Multiplication factor	-0.13	- 1.8	+ 0.7
Central spectral indices σ_f (U-235)/ σ_f (U-238)	-4.3	- 2.7	---
σ_f (Np-237)/ σ_f (U-238)	+3.0	+ 8.3	---
Probability of leakage of fission neutrons	+2.8	+ 2.4	- 0.9

a) The nuclear temperature of U-235 in the evaporation model was 0.36 MeV.

b) The calculational methods are described in the first column of Table I.

2. K. PARKER, "Neutron Cross Sections of U-235 and U-238 in the Energy Range 1 keV to 15 MeV, Part I.," AWRE 0-82/63, Atomic Weapons Research Establishment, Aldermaston, England (December 1963).

Lattice Measurements and Calculations

Organic-Cooled Heavy-Water-Moderated U-233-Fueled Lattice Experiments, G. A. Price, H. Windsor, W. Tunney, E. Hellstrand (BNL)

To provide data for evaluating the U-233-Th cycle in organic-cooled heavy-water-moderated reactors, measurements of material buckling and thermal-neutron distributions were made in several exponential assemblies. Fuel elements were composed of 31 and 24 "Kilorod" fuel pins (97 wt% ThO₂ + 3 wt% U²³³O₂, 8.96 g/cm³ density, Zircaloy-2 clad, 0.499-in. o.d., 0.430-in. i.d.) on 0.6767-in. pitch in a stainless-steel canister (4.509-in. o.d., 4.265-in. i.d.). The 24-pin element contained a solid aluminum plug 1.755-in. in diam. Dowtherm-A was used as the coolant in the fuel element. The stainless-steel canister served to reduce the material buckling. Otherwise, the material buckling would have been too large to permit meaningful measurements to be made in these cluster geometries.

Values of material buckling were derived from indium foil activations along the three principal axes of the assembly which was 42.5-in. square by 42-in. high. Thermal-neutron flux activations were observed by dysprosium and lutetium foils in the fuel pins, organic coolant, and heavy water. Table I gives the experimental results, as well as calculations using the METHUSELAH reactor code¹. This code was chosen to monitor the experimental results because of its convenience in handling cluster geometries.

Some preliminary comparisons of uniform lattices were made against HAMMER² to test the METHUSELAH library. The METHUSELAH results turned out to be considerably in error, due primarily to incorrect input data for the epithermal absorption and fission in U-233. New group cross sections for the three fast groups were prepared, based on Feiner and Esch's³ recommended data for the fission integral and the epithermal α .

In a new test run on a uniform lattice with the new U-233 cross sections, METHUSELAH gave a k_{eff} of 1.020 for a case in which the geometrical buckling was assumed to be equal to the material buckling that had been determined by an exponential experiment. No simple and justifiable change could be made in the METHUSELAH library to improve the results. Consequently, it is expected that discrepancies of the order of several percent in k_{eff} can be expected for the cluster lattices.

It is noted that the METHUSELAH bucklings are larger than the observed bucklings by as much as 3 m⁻² which corresponds to an overestimate of k_{eff} of about 0.04. Although the calculated thermal-neutron disadvantage factor $\bar{\phi}_m/\bar{\phi}_f$ is considerably less than that observed with dysprosium, it has only a slight effect on reactivity. The differences between calculations and experiments are probably due to an underestimate of the resonant neutron-

TABLE I

Lattice No.	Array	Fuel El	Pitch (in.)	Reflector	Coolant	D ₂ O Purity
1	4 × 4	31 pins	10.65 in.	boral	Dowtherm A	99.76%
2	4 × 4	31 pins	10.65 in.	boral + paraffin	Dowtherm A	99.75%
3	5 × 5	31 pins	8.50 in.	boral + paraffin	Dowtherm A	99.72%
4	5 × 5	31 pins	8.50 in.	boral + paraffin	air	99.69%
5	5 × 5	24 pins + Al plug	8.50 in.	boral + paraffin	Dowtherm A	99.61%

Lattice No.	$\bar{\phi}_m/\bar{\phi}_f$ (Dy act) (exp)	$\bar{\phi}_m/\bar{\phi}_f$ (1/v) (METH)	$B_x^2 + B_y^2$ (m ⁻²)	1/L ² (m ⁻²)	B_m^2 (exp)	B_m^2 (99.8% D ₂ O)	B_m^2 (METH)
1	-	-	14.58 ±0.15	13.56 ±0.16	1.02 ±0.22	1.07 ±0.22	4.1
2	4.65	3.89	14.80 ±0.18	14.15 ±0.13	0.65 ±0.22	0.71 ±0.22	4.1
3	3.70	3.11	14.69 ±0.19	9.97 ±0.11	4.72 ±0.22	4.78 ±0.22	7.7
4	3.50	3.23	14.40 ±0.13	5.83 ±0.07	8.57 ±0.15	8.66 ±0.15	9.2
5	3.20	2.71	14.82 ±0.16	13.35 ±0.08	1.47 ±0.18	1.62 ±0.18	3.7

capture rate, since the epithermal-neutron capture rate in these clustered lattices is quite high while the leakage probability is relatively low.

1. R. ALPAIR, "Methuselah I, A Universal Assessment Programme for Liquid Moderated Reactor Cells, Using IBM 7090 or Stretch Computers," AEEW-R 135, United Kingdom Atomic Energy Authority (1964).
2. JOHN E. SUICH and H. C. HONECK, "Heterogeneous Analysis by Multigroup Methods of Epithermal Reaction," *Trans. Am. Nucl. Soc.*, **8**, 2, 511 (1965).
3. F. FEINER and L. J. ESCH, "Survey of Capture and Fission Integrals of Fissile Materials," *Reactor Physics in the Resonance and Thermal Regions*, Am. Nucl. Soc. Nat'l. Topical Mtg., San Diego (February 1966).

Aerospace Nuclear Systems

Experimental Results from Large Cavity Reactor Critical, R. E. Hyland (NASA-Lewis), G. D. Pincock, J. F. Kunze, R. E. Wood (GE-Idaho)

Various fuel (93.2% ^{235}U) arrangements have been mocked up in a large cavity reactor critical experiment at the Idaho Test Station. The cavity reactor, with overall dimensions of 12-ft diam and 10-ft long, contains a cylindrical cavity 4-ft long by 6-ft diam and is surrounded by 35 in. of D_2O (0.22% H_2O). The fuel in foil (0.001-in. thick) is dispersed in the cavity in various patterns representing flowing fuel as in a gas-core reactor powerplant concept. A previous report¹ presented data covering the uniformly distributed fuel experiment. The results to be covered in this paper are an extension of those results to determine critical masses, flux and power shapes, and comparisons with a gaseous system. The areas of investigation are indicated in Table I.

Hydrodynamic calculations for a coaxial flow cavity reactor have provided uranium fuel distributions which were then simulated in the cavity critical experiment. Three variable fuel loadings have been measured with resultant critical masses ranging between 20 and 27 kg. These are shown in Table I and compared to the uniformly loaded configurations. These fuel distributions had axial zones with different concentrations and different dimensions. Offsetting of the fuel region from the radial center has produced small negative reactivity changes when performed on uniform, cylindrical distributions (0.1% Δk /3-in. displacement). However, when the smallest radial region of the nonuniform fuel distribution was shifted the same amount, the measured effect was

60% of the above but positive, presumably caused by more exposure to axially streaming flux.

Neutron lifetime measurements, in addition to fuel worths at various positions, have led to some preliminary insight as to controllability of gas-fueled cavity reactor powerplants, i.e., there is no plateau in the zero-power transfer function, and the neutron lifetime (nominally 3 msec) is essentially inversely proportional to fuel loading, and apparently strongly dependent on fuel radius as well.

Miscellaneous measurements have been completed indicating a 4-in.-thick section of "reactor grade" BeO is worth 2.0% $\Delta k/k$ less than the same thickness of Be reflector within the D_2O region. (The differences between Be and BeO are very dependent on few ppm boron impurity.) The worth of hydrogen ($N = 2 \times 10^{21}$ atoms/ cm^3), when inserted as polystyrene, is (-12% Δk) when uniformly dispersed between fuel and reflector moderator.

Calculations using diffusion theory have generally predicted moderator and structural effects satisfactorily, but fail on predicting k . Two-dimensional transport calculations using multithermal groups have required unusually long convergence times. For this reason, additional experiments are being performed which will determine the relative difference between dispersed uranium foils and UF_6 gas in a full cavity reactor.

1. J. H. LOFTHOUSE et al., "Cavity Reactor Critical Experiment," GEMP-473, and its summary in *Trans. Am. Nucl. Soc.*, 9, 340 (1966).

TABLE I
Experimental Investigations in Cavity Critical

Series	Fuel Region	Radius Ratio	Reflector	Miscellaneous	Critical Mass kg of U (93% ^{235}U)
1	Uniform	0.4 - 0.9	D_2O (170°F)	D_2O (Temp)	Ref. 1 (11 to 25 kg)
2	Uniform	0.67	$\text{D}_2\text{O} + \text{Be}$ (Var. Position)		19 - 20
3	Variable	0.67 (Entrance)	$\text{D}_2\text{O} + \text{Be}$ (Selected) ^b		26 - 27
4	Variable (Off-Center)	0.67 (Entrance)	$\text{D}_2\text{O} + \text{Be}$ (Selected) ^b		
5	Uniform (Off-Center)	0.67	$\text{D}_2\text{O} + \text{Be}$ (Selected) ^b	Neutron Lifetime	+0.15 kg
6	Uniform	0.67	D_2O	Inlet Effects	$\Delta m = -1.25$ kg
7	Uniform	0.67	D_2O	Exit Effects	$\Delta m = -1.0$ kg
8	Uniform	0.67	D_2O	Structural Effects	a
9	Uniform	0.67	$\text{D}_2\text{O} + \text{BeO}$		$\Delta m = +3$ kg
10	Uniform	0.67	$\text{D}_2\text{O} + \text{Be}$	Hydrogen in Cavity	a

^aExtrapolations from small sector measurements cannot be validly made

^b10 cm of Be spaced 6.7 cm from inner cavity wall

Reactor Parameters II

Interacting Arrays of Containers with ^{233}U Solution, R. C. Lloyd, E. D. Clayton (BNW)

Subcritical neutron interaction experiments¹ were performed with bare and Lucite-reflected arrays of bottles of ^{233}U solution. The effect on criticality of adding Lucite moderator between the bottles was also studied. These experiments provide data for nuclear safety guidance in handling, storage, and shipment of this material, and for checking interaction calculations.

The ^{233}U was in the form of uranyl nitrate hexahydrate [$\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$] at a concentration of about 330 g ^{233}U /liter; the uranium was contained in three-liter polyethylene bottles. The molarity of the solution was about 0.53. The bottles were 17.75-in. high, 4.7-in. o.d., with a wall thickness of 0.100 in. The average solution height was about 11.5 in.—960 g U/bottle.

The bottles were placed in arrays on a remotely operated split-table device, using neutron multiplication techniques to estimate the number of bottles for criticality. Extrapolations were also made on spacing to determine the critical spacing for fixed-number square arrays of the bottles. Arrays were single tier, except in one experiment where a double-tier array was used.

In the reflected arrays, the Lucite was placed touching the outside surface of the bottles, i.e., boxing in the array. The thickness of the top and bottom Lucite reflector was 4.5 in., and the side reflectors were 6 in. A

low-density aluminum honeycomb was used to provide accurate spacing between bottles and to provide assurance of bottles remaining upright for safety. In the bare assemblies, stability of the outer bottles was maintained by aluminum frames which were magnetically mounted on a 0.03-in. steel base plate. Aluminum honeycomb was used to support the baseplate and reduce neutron reflection. The experimental results are presented in the following table.

An outer-bottle-surface to outer-bottle-surface spacing of 0.75 in. was determined for criticality of the double-tier 3×3 bottle bare array, which compares with 0.60-in. spacing for the 3×3 bottle single-tier array. The double-tier array had a spacing of 7.0 in. between the fuel of the upper and lower tiers. The critical number of bottles was much more sensitive to spacing in the bare arrays than in the reflected arrays.

The effect of moderation was determined by placing Lucite plates of different thickness between bottles of the arrays. This was done for the 2×2 bottle reflected array and for the 3×3 and 4×4 bottle bare arrays. The thickness of Lucite between bottles giving the largest reactivity increase in the reflected array was about one inch, and in the bare array about two inches.

1. R. C. LLOYD and E. D. CLAYTON, "Neutron Interaction Experiments With Bottles of ^{233}U Solution," Physics Research Quarterly Report April, May, June 1966, BNWL-340, Battelle Northwest (1966).

TABLE I
Interaction Data for Bottles of ^{233}U Solution

Exp No	Description	Config	No Bottles	Est Critical S-S Spacing* (in.)	Remarks
^{233}U -1	Bare Single Row	1×9	> 9	0	
^{233}U -2	Bare Double Row	2×3	6.1	0	
^{233}U -3	Bare Lattice	3×3	9	0.60	
^{233}U -4	Bare Lattice	4×4	16	1.16	
^{233}U -5	Refl Single Row	1×2	> 2	0	≈2.8 Bottles Table stopped at 0.9 in.
		1×3	< 3	0	
^{233}U -6	Refl Single Row	1×3	3	0.6	
^{233}U -7	Refl Lattice	2×2	4	2.18	
^{233}U -8	Refl Lattice/ $\frac{1}{2}$ -in. Mod	2×2	4	2.48	$\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -9	Refl Lattice/ $\frac{1}{2}$ -in. Mod	2×2	4	2.58	$\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -10	Refl Lattice/1-in. Mod	2×2	4	2.66	1-in. Lucite Mod Between Bottles
^{233}U -11	Refl Lattice/ $1\frac{1}{2}$ -in. Mod	2×2	4	2.50	$1\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -12	Refl Lattice	3×3	9	3.98	
^{233}U -13A	Bare/1-in. Lucite Mod	2×3	6.3	1.00	1-in. Lucite Mod Between Bottles
^{233}U -13B	Bare/1-in. Lucite Mod	3×3	9	1.60	1-in. Lucite Mod Between Bottles
^{233}U -14	Bare/ $\frac{1}{2}$ -in. Lucite Mod	3×3	9	1.17	$\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -15	Bare/ $1\frac{1}{2}$ -in. Lucite Mod	3×3	9	1.78	$1\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -16	Bare/ $1\frac{1}{2}$ -in. Lucite Mod	3×3	9	1.87	$1\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -17	Bare/2-in. Lucite Mod	3×3	9	1.90	2-in. Lucite Mod Between Bottles
^{233}U -18	Bare/2-in. Lucite Mod	4×4	16	2.50	2-in. Lucite Mod Between Bottles
^{233}U -19	Bare/ $2\frac{1}{2}$ -in. Lucite Mod	4×4	16	2.47	$2\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -20	Bare/ $1\frac{1}{2}$ -in. Lucite Mod	4×4	16	2.42	$1\frac{1}{2}$ -in. Lucite Mod Between Bottles
^{233}U -21	Bare—Double Tier	$3 \times 3 \times 2$	18	0.75	

*Outer-bottle-surface to outer-bottle-surface spacing

Critical Lattices of U(4.89) Metal Rods in Water, E. B. Johnson (ORNL)

The critical dimensions of lattices of U(4.89) [U(4.89) represents uranium enriched to 4.89 wt% in the ^{235}U isotope.] metal rods are reported. All of the available rods are 30-cm long and have diameters of 1.31, 2.07, and 2.49 cm. Lattices of 30- and 60-cm height have been assembled in both square and triangular patterns.

The optimum lattice spacing for each rod diameter has been established by varying the separation and determining the number of rods required for criticality at each of several spacings. Figure 1 shows the data for the 1.31- and 2.07-cm-diam rods; critical mass is plotted as a function of the average ^{235}U density in the lattice.

Clark¹ has calculated masses of critical spherical water lattices of U(5.00) metal rods for a variety of diameters and uranium concentrations (or lattice spacings). The results for 1.52- and 1.02-cm-diam rods were interpolated to allow the comparison with the measurements with 1.31-cm rods shown by curves A and B. of Fig. 2. Curves B and C were obtained from the data of Fig. 1 by equating bucklings using an extrapolation distance of 6.0 cm. No calculations of the larger diameter rods are available for comparison with curve C.

It has been reported² that a difference of as much as a factor of 2 was found between the critical masses of lattices of large cylindrical annuli of U(1.95) metal (18.3-cm o.d. and 6.6-cm i.d.) when arranged in water in square and triangular patterns at the same average ^{235}U

Fig. 1. Critical Mass of Lattices of U(4.89) Metal Rods in Water.

Fig. 2. Comparison of Calculated and Experimental Critical Spherical Lattices of Uranium Metal Rods in Water.

density, the triangular pattern having the lower mass. In the case of the U(4.89) rods, no consistent difference was observed between lattices of the 1.31-cm-diam rods arranged in the two patterns.

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Experimental Buckling Measurements with 2.1 wt% Enriched Uranium Tubes in Light Water, C. L. Brown, L. E. Hansen (BNW), H. Toffer (Douglas United Nucl)

Exponential and critical approach experiments with 1.002, 1.25, and 1.95 wt% ^{235}U enriched uranium tubes in light water were reported¹ in 1965. Although these experiments nearly doubled the criticality data available on tubular fuels, the enrichments and tube sizes covered were still too few for theoretical correlations and extrapolation to different tube sizes at higher enrichments. The experiments described in the present paper extend the criticality data to 2.1 wt% ^{235}U enriched uranium tubes of two sizes: 2.33-in. o.d., 1.77-in. i.d.; and 1.38-in. o.d., 0.63-in. i.d. Experimental results are also given of a special series of criticality experiments, which include two mixed lattices of 0.95 and 2.1 wt% ^{235}U enriched tubes, a poisoned lattice, and a lattice containing lithium-aluminate target rods. The results of the measurements are presented in Table I.

Material bucklings and extrapolation distances were obtained using standard exponential and critical approach techniques. The tubes were arranged in columns in a hexagonal pattern and several lattice spacings measured for each fuel. The 2.33-in.-o.d. fuel columns were 26.5-in. long, the same length as the fuel tubes; the length

of the 1.38-in.-o.d. fuel columns was 49 in., being comprised of seven 7-in.-long fuel elements in aluminum cans. Extrapolation distances were determined by measuring relaxation lengths in three assemblies of different diameters. In the critical approach experiments, the number of tubes for criticality was determined from the extrapolation of inverse multiplication curves for each lattice. Each lattice contained up to 96% of the number of tubes for criticality.

Exponential measurements with mixed enrichments were performed with 0.95 and 2.1 wt% enriched tubes in an alternating tube arrangement and in an alternating ring arrangement. Only the 2.2-in. lattice pitch was measured. Using the extrapolation distance measured for the 2.1 wt% tubes alone (6.68 cm), material bucklings were estimated to be 58.85 m^{-2} and 60.14 m^{-2} for the alternating tube and ring arrangements, respectively; compared to 29.3 m^{-2} for 0.95 wt% tubes alone, and 96.64 m^{-2} for 2.1 wt% tubes alone. It is interesting to note that for the mixed lattices, the estimated critical number of tubes in the alternating tube and alternating ring fuel arrangement are equal within the uncertainty of the experiment. Firm conclusions, however, cannot be drawn without measurements of additional fuel patterns and lattice spacings.

Measurements with 2.1 wt% enriched 2.33-in.-o.d. tubes containing 1.08-in.-o.d. $LiO_3-Al_2O_3$ targets were made in four lattices. The lithium aluminate contained 10.5 wt% lithium. The addition of the target rod to the 2.1 wt% tubes increased the critical mass by a factor of about 20.

The poisoned lattices consisted of a 6:1 ratio of 2.1 wt% enriched uranium 2.33-in.-o.d. tubes and 2.4-in.-o.d. cans containing ground Pyrex glass. Each can was surrounded by a ring of six uranium tubes. Only the 2.8-in. lattice pitch was measured and seven rings filled the lattice. The exponential measurement indicates a buckling of 34.04 m^{-2} and 140 fuel tubes for criticality (compared to 28.3 tubes for criticality in the 2.8-in. lattice without poison).

The HAMMER² code was used to compute material bucklings and critical masses. Calculations were made both with and without Seghal resonance parameters for ²³⁸U. The critical number of tubes for mixed and poisoned lattices were computed using HFN³ and EXTERMINATOR⁴. Multigroup constants (18 energy groups) were obtained using a modified version of the HAMMER code. The results of calculations completed to date are given in Table II.

TABLE I
Results of Critical Approach and Exponential Measurements
with 2.1 wt% Enriched Uranium Metal Tubes in Light Water

Lattice Spacing in.	Vw/Vu	Uranium Height in.	No. of Tubes for Critical ^a	Extrapolation Distance cm	Material Buckling m^{-2}
Uranium Tube: 1.768-in. i.d.; 2.329-in. o.d.—0.035-in. Zr Clad					
2.44	1.64	26.5	42.1	8.32	80.80
2.80	2.54	26.5	28.3	7.79	90.04
3.10	3.39	26.5	25.8	7.50	85.93
3.40	4.32	26.5	28.3	7.31	73.31
Uranium Tube: 0.633-in. i.d., 1.375-in. o.d.—0.055-in. Al Outside, Bare Inside					
1.95	1.57	49.0	49.1	7.15	96.67
2.10	2.04	49.0	42.3	6.80	96.69
2.20	2.36	49.0	38.8	6.68	96.64
2.40	3.04	49.0	37.5	5.86	93.84
2.60	3.78	49.0	42.6	5.73	78.70
Uranium Tube: 1.768-in. i.d.; 2.329-in. o.d.—0.035-in. Zr Clad LiO ₃ -Al ₂ O ₃ Target: 1.075-in. diam—0.130-in. Zr Clad					
2.44	0.86	26.5	--	--	< -4.00
2.8	1.77	26.5	--	9.84	6.39
3.1	2.61	26.5	--	9.08	14.88
3.4	3.55	26.5	--	8.14	9.82
Poisoned Lattice— $\frac{1}{2}$ Ratio Poison Columns to 2.1 wt% Uranium Tubes Uranium Tube: 1.768-in. i.d., 2.329-in. o.d.—0.035-in. Zr Clad Poison Column: 2.4-in.-diam Can Containing Ground Pyrex Glass					
2.8	2.75	26.5	> 55.6	24.2	34.04
Mixed Lattice—Alternate Tubes of 0.95 wt% and 2.1 wt% Enriched Uranium 2.1 wt% Tubes: 0.633-in. i.d.; 1.37-in. o.d.—0.055-in. Al Outside, Bare Inside 0.95 wt% Tubes: 0.488-in. i.d.; 1.37-in. o.d.—0.055-in. Al Outside, Bare Inside					
2.2	2.20	49.0	34.85 (0.95%) 35.85 (2.1%)	--	--
Mixed Lattice—Alternate Rings ^b of 0.95 wt% and 2.1 wt% Enriched Tubes 2.1 wt% Tubes: 0.633-in. i.d.; 1.375-in. o.d.—0.055-in. Al Outside, Bare Inside 0.95 wt% Tubes: 0.488-in. i.d.; 1.375-in. o.d.—0.055-in. Al Outside, Bare Inside					
2.2	2.19	49.0	36.0 (0.95%) 35.0 (2.1%)	--	--

^aEstimated from least-squares fit of inverse multiplication measurements.

^bInnermost seven tubes are 2.1 wt%.

TABLE II
Comparison of Calculations and Measurements

Vw/Vu	No. of Tubes for Criticality		Material Buckling, m^{-2}	
	Exp	Calc	Exp	Calc
2.1 wt% 1.375-in.-o.d., 0.633-in.-i.d. Tubes				
1.57	49.1	49.2	96.67	98.41
2.04	42.3	41.2	96.69	102.74
2.36	38.8	38.8	96.64	101.44
3.04	37.5	38.5	93.84	91.62
3.78	42.6	43.4	78.70	75.38
Mixed Lattice—Alternating 0.95 wt% and 2.1 wt% Tubes				
2.20	34.85 (0.95%)	35.2		
	35.85 (2.1%)	35.2		
Mixed Lattice—Alternating Rings of 0.95 wt% and 2.1 wt% Tubes				
2.19	36.0 (0.95%)	35.9		
	35.0 (2.1%)	32.4		

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2. Personal Communication, J. E. SUICH.
3. J. R. LILLEY, "Computer Code HRN-Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," HW-71545, Hanford Works (1961).
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Measurements and Analyses of Some ^{233}U O_2 - ThO_2 Water Lattice Experiments, Bal Raj Sehgal, H. H. Windsor, W. J. Tunney (BNL)

Measurements, interpretation, and analyses of a series of exponential water-moderated uniform lattice experiments with 3% ^{233}U - ^{232}Th oxide rods are reported in this note. The fuel is vibratorily compacted to a density of 8.96 g/cm³ in Zircalloy-2 tubes of 1.27-cm o.d. and 0.8-mm wall thickness¹.

Bucklings were measured by both the modified variable loading² and the flux shape techniques. The interpretation included smoothing of the measurements and use of calculations to correct for the nonasymptotic behavior of the lattices. The interpreted results are quoted as "Best" values in Table I.

The standard cadmium ratio technique was used for the measurement of ρ_{ex} , the ratio of epi- to subcadmium captures in ^{232}Th . The 27.4-day half-life ^{233}Pa produced in ThO_2 wafers was γ counted. The differential technique was used to determine ξ_{Dy} , the dysprosium thermal disadvantage factor. Dysprosium aluminum foils 1.19-mm diam $\times$ 0.127-mm thick were distributed throughout a unit cell.

The HAMMER Code³ is used in the analyses. This

code combines the codes THERMOS, ZUT, TUZ, FOG, and a modified MUFT in which a collision-probability cell calculation is done for the epithermal flux.

The thermal cross-section library is constructed to conform with the evaluated cross sections reported⁴ in BNL-325. The GASKET-FLANGE Kernel⁵ is used for hydrogen. The resonance parameters for ^{233}U and ^{232}Th are taken from BNL-325. The epithermal smooth cross sections for ^{233}U are constructed to normalize to the total fission and capture integrals recommended by Feiner⁶ and those for ^{232}Th are based on an analysis of ^{232}Th resonance integral⁷.

The results for k_{eff} at the "Best" B_m^2 values and the calculated and measured values for reflector savings λ are shown in Table I. The measured and calculated results for ρ_{ex} and ξ_{Dy} are shown in Table II.

It is seen that the experimental ξ_{Dy} is about 2% lower than the calculated values and that the k_{eff} increases with volume ratio and decreases with poison concentration. The latter trend is partially removed if the calculated values of ξ_{Dy} are reduced by 2%. The calculated ρ_{ex} and λ values are in good agreement with experiments. It seems, however, that the captures in the thermal range may be underestimated.

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TABLE I
Results for B_m^2 , k_{eff} , and Reflector Savings λ

Mod/Fuel Volume Ratio	H ₃ BO ₃ Concentration (mg/cm ³)	"Best" Values of Measured B_m^2 (m ⁻²)	Calculated k_{eff}	"Best" Values of Measured λ (cm)	Calculated λ (cm)
0.997	0.0	75.8 ± 1.0	1.0007	7.65 ± 0.20	7.79
	13.3	42.1 ± 0.5	0.9946	5.30 ± 0.10	5.47
	25.0	18.8 ± 0.5	0.9947	5.10 ± 0.10	5.11
1.384	0.0	85.9 ± 0.6	1.0059	7.30 ± 0.10	7.34
	7.82	50.6 ± 0.5	1.0043	5.60 ± 0.10	5.67
	22.9	3.65 ± 0.5	0.9993	5.10 ± 0.10	5.12
1.7134	0.0	88.7 ± 1.5	1.0126	7.10 ± 0.25	7.10
	4.74	60.4 ± 0.5	1.0051	5.70 ± 0.10	5.87
	18.88	-1.8 ± 0.2	1.0003	5.25 ± 0.15	5.18
2.1943	0.0	89.1 ± 1.0	1.0174	6.90 ± 0.10	6.85
	4.71	50.5 ± 0.2	1.0085	5.75 ± 0.05	5.78
	13.3	-2.1 ± 1.0	1.0029	5.50 ± 0.10	5.31
3.0043	0.0	85.3 ± 0.5	1.0158	6.60 ± 0.10	6.60
	4.99	26.5 ± 0.3	1.0082	5.85 ± 0.20	5.69
	7.82	1.99 ± 0.2	1.0019	5.60 ± 0.15	5.50
4.2722	0.0	69.1 ± 0.7	1.0168	6.45 ± 0.10	6.42
	2.68	22.6 ± 0.4	1.0068	6.0 ± 0.15	5.90
	4.96	-7.67 ± 0.2	0.9972	6.0 ± 0.05	5.58
6.8449	0.0	32.3 ± 0.2	1.0089	6.25 ± 0.05	6.31
	1.7	-10.8 ± 1.0	0.9984	6.40 ± 0.20	5.85
9.2747	0.0	-0.82 ± 0.3	1.0023	6.65 ± 0.05	6.36

TABLE II
Results for ζ_{Dy} and ρ_{b2}

Mod/Fuel Volume Ratio	H ₃ BO ₃ Concentration (mg/cm ³)	Measured ζ_{Dy}	Calculated ζ_{Dy}	Measured ρ_{b2}	Calculated ^a ρ_{b2}
0.997	0.0	1.219 ± 0.024	1.262	1.380 ± 0.042	1.401
1.384	0.0	1.257 ± 0.024	1.283	0.928 ± 0.038	0.955
1.7134	0.0	1.270 ± 0.024	1.299	0.754 ± 0.024	0.754
2.1943	0.0	1.302 ± 0.024	1.321	0.607 ± 0.026	0.581
3.0043	0.0	1.325 ± 0.024	1.354	0.435 ± 0.013	0.427
4.2722	0.0	1.379 ± 0.024	1.403	-	0.308
6.8449	0.0	1.430 ± 0.034	1.467	0.218 ± 0.008	0.211
9.2747	0.0	1.447 ± 0.034	1.511	0.170 ± 0.007	0.171

^a Calculated for Cadmium cutoff energy of 0.51 eV.

Experimental and Analytical Determination of the Reactivity Worths of ²³³U, ²³⁵U, ²³⁶U, and ²³⁷Np Samples in a Series of ²³⁵U-Graphite Critical Assemblies, R. G. Bardes, E. M. Gillette, R. J. Nirschl, R. C. Traylor (GA)

As part of an AEC program directed toward providing a secure technical foundation for the development of large High-Temperature Gas-Cooled Reactors, General Atomic has constructed a series of nearly homogeneous, enriched uranium-graphite critical assemblies having varying degrees of spectral hardness. The central reactivity worths

of a number of materials of interest to large HTGR systems have been measured in three of these cores to date.

The HTGR Critical Assembly consists of two aluminum honeycomb sections mounted on a split bed assembly. The fuel elements used in these assemblies consist of fully enriched uranium-graphite compacts loaded into aluminum tubes. The radial reflectors were constructed from cylindrical graphite columns; no axial reflector was present. The assemblies were designed to provide a simple cylindrical geometry, thus making an analytical model spatially separable. Therefore, in analyzing the experimental results, the major emphasis was on the

energy dependence of the problem. Table I shows the values of some of the pertinent core parameters. The nominal axial dimension is 180 cm.

The special elements used in the reactivity worth measurements consisted of graphite compacts containing the isotopes ^{233}U , ^{235}U , ^{236}U , and ^{237}Np loaded into sealed aluminum tubes. These elements were substituted in place of a special all-graphite element at the central location and the change in reactivity measured using calibrated control rods. Measurements were also made on a series of boron- and gold-loaded special elements that were used as standards.

The analysis of the experimental results was carried out using the one-dimensional multigroup diffusion code GAZE¹. The special elements were represented explicitly as the central zone in a 30-group radial calculation. The GAM² and GATHER³ codes were used to average the multigroup cross sections. The calculated reactivity worth of the special elements was derived from differences in the effective multiplication constant of a series of these GAZE calculations in which only the central zone was changed.

The experimental and analytical results for ^{235}U are shown in Fig. 1. Similar results for ^{233}U , ^{236}U , and ^{237}Np are shown in Table II. The cross sections used in analyzing these results are discussed in the references⁴⁻⁷.

The agreement in the cases of the boron and gold standards was generally very good, thus giving confidence in the methods used.

In view of the discrepancies between calculation and measurement in the case of the eigenvalues and reactivity coefficients of ^{235}U , some of the above results have been reanalysed using the ENDF/B file data⁸ for ^{235}U . To date, the results are very encouraging, the above discrepancies having been virtually eliminated; however, they are not yet complete.

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TABLE I
Core Parameters for the
C/ ^{235}U 2500, 1718, and 859 Cores

C/ ^{235}U Atom Ratio	$\bar{\sigma}_a$ (Boron) ($E < 2.38$ eV)	Equivalent Core Radius (cm)	Equivalent Reflector Radius (cm)	Critical Mass (kg)	Measured k_{eff}	Calculated k_{eff}
2500	487	64.0	71.5	23.2	1.018	1.013
1718	286	59.0	67.4	29.2	1.013	1.003
859	228	56.5	63.0	54.2	1.012	0.985

TABLE II
Experimental and Calculated
Worths of ^{233}U , ^{236}U , and ^{237}Np

Element	Element Loading ($\text{g}/\text{cm}^3 \times 10^3$)	C/U = 2500		C/U = 1718		C/U = 859	
		Expt ^a (\$)	Calc (\$)	Expt ^a (\$)	Calc (\$)	Expt ^a (\$)	Calc (\$)
^{233}U 1	0.625	0.071	0.074	0.053	0.054	0.033	0.034
2	1.688	0.181	0.192	0.135	0.141	0.079	0.087
3	5.901	0.560	0.601	0.415	0.446	0.243	0.276
^{236}U 1	0.155	--	--	--	--	--	--
2	0.448	-0.021	-0.022	-0.022	-0.021	-0.020	-0.020
3	1.799	-0.057	-0.071	-0.057	-0.078	-0.050	-0.076
^{237}Np 1	0.518	-0.089	-0.076	-0.077	-0.075	-0.064	-0.060
2	1.552	-0.248	-0.221	-0.227	-0.196	-0.185	-0.157
3	5.661	-0.796	-0.750	-0.736	-0.733	-0.589	-0.587

^a The experimental results have an estimated accuracy of $\pm 1\%$ within a minimum accuracy of $\pm \$0.001$.

Fig. 1. Worth of special ^{235}U elements.

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Burnup Analysis and Plutonium Calculations

Evaluation of Some Nuclear Calculational Methods for Use in Plutonium Recycle, J. Celnik, S. Kellman (UNC)

Criticality calculations have been performed for various hexagonal lattices of rods with a plutonium enrichment of 2 wt% in UO_2 (natural), ^{240}Pu isotopic ratios of 8 to 26% at moderator/fuel volume ratios of 1.5 to 9.8. The calculations, utilizing the FORM-TEAM-TEMPEST¹⁻³, LASER⁴, and THERMOPILE⁵ programs have been compared with data obtained from subcritical experiments at the Battelle Northwest Laboratory⁶.

For the slowing down problem, both FORM and LASER employ a standard multigroup solution of the Fourier-transformed transport equation in a homogeneous medium. Self-shielding factors, applied to the resonance cross sections of ^{238}U , are calculated by use of a resonance integral correlation⁷ which matches the measurements of Hellstrand⁸ for both UO_2 and U metal rods. There are only minor cross-section differences between codes.

For the thermalization problem, zero-dimensional TEMPEST-II calculates the neutron spectrum utilizing the Wigner-Wilkins distribution of neutrons scattered by a light moderator. Use is made of the TEAM program to compute group-averaged disadvantage factors, employing the integral transport-theory formulations of Amouyal and Benoist⁹ that are applied to the microscopic cross-section data in the TEMPEST calculation. LASER, utilizing the thermalization transport-theory program THERMOS calculates the spectrum in a one-dimensional lattice cell composed of cylindrical rod, cladding, and moderator. The THERMOPILE Monte Carlo code treats thermal-neutron transport in detailed three-dimensional geometries, employing the Nelkin scattering kernel in water.

From the results of the calculations given in Table I, it is seen that while LASER predicts criticality to within ± 1 to 2% for a wide range of lattice pitches, FORM-TEAM-TEMPEST calculates values of k_{eff} that increase with lattice pitch, varying from an underprediction of 1% at a moderator/fuel volume ratio of 1.5 to an overprediction of 4% at a moderator/fuel volume ratio of 9.8.

The softer thermal spectrum and the reduced flux depression in the fuel at the plutonium resonance energies calculated by TEAM-TEMPEST are evident from the consistently larger cell-averaged absorption and fission cross-section results. For the tight and intermediate lattices, however, the ratios of Σ_f/Σ_a are in good agreement. In looser lattices, use of either the Wigner-Wilkins or Wilkins spectrum in TEAM-TEMPEST yields similar results, in disagreement with those of both LASER and THERMOPILE.

Computed values of fractional absorption in the cell are listed in Table II. The codes use the same thermal-uranium and plutonium cross sections to within a few percent.

TABLE I
Criticality of Hanford 2 wt% Pu Criticals as Calculated by FORM-TEAM-TEMPEST and LASER

H/Pu	FORM-TEAM-TEMPEST			LASER		
	Atom Ratio ^a	K_{fast}	K_{th}	K_{eff}	K_{fast}	K_{th}
237		0.166	0.821	0.987	0.170	0.928
552		0.0844	0.936	1.020	0.0860	0.922
1568		0.0388	0.998	1.036	0.0395	0.952

^a7.65 at. % ^{240}Pu

TABLE II
Fractional Thermal Absorptions in Cell Occurring in Various Materials (0-1.855 eV)

Material	(H/Pu = 237)		(H/Pu = 552)		(H/Pu = 1568)	
	TEMPEST	LASER	TEMPEST	LASER	TEMPEST	LASER
Hydrogen	0.061	0.058	0.145	0.148	0.342	0.371
^{238}U	0.117	0.112	0.121	0.116	0.100	0.093
^{235}U	0.073	0.070	0.073	0.070	0.060	0.054
^{239}Pu	0.686	0.694	0.620	0.624	0.475	0.459
^{240}Pu	0.055	0.058	0.032	0.034	0.017	0.018
^{241}Pu	0.006	0.006	0.006	0.005	0.005	0.004

In tight lattices, where no large discrepancies in flux disadvantage ($\bar{g}$) factors occur, the absorption distributions calculated by TEAM-TEMPEST and LASER are about the same. However, in loose lattices where $\bar{g}(\text{fuel})$ computed by TEAM-TEMPEST is 10% greater than that of LASER fractional absorption in ^{240}Pu calculated by TEAM-TEMPEST is greater than that computed by LASER, while absorption in H is less. Neutron spectra, flux-averaged cross sections, and disadvantage factors calculated by use of FORM-TEAM-TEMPEST and LASER will be compared with results of THERMOPILE.

From these results, it appears that the homogenization treatment of TEAM-TEMPEST leads to incorrectly calculated ratios of flux in moderator to flux in fuel in very loose lattices, but is adequate for calculation of mixed-oxide systems in lattices representative of operating power reactors.

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Analysis of PuO_2 - UO_2 Critical Experiments, W. J. Eich (WAPD)

A series of critical experiments has been conducted at the Westinghouse Reactor Evaluation Center with plutonium-bearing fuel moderated by light water. The Zircaloy-clad fuel consisted of 2 wt% PuO_2 - UO_2 lent for the purpose of these experiments by BNWL¹. The uranium isotopic ratio corresponded to the natural proportion; two types of fuel were used which differed in plutonium isotopic ratios and which can be characterized by their ^{240}Pu content: 8 and 24%.

Analyses of these experiments have been performed using the same methods, codes, and cross-section libraries as had previously been applied to critical experiments involving uranium fuel. In this analytical approach, the radial plane of each critical assembly, including both core and reflector, is described explicitly

Fig. 1. Eigenvalues corresponding to ESADA plutonium critical experiments.

using the AIM-5 Code³. The fast group constants used for the reflector had been developed through a correlation with the P3MG Code⁴ following a procedure described by Eich and Williams⁴ and Eich⁵. Five neutron groups are used. The core constants for the top three groups are derived from the FORM Code⁶. The constants for group 4 and the thermal group were derived from the LASER Code⁷ edits for the range below 1.855 eV. The resonance capture rate in group 3 (5530 to 1.855 eV) for ^{235}U and ^{239}Pu were normalized to MONTE CARLO estimates⁸.

Two trends are apparent in the eigenvalue or k_{eff} results of the analyses of these critical experiments. Figure 1 displays a plot of the eigenvalues corresponding to the most significant cylindrical critical assemblies. The point corresponding to a concentric two-region experiment in which different lattice pitches were used in each region is shown plotted vs both abscissa points. Against the banded background indicating the eigenvalue scatter for a large number of assorted UO_2 and U-metal

assemblies, the points corresponding to the mixed-oxide criticals show an overall increase of eigenvalue with increasing lattice pitch or thermalization of the spectrum and, at a given pitch, an increase of eigenvalue with ^{240}Pu content. The one repeated point reflects both trends. The Winfrith thermal representation⁹ for ^{239}Pu was used in the analyses from which the figure was derived but the same trends were observed when other ^{239}Pu representations were employed.

Further effort established that both eigenvalue trends are related to a defective analytic representation of the capture in the 1-eV ^{240}Pu resonance. Normalization of this capture rate to that determined by MONTE CARLO calculations which have been completed together with a reduction in the value 2.89, used as the number of fission neutrons per ^{239}Pu fission is expected to reduce the scatter and to adjust the magnitude of the plutonium eigenvalues to those associated with uranium criticals.

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Criticality of Plutonium Compounds in the Undermoderated Range, $\text{H}:\text{Pu} \leq 20$, L. E. Hansen, E. D. Clayton (BNW)

A variety of plutonium compounds exists for which experimental criticality data are particularly lacking. To obtain guidelines for nuclear criticality safety use, a survey utilizing transport-theory calculations was made to determine the critical masses of bare and water-reflected spheres as a function of density and $\text{H}:\text{Pu}$ ratio for 12 of these compounds in the undermoderated range ($\text{H}:\text{Pu} \leq 20$). Core density exponents were also derived which permit critical masses to be predicted for compounds with densities ranging down to one-fifth of their theoretical values. The validity of the calculations was examined by comparing results with the limited criticality data on homogeneous PuO_2 systems in the undermoderated range.

Comparisons were also made for Pu-metal systems for three heterogeneous plutonium-fueled assemblies.

For unreflected assemblies, the critical mass for a given H:X ratio varies inversely as the square of the density, but in the case of a reflected assembly, no single density exponent exists. The density exponent may theoretically have values between 1.0 and 2.0, depending on the relative effectiveness of the core and reflector^{1,2}. The exponent will be constant, however, at a given H:X ratio for a range of densities.

For reflected systems, the critical mass (M_c) at some core density (ρ_{core}) may be expressed as

$$M_c = M_{c_0} \left(\frac{\rho_{0 \text{ core}}}{\rho_{core}} \right)^m \left(\frac{\rho_{0 \text{ refl}}}{\rho_{refl}} \right)^n,$$

where $\rho_{0 \text{ core}}$ and $\rho_{0 \text{ refl}}$ are the theoretical densities of the core and reflector, and M_{c_0} is the critical mass at those densities. Water-reflected and spherical critical masses (M_{c_0}) for eight of the compounds considered; as well as for plutonium atoms in water, are given in Fig. 1.

To determine the core density exponent (m), the spherical critical mass was computed at core densities ranging from $0.1 \leq \rho/\rho_0 \leq 1.0$, where ρ_0 is the theoretical concentration of the compound at a specific H:Pu ratio. The core density exponents were found to be constant over the range $0.2 < \rho/\rho_0 \leq 1.0$.

Density exponents were also computed for plutonium atoms in water at various H:Pu ratios as a function of water-reflector thickness.

The spherical critical radii were computed by means of the DTF transport-theory code using an S_4 approximation in the solution of the transport equation³. Multigroup constants (18 energy groups) used in the DTF code were obtained from the GAMTEC-II computer code⁴.

The results of critical experiments, some heretofore unreported, performed at the Pacific Northwest Laboratory on PuO_2 -polystyrene assemblies and at Los Alamos on ^{239}Pu -metal systems compare favorably with computed critical masses. These data partially confirm the computational method.

The use of these data, in conjunction with density relationships, provides a simple method of obtaining realistic estimates of critical dimensions and masses for some of the most frequently encountered compounds of plutonium.

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Fig. 1. Water-reflected spherical mass of homogeneous ^{239}Pu compound-water mixtures.

assemblies basically reflect the differences in critical mass.

Transferring of the gas into the tank by the heat pump method was at a rate of about 0.4 kg of UF_6 per hour with a 50°C temperature differential. The system multiplication was nil with the UF_6 in solid form, all probably at the bottom of the tank. The increase in multiplication as the complete gaseous phase was obtained was smooth; but upon cooling of the tank, the initial condensation on the walls created a small positive transient reactivity increase that rapidly turned negative as the condensate thickened.

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Aerospace Nuclear Safety

Cavity Reactor Gas-Core Critical Experiment, L. S. Masson, G. D. Pincock, J. F. Kunze, R. E. Wood (GE-Idaho), R. E. Hyland (NASA-Lewis)

Previous critical experiments^{1,2} involving the gaseous cavity reactor concept have used metal foils or graphite-uranium rods to simulate the gas in the cavity. On May 17, 1967, an actual gaseous (UF_6) core configuration was made critical, and subsequent measurements were made to compare the results from the real gas configuration with those from a mockup foil configuration of the identical dimensions. The gaseous fuel was contained inside an aluminum tank with a diameter of 122 cm and a length of 109 cm. The surrounding moderator-reflector tank was a cylinder with an inside diameter of 183 cm and a length of 122 cm and an outside diameter of 366 cm and length of 305 cm. The tank was filled with D_2O to a nominal thickness of 89 cm and contained a 10.2-cm-thick beryllium slab in the D_2O spaced 6.7 cm from the cavity wall. The reactor assembly was as shown in Fig. 1. The prior mockup critical experiment was performed using 0.0025-cm-thick sheet uranium metal dispersed throughout the fueled volume and supported on aluminum screens. In both experiments, the fuel was 93% enriched in ^{235}U .

A comparison summary of the experimental results is shown in Table I. The corrected critical loadings differed by about 12%, which ostensibly can be attributed mainly to the partially nonhomogeneous distribution of fuel in the mockup experiment. A rough statistical analysis of the nuclear path-length variations in the mockup with respect to the optimum conditions in the gas configuration show that a 5 to 10% difference in interaction rate can be expected.

The mockup reactor also contained 74 kg of Al which was used to support the fuel and was not in the UF_6 reactor. The critical mass for the mockup was corrected for this aluminum for the fluorine in UF_6 and for other minor material differences, as shown in the table by using measured material worths. The largest correction was for Al support screens, these being equivalent to 2 kg of ^{235}U . All of the other data in Table I obtained from the two

TABLE I
Comparison of Experimental Results

	Mockup Assembly	UF_6 Assembly
^{235}U Critical Mass (kg)	25.85	21.30
Correction for Al Excess in Mockup (kg of ^{235}U)	-2.01	
Correction for F and Teflon Insulation (kg of ^{235}U)	-0.03	
Correction Critical Mass of Mockup (kg of ^{235}U)	23.81	
Fuel Worth $\Delta k/k/\Delta m/m$ ^{235}U	0.185	0.174
Ratio of Fission Rates Radially from Outer Edge of the Fuel to the Core Center	2.78	2.47
Infinitely Dilute ^{235}U Cadmium Ratio at Core Center	6.6	6.9
Infinitely Dilute ^{235}U Cadmium Ratio at End of Fueled Region on Axial Centerline	14.2	13.1
Gold-Foil Activation Ratio Radially from the Outer Edge of the Fuel to the Core Center	1.73	1.57
Infinitely Dilute Gold-Foil Cadmium Ratio at Core Center	1.17	1.22
Infinitely Dilute Gold-Foil Cadmium Ratio at Ends of Fueled Region on Axial Centerline	1.53	1.47

Fig. 1. Gas-core cavity reactor experiment.

Critical Assembly Analysis and Criticality

Analysis of Some ^{235}U -Graphite Critical Experiments, *Bal Raj Sehgal (BNL)*

A series of graphite-moderated critical experiments with 93%-enriched uranium-aluminum fuel plates has been completed at Brookhaven.¹ In four of these experiments the fuel was 0.015-in. thick and the $\text{C}/^{235}\text{U}$ atom ratios varied from 5700 to 23 000. These cores were bare, of simple geometry, nearly homogeneous, and primarily thermal. Critical bucklings and spectral indexes were measured among other parameters. A theoretical analysis of these measurements is attempted in this note.

These lattices are characterized mainly by their high leakage rate (about half of the neutrons). Two methods are used for estimating the leakage and the lattice reactivities. First is the conventional method of the B_1 approximation, in which the first two spatial moments are correct and good estimates are made of the higher moments. This method is incorporated in the HAMMER code² which also provides the lattice reactivities. In the second method, leakage is calculated by the moments method and is combined with the zero buckling calculation (done with the HAMMER code) to give the lattice reactivity.

The MOCA code³ is used to calculate the first three even moments of the slowing down distribution ($0.625 \text{ eV} \leq E \leq 10 \text{ MeV}$). Unfortunately, the moments expansion does not converge rapidly enough and the higher moments have to be estimated. These were estimated by fitting the MOCA moments to a combination of age, diffusion, and single collision kernel moments. The thermal leakage is calculated in the B_1 approximation since a Monte Carlo code is unavailable.

In both the thermal and epithermal ranges, the group cross sections were adjusted to reproduce the measured L^2 and τ for graphite.^{4,5} In the thermal range, the GASKET-FLANGE⁶ kernel is used for graphite and the ^{235}U cross sections conform to the evaluation of BNL-325.⁷ The resonance parameters of ^{235}U are taken from BNL-325, and the smooth cross sections in the epithermal range are constructed to normalize to the total fission and capture integrals recommended by Feiner.⁸

The results of the calculations for k_{eff} are shown in Table I, along with the nonleakage probabilities $P_f(B^2)$ and $P_t(B^2)$ calculated for the fast and thermal ranges in the moments method. The estimated uncertainties in these quantities are also shown. Table II compares the results of calculations and experiments for cadmium ratios and activation of various foils.

It is seen that the calculated results for k_{eff} are in good agreement with the measurements and that the B_1 approximation describes the leakage effects quite accurately even in such high leakage lattices. The calculated results for cadmium and activation ratios agree fairly well with the measurements.

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TABLE I
 $\text{C}/^{235}\text{U}$ Lattices Results for k_{eff}

$\text{C}/^{235}\text{U}$ atom ratio	Measured $B_c^2 \text{ (m}^{-2}\text{)}$	HAMMER	MOMENTS METHOD		
		k_{eff}	MOCA $P_f(B^2)$	B_1 Approx $P_t(B^2)$	k_{eff}
5724	16.88 ± 0.08	0.9998 ± 0.0025	0.59474 ± 0.00589	0.81434	0.9996 ± 0.0103
11450	13.46 ± 0.05	0.9987 ± 0.0020	0.67353 ± 0.00452	0.76433	0.9968 ± 0.0069
17180	11.02 ± 0.03	1.0014 ± 0.0014	0.72618 ± 0.00375	0.74263	0.9992 ± 0.0053
22900	9.33 ± 0.02	0.9989 ± 0.0010	0.76445 ± 0.00302	0.73190	0.9969 ± 0.0041

TABLE II
 $\text{C}/^{235}\text{U}$ Lattices
Results for Foil Cadmium Ratios and Activation Ratios

Foil	Cadmium cutoff energy (eV)	$\text{C}/^{235}\text{U} = 5724$				$\text{C}/^{235}\text{U} = 22\ 900$			
		Cadmium calculated	Ratio ^a measured	Activation calculated	Ratio ^b measured	Cadmium calculated	Ratio measured	Activation calculated	Ratio measured
^{55}Mn	0.729	5.73	5.69 ± 0.07	-	-	17.80	17.32 ± 0.10	-	-
^{239}Pu (fis)	0.620	19.71	19.70 ± 0.40	1.240	1.28 ± 0.009	52.91	52.00 ± 1.00	1.086	1.123 ± 0.010
^{235}U (fis)	0.648	10.67	10.89 ± 0.07	0.865	0.862 ± 0.005	36.26	35.50 ± 0.80	0.953	0.955 ± 0.010
^{197}Au	1.417	1.333	1.342 ± 0.009	3.40	3.38 ± 0.05	2.11	2.09 ± 0.02	1.811	1.835 ± 0.015
^{115}In	1.334	1.451	1.449 ± 0.007	3.53	3.42 ± 0.04	2.24	2.24 ± 0.02	1.877	1.868 ± 0.010

^aCadmium ratio = activity of bare foil/activity of cadmium-covered foil.

^bActivation ratio = $\frac{\text{Activity of bare foil in lattice}}{\text{Activity of Mn foil in lattice}} / \frac{\text{Activity of bare foil in thermal column.}}{\text{Activity of Mn foil in thermal column.}}$

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Studies of Critical Assemblies of Homogeneous Mixtures of Plutonium Oxide and Polystyrene, Alan M. Baxter (GA), E. D. Clayton, L. E. Hansen (BNW)

A series of criticality experiments has been conducted at the Pacific Northwest (PNL) Plutonium Critical Mass Laboratory with plutonium dioxide/polystyrene compacts with H/Pu ratio of 5 as fuel. Data on the fuel compacts are given in Table I. These experiments were part of the General Atomic Gas-Cooled Fast Reactor (GCFR) program to check calculational methods and basic neutronic data used in the GCFR design; and were also part of the PNL program to determine critical sizes of plutonium mixtures over the entire range of densities possible in separation plants.

The experiments consisted of the measurement of the critical size and mass of these fuel compacts stacked as rectangular prisms of near-cubic geometry. Flux distributions in the fuel blocks were measured with manganese-copper wires, and ^{239}Pu and ^{235}U fission ratios were obtained. The reactivity effects of the resonance absorber elements dysprosium, hafnium, and gadolinium, which are being considered as poisons to control the consequences of steam or water entry into the GCFR core, were also measured by stacking thin layers of these materials in the critical assemblies. The experiments also served as clean integral checks on basic cross-section data, particularly in the region of resolved resonances (5 to 300 eV) where approximately 30% of the fissions in these assemblies occur. Recent intercomparison studies on dilute-Pu-fueled fast assemblies¹ have amply demonstrated the need for such integral checks for ^{239}Pu . Results of interest are presented in Table II. Calculations were performed using multigroup transport theory (20 energy groups) with multigroup constants averaged from basic cross-section data using the GGC and GAF/GAR codes.²

Excellent agreement, to within 0.3% in k_{eff} , between calculation and experiment was obtained for the unpoisoned plutonium assemblies using a new ^{239}Pu cross-section evaluation³ which included resonance parameters based on the data of Schmidt⁴ and others and the high-energy σ_f measurements of White et al.⁵ By comparison,

TABLE I
Fuel Compact Data

Average fuel compact dimensions	= $2.038 \times 2.03 \times 1.528$ in.
Average fuel compact composition	
PuO ₂	= 80.63 wt%
Polystyrene	= 19.37 wt%
PuO ₂ -polystyrene/block	= 325.156 g
Plutonium/block	= 299.95 g
H/Pu ratio	= 5.031
% ^{240}Pu in the plutonium	= 12%

TABLE II

Unreflected Critical Configurations of Plutonium Assemblies

Assembly	Length (in.)	Width (in.)	Height (in.)	Mass of Plutonium (kg)	Poison		Calculated k_{eff}
					Type	Mass (kg)	
1	14.41	14.22	11.14	80.44	-	-	0.998
2	18.28	12.13	11.05	81.60	-	-	1.007
3	18.26	12.33	11.14	81.84	-	-	1.002
4	18.28	12.36	14.59	92.302	Dy ₂ O ₃ ^a	6.005	0.950
5	18.26	12.36	14.05	89.875	Gd ₂ O ₃ ^a	3.668	0.990
6	18.30	12.38	14.15	90.218	HfO ₂ ^a	10.678	0.994

^aThe poison was contained in three $\frac{1}{8}$ -in.-thick layers spaced at $1\frac{1}{2}$ -in. intervals and separated by 3 in. of 5/1 H/Pu ratio fuel from the assembly base.

calculations using earlier ^{239}Pu evaluations⁶ overestimated k_{eff} by almost 4%.

The hafnium- and gadolinium-poisoned assemblies were calculated to within 1% in k_{eff} using cross-section data for these elements which included new resonance parameter sets compiled for their important isotopes.⁷

However, calculations based on the dysprosium cross-section evaluation⁷ overestimated the worth of this material by about 5% in k_{eff} . In obtaining these cross sections, the resonance absorption integral had been normalized to the reported measured value⁸ because use of the available resonance parameters gave a value some 300 b lower. Use of the unnormalized data in the calculations increased k_{eff} by some 6%, indicating that the measured resonance absorption integral value may be too high.

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Criticality of ^{233}U Aqueous Nitrate Solution in Reflected and Unreflected Arrays, J. T. Thomas (ORNL)

Uranium-233 nitrate solution at a concentration of 333 g U/liter (325 g ^{233}U /liter) was utilized in a brief study to determine the critical parameters of arrays of subcritical components each containing 4.30 liters of solution. The data are directly applicable to nuclear safety specification and serve as bases to evaluate models and methods of computing such systems. Similar experiments with ^{235}U have been reported.¹ The containers were cylindrical, fabricated of 0.25-mm-thick stainless steel, and had outside diameters and heights of 18.28 and 17.67 cm, respectively. The capacity of a container was 4.63 liters. The solution had a specific gravity of 1.468, an H: ^{233}U atomic ratio of 73, and the uranium isotopic content reported in Table I.

All assembled arrays were three-dimensional, having an equal number of units along the three directions of the array. Each cylinder was supported in an array, with its axis vertical, by a 1.6-mm-thick aluminum plate positioned in space on four stainless-steel rods of 5-mm diam. The reflector material surrounding an array of cylinders was polyethylene ($\rho = 0.93 \text{ g/cm}^3$), used in a 15.2-cm thickness only and located from the peripheral cylinders of the array of distance equal to one-half the surface separation between cylinders. The center-to-center spacing of the units in 8- and 27-unit arrays for both the reflected and the unreflected conditions are presented in Table II.

TABLE I
Isotopic Analysis of Uranium

Isotope	wt%	Isotope	wt%
^{233}U	97.54	^{235}U	0.001
^{234}U	1.047	^{238}U	1.386
^{235}U	0.026	^{232}U	(6.47 ppm)

TABLE II
Criticality Parameters for Unreflected and Reflected Arrays of 1.432-kg Units of Uranyl Nitrate Solution Containing 97.6% ^{233}U .

Number of Units in Array	Polyethylene Reflector Thickness (cm)	Center-to-Center Separation of Units Horizontal (cm)	Center-to-Center Separation of Units Vertical (cm)	Average Uranium Density (g/cm ³)	KENO Code k_{eff}
8	0	20.44	19.13	0.179	0.991 ± 0.005
27	0	25.72	24.57	0.088	1.007 ± 0.005
8	15.2	31.95	30.36	0.046	--
27	15.2	41.03	38.57	0.022	--

Monte Carlo calculations of the unreflected arrays using the KENO² code with the Hansen-Roach³ 16-group cross-section set for ^{233}U have been accomplished and agree well with experiment. These values of k_{eff} are also given in Table II.

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The Nuclear Safety of Aqueous Solutions of ^{233}U and ^{235}U , J. Wallace Webster (ORNL)

Increased basic knowledge and operating experience have made desirable a revision of USASI Standard N6.1-1964, "Safety Standard for Operations with Fissionable Materials Outside Reactors," and such a revision is now being prepared by Subcommittee 8 of the Standards Committee of the American Nuclear Society. The computations reported in this paper were made to aid the Committee in establishing safe limits for process variables for uniform aqueous solutions of ^{233}U and ^{235}U , assuming an infinite water reflector. The limits are concerned with the following parameters: mass of fissile material, infinite cylinder diameter, infinite slab thickness, volume, chemical concentration, and density of fissile material per unit area of a slab-shaped volume of solution.

The values of the maximum k_{eff} as a function of these parameters were computed with other variables having their optimum values, and the results are presented in detail by Webster¹ in the form of 12 principal figures plus the necessary working graphs.

The computations, which were all carried out using the S_N method and Hansen-Roach cross sections,² were compared with a broad range of aqueous-solution experiments performed at the Oak Ridge Critical Experiments Facility and the uncertainty in the computed k_{eff} 's was found to be about 1%. Thus, limits corresponding to $k_{\text{eff}} = 0.99$ are presented in the table.

The parameter that proved to be the most interesting from a computational standpoint was the minimum thickness of a water-reflected infinite slab of $^{233}\text{UO}_2\text{F}_2$. Fox et al.³ conducted a difficult series of critical experiments in 1958 to seek this value. From these experiments, which were with $\text{U}(93.2\%)\text{O}_2\text{F}_2$ near optimum concentration in a methacrylate plastic container and water reflector, they obtained an estimate of 1.76 in. as the critical

TABLE I

Safe Limits for Aqueous Solutions of Uranium
(Computed $k_{\text{eff}} = 0.99$)

Parameter	^{235}U	^{233}U
I Mass (g)	780	550
II Cylinder diameter (cm)	14.1	11.7
III Slab thickness (cm)	4.7	3.0
IV Volume (liters)	5.9	3.5
V Concentration (g/liter)	11.5	11.0
VI Density of U per unit area for slabs (g/cm ²)	0.41	0.36

thickness by extrapolating the experimental results to the case of infinite transverse dimensions and by a substitution measurement of the worth of the plastic vs water. Using the S_N method, however, to extrapolate the results to infinite transverse dimensions, and to evaluate the effect of replacing the plastic with water, a critical thickness nearer to 2.0 in. is obtained. This makes possible a

more than 10% increase in the safe limit recommended for the Safety Standard for the thickness of slab-shaped storage, processing, and shipment containers.

1. J. W. WEBSTER, "The Nuclear Safety of Aqueous Solutions of ^{235}U and ^{235}U ," ORNL-CDC-2, Oak Ridge National Laboratory Criticality Data Center Report (1967).
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Minimum Critical ^{235}U Enrichment for Uranyl Nitrate Hydrogenous Systems, S. R. Bierman, G. M. Hess (BNW)

The critical parameters of slightly enriched uranium systems are strongly dependent on both the enrichment and degree of moderation. Previous experiments¹ performed in the Physical Constants Test Reactor (PCTR) of Battelle-Northwest have shown the minimum critical ^{235}U enrichment for homogeneous mixtures of UO_3 in water to be 1.034 ± 0.010 wt%. Additional experiments² with 3.04 wt% enriched UO_3 and uranyl nitrate $\text{UO}_2(\text{NO}_3)_2$ hydrogenous mixtures established that the presence of the nitrate radical had a pronounced effect on criticality. The 3.04 wt% enriched UO_3 had a maximum k_{∞} of 1.35, as compared to 1.15 for the 3.04 wt% enriched uranyl nitrate. Subsequent calculations³ have indicated the minimum critical enrichment for uranyl nitrate to be about 2 wt%.

The minimum critical enrichment is the enrichment required to obtain an infinite neutron multiplication factor of unity under conditions of optimum moderation. Knowledge of the enrichment value and the H/U ratio of the homogeneous mixture that gives the maximum k_{∞} of unity provides a fundamental point for comparison with reactor theory. A firm knowledge of this enrichment value is also of considerable interest for criticality control purposes in processing and handling slightly enriched uranyl nitrate solutions. Below this enrichment a self-sustaining neutron chain reaction would be impossible and the problems associated with nuclear safety would be greatly reduced in situations where this enrichment is not exceeded. Consequently, a series of measurements was made in the PCTR to determine the enrichment at which k_{∞} is just unity at optimum moderation. The results of these experiments are presented graphically in Figs. 1 and 2. The normal PCTR null reactivity technique⁴ was employed to obtain these results, except that a negative neutron poison approach had to be taken in those systems having a k_{∞} less than unity. The minimum critical enrichment is 2.104 ± 0.010 wt% and occurs at a H/U ratio of about eight. As can be seen in Fig. 2, this enrichment is within 1% of that calculated by Ridgway.³

1. V. I. NEELEY and H. E. HANDLER, "Measurement of Multiplication Constant for Slightly Enriched Homogeneous UO_3 -Water Mixtures and Minimum Enrichment for Criticality," HW-70310, Hanford Atomic Products Operation (August 21, 1961).
2. V. I. NEELEY, J. A. BERBERET, and R. H. MASTERSON, " k_{∞} of Three Weight Percent U^{235} Enriched UO_3 and $\text{UO}_2(\text{NO}_3)_2$ Hydrogenous Systems," HW-66882, Hanford Atomic Products Operation (September, 1961).

Fig. 1. PCTR reactivity response as a function of ^{235}U enrichment and neutron moderation.

Fig. 2. Minimum critical ^{235}U enrichment for homogeneous hydrogenous uranyl nitrate.

3. K. R. RIDGWAY, "Calculated Nuclear-Safety Parameters of Low Enriched Uranium," *Trans. Am. Nucl. Soc.*, 9, 134 (June, 1966).
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Physics of Uranium-233 Reactors

Crystallization Characteristics of Acidic Uranyl Nitrate Solutions,* R. C. Kispert, J. H. Cavendish, N. R. Leist (Nat'l Lead)

An important consideration in the safe transportation or processing of enriched-uranium solutions in ordinary equipment is solution crystallization, which may cause the uranium concentration to exceed the safe limits established by the isotopic level. Process operations require solution stability, which can be maintained by temperature control. In the transportation of solutions, the free-acid concentration may be adjusted to ensure that should crystallization occur, the maximum uranium concentration of the solution will never exceed the safe limit.

Crystallization temperatures of pure solutions of uranyl nitrate in nitric acid may be found in the literature; however, similar data for impure solutions are unavailable. Therefore, it was the purpose of this work to develop criteria for predicting the crystallization temperature of impure uranyl nitrate solutions.

The crystallization behavior of pure solutions is presented in Fig. 1 as a function of the total- and free-nitrate concentrations. Temperatures to the left of the eutectic concentrations decrease with increasing nitrate concentrations and fall on a common line for all free-acid concentrations. In this concentration region, ice is the crystallized species and demonstrates the phenomenon of freezing-point depression. If the eutectic concentration is exceeded, then the free-acid concentration becomes a parameter, and the crystallization temperature of the solution increases rapidly with increased nitrate concentrations. Solution stability in the region to the right of the eutectic is controlled by uranyl nitrate solubility.

To test this correlation for impure solutions, 43 acidic uranyl nitrate solutions ranging in concentration from 3.7 to 12.4 M NO_3^- were prepared, each containing a single nitrate impurity such as iron, aluminum, calcium, copper, chromium, or ammonium. The crystallization temperature was determined for each solution and related to the total and free nitrate concentrations defined below. Concentration units are expressed in molarity.

$$\text{Total } \text{NO}_3^- = [\text{NO}_3^-]_{\text{UO}_2(\text{NO}_3)_2} + [\text{NO}_3^-]_{\text{HNO}_3} + [\text{NO}_3^-]_{\text{impurity}}$$

$$\text{Free } \text{NO}_3^- = [\text{NO}_3^-]_{\text{HNO}_3} + [\text{NO}_3^-]_{\text{impurity}}$$

It is necessary to subtract the NO_3^- complexed with uranyl ion from the total nitrate concentration to correlate the free nitrate in a manner similar to pure solutions. Equations (1) and (2) reduce to the parameters for pure solutions and, thus, become compatible with Fig. 1. No effect was observed from the nature of the cation species. Consequently, the crystallization temperature of impure solutions may be predicted from knowledge of the nitrate concentrations.

An explanation of the correlation can be offered by considering the solubility of uranyl nitrate to obey an equilibrium for which a constant may be written and related to the crystallization temperature.

$$K = [\text{UO}_2^{++}] \cdot [\text{NO}_3^-]^2$$

Values of the equilibrium constant were found to be independent of the free-acid or total-nitrate concentrations in the lower concentration range. Some departure was observed at higher nitrate concentrations and probably results from the increasing non-ideality of the solution.

Fig. 1. Crystallization characteristics of uranyl nitrate-nitric acid solutions.

Using activities instead of concentration units may yield a more rigorous relation.

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²³³U-Th Seed-Blanket Critical Experiments, S. Milani (W-BAPL), *Invited*

Critical experiments utilizing ²³³U and Th fuels have been conducted at the Bettis Atomic Power Laboratory since 1964. The critical assemblies were of the seed-and-blanket type and the experiments were performed in support of the Light-Water Breeder program. Two sets of experiments have been conducted to date. The first set consisted of 15-in. high assemblies which contained 26 wt% ²³³UO₂-ZrO₂ seed fuel in which the uranium was either ²³³U or ²³⁵U, and blanket fuel of ThO₂ or 1 wt% ²³³UO₂-ThO₂. All elements were rod type with Zr clad. The second set consisted of 12 wt% ²³³UO₂-ThO₂ seed fuel 28-in. high, and ThO₂ blanket fuel. Future experiments are planned in this second set with lower concentrations of ²³³U in the seed fuel and with blanket fuel that contains ²³³U.

Some results from the first set of experiments have been reported previously¹⁻⁴ along with a complete description of the fuel and assemblies. In addition to the reported experiments which were conducted at 68°F, two of these assemblies were studied at 480°F in the Bettis High Temperature Test Facility. Each assembly contained a blanket of 1 wt% ²³³UO₂-ThO₂ and the seed for one contained ²³³UO₂-ZrO₂ and for the second, contained ²³³UO₂-ZrO₂. Essentially the same agreement between experiment and analysis was found for the hot experiments as was found for the cold. For example, the critical eigenvalues as calculated by four-group, two-dimensional diffusion-theory calculations using the experimentally determined axial bucklings were within 0.4% of similarly calculated cold eigenvalues for the two assemblies.¹ Also, all the experiments of this set reported previously were of the "clean" type; i.e., very little structural material other than the fuel rods existed in the active region of the core. An additional "design"-type experiment was performed in which Zr structural material was included. This assembly had a ²³³U bearing seed, an inner blanket bearing ²³³U, and an outer blanket of ThO₂ rods. A converged source RECAP⁵ Monte Carlo calculation in which each fuel rod was described explicitly in three dimensions was performed. The critical eigenvalue from this problem was in close agreement with experiment.

The first assembly in the second set was a slab-type seed-and-blanket assembly. The seed had critical dimensions of 5.6 × 22 × 28 in. and consisted of 0.25-in. Zr-clad rods that contained 12 wt% ²³³UO₂-ThO₂ fuel pellets. The seed pitch was 0.310-in. triangular and the H/U = 30. The blanket reflected the seed on two sides and consisted of 0.625-in. Zr-clad ThO₂ rods. It had a height of 42 in. and extended above and below the seed. Each section of the blanket was 24-in. long and 10.9-in. thick. The assembly was clean critical in that little structural material existed in the fuel region and the control rods that were Hf blade-type rods were almost fully withdrawn at critical. The measured rods-out eigenvalue was 1.003. A preliminary calculated eigenvalue using a two-dimensional diffusion-theory calculation similar to those of the first set gave an eigenvalue of 0.986. This is approxi-

mately 2% lower than the eigenvalues obtained from the first set of experiments.

Experiments in the future will utilize lower weight percents of UO₂-ThO₂ in the seed and ²³³U-bearing blankets. Emphasis will be placed on assemblies that mockup the LWB design core.

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3. S. H. WEISS et al., "Activation Shapes in ²³³U and ²³⁵U-Th Seed-Blanket Critical Assemblies," *Trans. Am. Nucl. Soc.*, 10, 537 (1967).
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Reactor Physics Parameters of ²³³UO₂-ThO₂ Lattices in D₂O, Henry H. Windsor, William J. Tunney (BNL), *Invited*

The measurement of reactor physics parameters of uniform lattices of 3% ²³³UO₂-97% ThO₂ fuel rods moderated with D₂O are reported in this note. The fuel is vibratorily compacted to a density of 8.96 g/cm³ in Zircaloy-2 tubes of 1.27-cm-o.d. and 0.8-mm wall thickness as described in detail in a previous report.¹

The D₂O-moderated experiments were carried out in a 75.405-cm-i.d. tank surrounded with 0.635-cm boral sheet. A 15.24-cm paraffin reflector was installed outside the boral sheet for the 11.481-, 6.510-, 3.759-, and 2.170-cm-pitch lattices. This reflector was not present for the rest of the lattices; consequently, it was necessary to account for room-return effect in the buckling analyses of these lattices. Preliminary results of the buckling analyses as well as the microscopic measurements are presented in Table I.

Measurements were made of ξ_{Dy} (dysprosium disadvantage factor), ρ_{02} (ratio of epi- to subcadmium capture ratio in ²³²Th), and δ_{02}^{23} (ratio of fission in ²³²Th to fission in ²³³U). The techniques used in measuring ξ and ρ were reported elsewhere² and will not be reported here. Of more interest is a new technique of measuring the fast fission ratio.

The technique makes use of the sensitivity of some plastics to radiation damage from energetic charged particles as described by Fleischer et al.³ The plastic foils are used to record the fissions that occur in adjacent thin films of ²³³U and ²³²Th that are simultaneously exposed between fuel pellets in flux symmetric positions in the lattice. The thin films were electrodeposited on 0.025-mm × 1.08-cm steel or platinum with ²³²Th film thickness of 200 μ g/cm², and ²³³U film thicknesses of 0.5 μ g/cm², and the plastic used was 0.152-mm × 1.08-cm "lexan."

TABLE I
Material Bucklings, ρ_{α} , ζ_{Dy} , and δ_{α}^{23} of the 3% ^{235}U 97% ThO_2 D_2O Lattices

Lattice No.	Mole % D_2O	Pitch cm	B_1^2 m^{-2}	$-B_m^2$ m^{-2}	B_m^2 m^{-2}	Mole % D_2O	ζ_{Dy}	ρ_{α} Cd Ratio	ρ_{α} Thermal Activation	δ_{α}^{23}
1	± 0.05									
2	99.25	11.481	35.96 ± 0.06	24.67 ± 0.05	11.29 ± 0.20	99.22	1.423 ± 0.017	0.089 ± 0.005		
3	98.95	9.462	34.69 ± 0.35	20.02 ± 0.13	14.67 ± 0.37	99.22	1.387 ± 0.015	0.104 ± 0.005		
4	99.34	7.821	35.43 ± 0.26	16.30 ± 0.05	19.13 ± 0.27	99.23	1.374 ± 0.013	0.166 ± 0.008		
5	99.25	6.507	34.68 ± 0.1	12.36 ± 0.09	22.32 ± 0.14	99.25	1.345 ± 0.012	0.234 ± 0.008		
6	99.33	5.738	34.81 ± 0.05	9.81 ± 0.12	25.00 ± 0.16	99.30	1.333 ± 0.015	0.279 ± 0.011		
7	99.30	4.338	34.50 ± 0.25	5.86 ± 0.14	28.64 ± 0.29	99.33	1.294 ± 0.016	0.559 ± 0.018	0.634 ± 0.060	0.0047 ± 0.0007
8	99.26	3.758	34.24 ± 0.2	4.39 ± 0.07	29.85 ± 0.22	99.20	1.259 ± 0.013	0.780 ± 0.032	0.840 ± 0.058	0.0056 ± 0.0007
9	99.30	2.170	33.16 ± 0.16	12.62 ± 0.1	20.54 ± 0.20	99.24	1.173 ± 0.015	5.19 ± 0.54	4.66 ± 0.19	0.0117 ± 0.0008

Each fission that occurred in the thin film caused a damaged area in the plastic that resulted in a short track when etched in sodium hydroxide. The plastic was then photographed resulting in a $170 \times$ blowup from which the tracks were counted visually. This count yielded the relative fission rate in the two thin films that had been calibrated previously by absolute alpha counting.

The advantage of this technique is that it does not require a double fission-chamber calibration as in the gross fission-product counting technique, or elaborate chemistry and reliance upon fission-product yields as required in the method that separates and counts a single fission product.

This technique is particularly well suited for this fuel because of the small fission cross section of ^{232}Th . The quantities of electrodeposited ^{232}Th and ^{235}U can be controlled to give the same fission rate in the two thin films, thus making the accuracy of the measurement somewhat independent of the value of δ . The results of all the measurements are shown in the table.

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2. BAL RAJ SEHGAL et al., "Measurement and Analysis of Some $\text{U}^{235}\text{O}_2\text{-ThO}_2$ Water Lattice Experiments," *Trans. Am. Nucl. Soc.*, 10, 193 (1967).
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Analyses of Brookhaven ^{235}U - ^{232}Th Oxide, Light- and Heavy-Water-Moderated Lattice Experiments, Bal Raj Sehgal (BNL), Invited

A series of exponential lattice experiments with 3% ^{235}U -enriched ^{232}Th -oxide fuel in light and heavy water have been completed recently at Brookhaven. The details of the experiments and the parameter data have been reported.^{1,2} In addition, results of the preliminary calculations for the light-water lattices were reported.¹

These experiments constitute a comprehensive set of integral data to check the calculational methods and the cross-section libraries for ^{235}U and ^{232}Th . A good range of reactor spectra has been provided by varying the pitch such that the fraction of fissions of ^{235}U above 0.625 eV varies from 0.04 to 0.6. The core leakage has been varied from 4 to 36% of the neutrons by boron poisoning the light-water moderator.

The main tool of analyses of these lattices is the HAMMER multigroup code.³ The inadequacies of the resonance treatment in the HAMMER Code are corrected by inserting the Monte Carlo calculations of resonance capture with the REPETITIOUS Code⁴ in the energy band of 17.6 to 1234 eV.

The cross-section libraries reported earlier¹ have been modified by using the ENDF/B cross sections for ^{235}U in the thermal energy range.⁵ With this change, the results of K_{eff} for the light-water lattices are somewhat improved. However, there are still trends in the calculated values of K_{eff} . A plot of K_{eff} vs τB^2 shows that the leakage may be underestimated by $\approx 2\%$ for these lattices. If the calculated τ is increased by $\approx 2\%$, the average value of K_{eff} for the light-water lattices is $\approx 1.0 \pm 0.0046$.

The calculated values of K_{eff} for the D_2O -moderated lattices are, on the average, low by $\approx 1.3\%$. These lattices are bare cores and the leakage rate is very high ($\approx 35\%$ of the neutrons). A systematic overestimate of the measured buckling by $\approx 0.5 \text{ m}^{-2}$ could account for the discrepancy in reactivity between calculations and measurements.

Calculations of age for the Roberts-Pettus experiment⁶ of $^{232}\text{ThO}_2$ rods in mixtures of light and heavy water did not agree well with the measurements. For example, the calculated age for the pure light-water case was $\approx 10\%$ less than the measured age.

The calculated values of ζ_{Dy} , the Dy activation thermal disadvantage factor are $\approx 2\%$ higher than the measured values for both the light- and heavy-water lattices. The calculated values of ρ_{α} , the ratio of epi- to subcadmium captures in ^{232}Th , are in fairly good agreement with the measurements. The calculations of δ_{α}^{23} , the ratio of the fissions in ^{232}Th to those in ^{235}U , do not agree well with the three values that were measured in the D_2O moderator.

Thus, a state of satisfactory agreement between calculations and measurements has not yet been achieved for ^{235}U - $^{232}\text{ThO}_2$ lattices. It is hoped that the ENDF/B set of cross sections may give better results.

1. B. R. SEHGAL, H. H. WINDSOR, and W. J. TUNNEY, "Measurements and Analyses of Some ^{235}U $\text{O}_2\text{-ThO}_2$ Water Lattice Experiments," *Trans. Am. Nucl. Soc.*, 10, 193 (1967).
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Lattice Measurements and Calculations

Buckling Measurements in a Heavy-Water Superlattice, C. J. Tanner, D. G. Andrews (U of Toronto)

Experiments were performed in the critical heavy-water facility ZED-2¹ located at AECL's Chalk River Laboratories to measure the buckling of a superlattice. The lattice shown in Fig. 1 contained 45 19-element natural uranium metal clusters² and 44 ZEEP rods³ in a square lattice. The 19-element U-metal clusters consisted of 19 uranium metal rods of 1.31-cm diam,

Fig. 1. Superlattice of ZEEP rods and 19-element uranium metal clusters.

sheathed in aluminum of thickness 0.102 cm, arranged as follows: 1 in the center of the bundle, 6 on a 3.55-cm-diam circle, and 12 on a 6.85-cm-diam circle. Each bundle was housed in an aluminum tube 8.9-cm o.d. and 0.16-cm thick. D₂O moderator acted as coolant. ZEEP rods consisted of 3.25-cm-diam natural uranium metal rods sheathed in aluminum of thickness 0.102 cm. There was no coolant space. Measurements were made at three lattice pitches: 18, 22, and 28 cm.

Buckling was measured by flux mapping with manganese foils.² Bucklings were also measured for a lattice containing only 53 19-element U-metal clusters and a lattice containing only 89 ZEEP rods. The measurements were made for the same pitches as above (18, 22, and 28 cm).

Table I shows the results of the measurements. The buckling for 19-element U-metal clusters at 18-cm square lattice pitch was obtained by interpolation from previous measurements.² Buckling values have been corrected to a moderator condition of 20°C and 99.58 at.% D₂O.

TABLE I
Results of this Study

Lattice Pitch cm	Source	Bucklings		
		19-Element Rods m ⁻²	ZEEP Rods m ⁻²	Superlattice m ⁻²
18	Measured	3.900 ± 0.07	7.180 ± 0.075	6.406 ± 0.038
	MICRETE	-	7.041 ^a	6.420
	HERESY	-	7.091 ^a	6.475
22	Measured	5.974 ± 0.057	5.361 ± 0.020	6.113 ± 0.026
	MICRETE	5.793 ^a	5.311 ^a	6.109
	HERESY	5.898 ^a	5.338 ^a	6.139
28	Measured	5.322 ± 0.058	3.322 ± 0.015	4.504 ± 0.057
	MICRETE	5.285 ^a	3.278 ^a	4.452
	HERESY	5.314 ^a	3.283 ^a	4.453

^aThe measured buckling was used in calculating the k_{∞} and ρ used as input parameters for these cases.

Also included in Table I are bucklings as calculated by the computer programs MICRETE⁴ and HERESY.⁵ These programs are based on the source-sink method developed by Feinberg and Galanin,⁶ and Stewart⁴ independently. Input parameters were taken from the computer program POOF.⁷

No simple relationship appears to exist between the buckling of the superlattice and the bucklings of its components. Both MICRETE and HERESY, however, give good values for the superlattice buckling.

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Buckling and Parameter Measurements in the Th-D₂O Organic System, D. J. Pellarin, N. P. Baumann, R. M. Satterfield, V. D. Vandervelde (duPont-SRL)

Reactor physics parameters and bucklings were measured for the Th-D₂O organic reactor system in support of the Atomic Energy Commission's development program for the HWOCR.¹ No previous experimental work has been reported for this system. The primary objective of the experiments was to check the HAMMER² calculational methods for conceptual design studies of such systems. The experimental measurements are compared with HAMMER calculations in Table I and agree within about $\pm 25 \mu B$, which is equivalent to k_{∞} differences of about $\pm 0.0075\%$. Two HAMMER calculational models were used for the 85-rod clusters in HWOCR: 1) All material within the cluster was homogenized into a single region with cross sections given by volume-weighted averages; and 2) the cluster was divided into concentric cylindrical rings of fuel and coolant.

Substitution buckling measurements were made in the critical PDP³ and the subcritical SE⁴ using 2 and 3 concentric tube assemblies of 2% ²³⁵U-Th fuel at regular square and rectangular lattice pitches ranging from 9.0 to 12.12 in. with D₂O, air, and Dowtherm A as coolants.⁵ Similar measurements were also made using 85-rod clusters of THUD⁶ 2% ²³⁵UO₂-ThO₂ fuel at two internal rod pitches in the SE.

Detailed parameter measurements made in the SE included determinations of the thermal-neutron distributions in terms of subcadmium foil activation; thermal-neutron temperatures in terms of Lu/1/v activity ratios; epithermal-neutron fractions in terms of ²³²Th capture and ²³⁵U fission cadmium ratios; ²³⁵U production in terms of neutron captures in ²³²Th; and lattice fissions in terms of the ²³²Th/²³⁵U fission ratios. The HAMMER computations generally agreed with the intracell activation profiles. The spectral index measurements agreed poorly because the computations overestimated the spectral index in all coolants.

A counting technique applicable to low flux conditions was developed to measure neutron captures in ²³²Th. Beta absorbers defined a 100- to 400-keV window for counting the dominant 1.23-MeV ground-state β decay of ²³³Th. The rejected higher energy β 's originated largely from fission products. Residual fission-product activity was monitored by a separate β detector with a 1.3-MeV β absorber. By placing copper and thorium foils in the fuel cluster and in a purely thermal reference flux ρ , the ratio of epicadmium-to-subcadmium capture in ²³²Th could be derived without using cadmium for the thorium. The

measured fuel average values of ρ in Table I are consistently high compared to those computed by HAMMER. This highness is believed to be partially due to incorrect cross sections. The cross sections used give higher thermal and lower resonance cross sections than current ENDF/B values, which include the effect of negative energy resonances.

Further details will be given in a forthcoming report.⁷

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TABLE I
Summary of Measurements and Comparison with HAMMER

Reactor Facility	Lattice Pitch (in.)	Coolant	Fuel Assembly Type ^a	D ₂ O Purity	HAMMER Calculations					
					Experimental		Homogeneous Model		Ring Model	
					B ² , m ⁻²	$\bar{\rho}$	B ² , m ⁻²	$\bar{\rho}$	B ² , m ⁻²	$\bar{\rho}$
SE	9 Δ	"Dowtherm" A	THUD I	99.70	6.23	--	6.17	--	5.78	--
SE	9.5	"Dowtherm" A	THUD I	99.70	5.31	--	5.49	--	5.19	--
SE	10.5	"Dowtherm" A	THUD I	99.61	4.38	--	4.75	--	4.52	--
SE	9.5	D ₂ O	THUD I	99.56	5.92	--	6.13	--	5.87	--
SE	9.5	H ₂ O	THUD I	99.52	4.78	--	5.04	--	4.74	--
SE	9.5	Air	THUD I	99.52	6.12	--	5.91	--	5.82	--
SE	9 Δ	"Dowtherm" A	THUD II	99.20	3.77	--	4.39	--	3.96	--
SE	9.5	"Dowtherm" A	THUD II	99.49	3.47	--	4.03	--	3.70	--
SE	10.5	"Dowtherm" A	THUD II	99.46	2.98	--	3.42	--	3.20	--
SE	9.5	D ₂ O	THUD II	99.17	5.02	--	5.25	--	5.08	--
SE	9.5	H ₂ O	THUD II	99.14	2.14	--	2.70	--	2.33	--
SE	9.5	Air	THUD II	99.16	5.07	--	5.14	--	5.02	--
SE	11.5	Air	THUD II	99.44	--	0.166	--	0.148	--	0.141
SE	11.5	D ₂ O	THUD II	99.43	--	0.182	--	--	--	0.161
SE	11.5	"Dowtherm" A	THUD II	99.43	--	0.184	--	--	--	0.164
SE	9.33 Δ	D ₂ O	THUD II	99.57	--	0.274	--	--	--	0.242
SE	9.33 Δ	Air	THUD II	99.57	--	0.266	--	0.239	--	0.225
SE	9.33 Δ	Air	THUD II	99.46	--	0.255	--	0.238	--	0.224
SE	9.33 Δ	"Dowtherm" A	THUD II	99.43	--	0.248	--	--	--	0.230
SE	∞	Air	THUD II	99.49	--	0.132	--	0.098	--	--
SE	9.5	"Dowtherm" A	TMT I	99.37	6.64	0.345 } 0.351 }	--	--	6.62	0.307
SE	9.5	D ₂ O	TMT I	99.40	7.46	0.318	--	--	7.12	0.363
SE	9.5	Air	TMT I	99.39	7.05	0.343	--	--	6.96	0.298
SE	10.5	"Dowtherm" A	TMT I	99.33	6.07	0.295	--	--	5.58	0.272
SE	12.12 Δ	"Dowtherm" A	TMT I	99.32	5.19	--	--	--	5.27	--
SE	9.5	"Dowtherm" A	TMT II	99.36	6.42	0.292	--	--	6.77	0.272
SE	10.5	"Dowtherm" A	TMT II	99.36	6.01	0.254	--	--	6.03	0.240
PDP	12.12 Δ	"Dowtherm" A	TMT I	99.50	5.38	--	--	--	5.46	--
PDP	12.12 Δ	D ₂ O	TMT I	99.50	6.22	--	--	--	6.12	--
PDP	12.12 Δ	Air	TMT I	99.50	6.41	--	--	--	6.16	--
PDP	12.12 Δ	"Dowtherm" A	TMT II	99.50	5.34	--	--	--	5.57	--
PDP	12.12 Δ	D ₂ O	TMT II	99.50	6.23	--	--	--	6.12	--
PDP	12.12 Δ	Air	TMT II	99.50	6.30	--	--	--	6.03	--
PDP	12.12 Δ	"Dowtherm" A	TMT III	99.50	3.90	--	--	--	3.79	--
PDP	12.12 Δ	D ₂ O	TMT III	99.50	5.11	--	--	--	5.17	--
PDP	12.12 Δ	Air	TMT III	99.50	5.29	--	--	--	5.26	--

^aTHUD I — 85 close-packed THUD rods at 0.310-in. Δ internal pitch.

THUD II — 85 wide-packed THUD rods at 0.351-in. Δ internal pitch.

TMT I — 3 concentric ²³⁵U-Th metal tubes with dimensions:

	o.d., inches	i.d., inches
inner	1.224	0.710
intermediate	2.510	2.034
outer	3.640	3.263

TMT II — Same as TMT I with a 1.210-in.-o.d. graphite cylinder substituted for the inner fuel cylinder.

TMT III — Same as TMT I with the outer fuel cylinder removed.

Nuclear Criticality Safety—I: Plutonium

Measurement of the Critical Mass of a Water-Reflected Plutonium Sphere, W. U. Geer, David R. Smith (LASL)

Several measurements of critical or near-critical reflected plutonium assemblies have been reported for a variety of reflector materials.^{1,2} Although these provide a reasonable estimate of the critical mass to be expected for a water-reflected metal sphere, a direct measurement was considered worthwhile.

A sphere of high purity plutonium was fabricated and machined to a mass of 5546 g with an average density of 19.74 g/cm³ at 20°C. A close-fitting Plexiglas shell about $\frac{1}{16}$ -in. thick provided an hermetic seal for the sphere. A 0.0005-in. coating of copper inside the shell prevented the plutonium alpha particles from decomposing the plastic.

This assembly was placed in a 4-ft-diam $\times$ 4-ft-deep tank in Kiva III of the Pajarito critical facility. Monitoring of the neutron counting rate due to the plutonium provided frequent checks of the multiplication of the sphere as water was slowly added to the tank.

The multiplication of the fully reflected sphere corresponded to a value of k_{eff} of about 0.99.

One-dimensional DTF calculations using the Hansen-Roach 16-group cross sections gave a correlation between the measured leakage multiplication and k_{eff} , and an estimate of the mass increment below delayed critical.

The sphere was then remachined to remove about 200 g, which was expected to double the margin below delayed critical for the reflected sphere. After encapsulation in new Plexiglas shells, the water-reflected multiplication of the sphere was determined again.

Corrections applied to the two measured multiplication values were intended to account for the effects of the plastic shells and the Plexiglas support stand. The values of k_{eff} equivalent to the corrected multiplications, when plotted against the sphere mass, extrapolated to criticality at 5785 g.

Mass spectrometric analysis of a sample of this plutonium provided the following isotopic abundances: ²³⁹Pu, < 0.01 at.%; ²³⁹Pu, 94.5 at.%; ²⁴⁰Pu, 5.18 at.%; ²⁴¹Pu, 0.30 at.%; and ²⁴²Pu, 0.02 at.%.

DTF calculations using the S8 angular approximation result in a value of 1.0037 for k_{eff} of the extrapolated critical sphere, if treated as 94.8% ²³⁹Pu and 5.2% ²⁴⁰Pu. When the indicated bias of 0.37% between calculation and experiment is applied, calculations yield 5425 g as the critical mass of ²³⁹Pu at 19.74 g/cm³. The uncertainty in this value is stated to be $\frac{1}{2}$ %.

The fabrication, processing, and handling of this large piece of plutonium introduced several interesting criticality safety problems which will be discussed.

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2. H. C. PAXTON, "Los Alamos Critical-Mass Data," LAMS-3067, Los Alamos Scientific Laboratory (May 1964).

Measured Critical Spacings of Plutonium Arrays—Part I, G. A. Pierce, J. R. Morton III (UCLRL-Livermore)

The experimental objectives of the Lawrence Radiation Laboratory's Plutonium Array Program are 1) to provide clean critical data to use in the normalization of computational techniques, and 2) to develop data expected to be useful for interpolation and extrapolation for application to actual systems of interest. The most common application is a consideration of the criticality safety of a storage or transportation array of fissile material. This is the first experimental program of critical plutonium metal arrays. An early report of this program is presented by Morton et al.¹

Phase I of the program consisted of small arrays of up to 27 3-kg parts of α -phase plutonium metal of density 19.5 g/cm³, 93.6 wt% ²³⁹Pu. Each part has the form of a right circular cylinder 6.53 cm in diameter and 4.63-cm tall. Each part is packaged inside a close-fitting, deep-drawn, seamless aluminum can with a steel lid. The average part weight is 3.026 $\pm$ 0.008 kg. This is somewhat less than half the critical mass of a reflected cylinder of the same diameter. Thus, two parts may be safely stacked together to produce a 6-kg unit which can also be arranged in arrays.

The near-cubical array geometries constructed in Phase I were assembled on a small split-table assembly machine. The centerpoint of the assemblies was located about 4 m from the nearest concrete wall, minimizing room return.

The assembly table consisted of two movable halves, an incline table and a horizontal table, each with a surface 46 $\times$ 66 cm. Operation consisted of raising the incline table 55 cm at 33 cm/min up a 45° ramp, closing the horizontal table 61 cm at 28 cm/min, and then closing the last 9.5 cm to full closure at 0.7 cm/min.

Before the formal array experiments were begun, it was necessary to develop working criteria for judging the criticality safety of handling and loading operations with the plutonium parts. Two series of experiments were performed for this purpose. The major objective of the first series was to determine the minimum number of 3-kg parts that would become critical when closely reflected by a massive polyethylene slab. This number was determined to be seven.

The primary purpose of the second series of safety experiments was to verify that a fissile system with a measured multiplication of 10 would remain safely subcritical in the presence of added massive hydrogenous reflectors. A given system with a measured multiplication of about 10 gave a measured multiplication of 44 when reflected on two opposite sides by such reflectors.

The three-dimensional arrays were set up in the form of rectangular parallelepipeds, and were generally arranged so that the surface-to-surface separations between adjacent plutonium parts were equal, both horizontally and vertically. Criticality could be achieved with the 2 $\times$ 2 $\times$ 2 and 3 $\times$ 3 $\times$ 3 arrays tabulated in Table I, with and without a side reflector. Moderated arrays were

TABLE I
Summary of Experimentally Determined Critical Spacings

Array (3-kg units)	Critical Surface-to-Surface Separation (cm)		Critical Center-to-Center Separation (cm)	
	Horizontal	Vertical	Horizontal	Vertical
2 × 2 × 2 Bare	0.77 ± 0.03	0.77 ± 0.03	7.30 ± 0.03	5.40 ± 0.03
2 × 2 × 2 Reflected ^a	1.11 ± 0.03	1.11 ± 0.03	8.74 ± 0.03	5.74 ± 0.03
3 × 3 × 3 Bare	3.07 ± 0.03	3.07 ± 0.03	9.60 ± 0.03	7.70 ± 0.03
3 × 3 × 3 Reflected ^a	3.63 ± 0.03	3.63 ± 0.03	10.15 ± 0.03	8.26 ± 0.03

^aOne side reflected with a massive hydrogenous reflector (nominal composition CH₂) 45.0-cm tall × 34.3-cm wide × 20.2-cm thick.

attempted, but moderator geometry prevented achievement of criticality.

1. J. R. MORTON III, G. A. PIERCE, L. L. GARDNER, and C. J. BALL, "Summary Report of Critical Experiments Plutonium Array Studies, Phase I," UCRL-50175, Lawrence Radiation Laboratory, Livermore, California (December 1966).

Measured Critical Spacings of Plutonium Arrays—Part II, H. F. Fimm, G. A. Pierce (UCLRL-Livermore)

Large critical arrays using as many as 128 3-kg plutonium parts have been constructed recently as part of Phase II of the Lawrence Radiation Laboratory Plutonium Array Program. Arrays 4 × 4 × 4 of 3- and 6-kg plutonium parts with and without moderator jackets were assembled on a new split-table machine, built when a smaller assembly machine proved inadequate for arrays larger than 3 × 3 × 3.

The plutonium parts have been described earlier.¹ The moderator jackets which were used in varying thicknesses were made of a plastic formulation similar to a type of moderator likely to be encountered in arrays of fissile material. The chemical composition of this material is 30.6 wt% C, 3.1 wt% H, 31.6 wt% N, and 34.7 wt% O.

The new assembly machine² can accommodate an array 1.8 m on a side with a total mass of over 900 kg. It features a low-mass base plate, rapid spring-aided scram, and highly accurate array element positioning

capability. Closure rates are currently 28 cm/min for 106 cm, followed by 0.76 cm/min for the last 7.8 cm.

The first experiment performed on the new machine repeated the 3 × 3 × 3 3-kg bare array done earlier on the small machine with the result that the experimentally determined critical surface-to-surface spacing was decreased by 0.05 cm. This was undoubtedly due to an increase of 95 cm in the array center-to-table surface distance (from 25 to 120 cm).

For the eight 4 × 4 × 4 array series which followed this experiment, vertical dimensions were fixed at the beginning of each series and only the horizontal dimensions were reduced to arrive at the critical spacings. This eliminated about half the handling operations, with a concurrent reduction in radiation dose to the operations staff.

For two arrays, the multiplication of 10 safety limit pertaining to hand assembly would have prevented completion of the experiment. This rule was supplemented with one allowing hand assembly of systems with multiplications greater than 10, provided that reflection of these systems with body simulators gives rise to multiplications no greater than 50. The body simulators used for these measurements were two upright polyethylene slabs, each 152-cm high × 36-cm wide × 15-cm deep, arranged in the most reactive configuration.

Results of the eight 4 × 4 × 4 experiments are summarized in Tables I and II. The errors are mostly due to positioning uncertainties and control element perturbation. No attempt has been made to correct for the neutronic effect of support structure and room return.

TABLE I
Summary of Critical Parameters, 4 × 4 × 4 3-kg Arrays

Moderator Cell		Horizontal Center-to-Center Critical Spacing (cm)	Vertical Center-to-Center Critical Spacing (cm)
Thickness (cm)	Mass (g)		
0.0	0	12.51 ± 0.01	7.86 ± 0.01
1.27	604 ± 7	14.19 ± 0.01	9.63 ± 0.01
2.54	1663 ± 10	14.55 ± 0.01	13.64 ± 0.01
3.81	3354 ± 30	14.62 ^a ± 0.35	13.63 ± 0.01

^aExtrapolated estimate. Array was terminated at 15.25-cm horizontal center-to-center spacing when moderator cells touched.

TABLE II
Summary of Critical Parameters, $4 \times 4 \times 4$ 6-kg* Arrays

Moderator Cell		Horizontal Center-to-Center Critical Spacing (cm)	Vertical Center-to-Center Critical Spacing (cm)
Thickness (cm)	Mass (g)		
0.0	0	17.28 ± 0.01	17.12 ± 0.02
1.27	931 ± 11	17.50 ± 0.01	25.79 ± 0.02
2.54	2401 ± 9	21.24 ± 0.01	25.82 ± 0.02
3.81	4614 ± 26	24.52 ± 0.02	25.82 ± 0.02

*Vertical center-to-center spacing of the two cylinders making up the 6-kg unit is 5.12 ± 0.02 cm. The gap width between cylinders was 0.48 ± 0.02 cm.

1. G. A. PIERCE and J. R. MORTON III, "Measured Critical Spacings of Plutonium Arrays, Part I," *Trans. Am. Nucl. Soc.*, 11, 378 (1968).
2. A. W. WILLIAMS et al., "Critical Assembly Machine for Plutonium Experiments," *Am. Soc. Mech. Eng.*, 67-WA/NE-3 (November 1967).

Critical Experiments with Homogeneous PuO_2 -Polystyrene at 5 H:Pu, S. R. Bierman, L. E. Hansen, R. C. Lloyd, E. D. Clayton (BNW)

One of the research programs at the Plutonium Critical Mass Laboratory of Battelle-Northwest is concerned with determining the criticality parameters of undermoderated plutonium as a function of moderation and ^{240}Pu isotopic concentration. This program is oriented toward providing experimental data that can be used in establishing criticality safety guides and in checking calculational models over the intermediate neutron energy spectrum. There is a particular need for data¹ against which theoretical analyses of ^{240}Pu effects can be compared.

The initial series of measurements in this program was made on 15 H:Pu fuel having ^{240}Pu isotopic concentrations of 2.2 and 8 wt%. The results from these initial experiments have already been reported^{2,3} as has a description of the experimental facility. This paper is concerned with the results and analysis of the latest series of analogous experiments in which the ^{240}Pu isotopic concentration has been increased to 11.5 wt% and the atomic H:Pu decreased to 5.

Experimental data were obtained from both bare and reflected rectangular parallelepipeds constructed from $2 \times 2 \times 1\frac{1}{2}$ -in. PuO_2 -polystyrene compacts having a plutonium concentration of 2.302 g/cm^3 . The bare critical assembly was near cubic in geometry with a critical buckling of $194.5 \pm 4.0 \text{ m}^{-2}$. The reflected critical assemblies ranged from a near cubic geometry to thin slabs for an infinite slab thickness of 5.98 ± 0.10 cm and a corresponding extrapolation length of 8.27 ± 0.18 cm as shown in Fig. 1.

Using a constant buckling conversion, critical sizes have been derived from these experimental data for spherical, cylindrical, cubic, and slab geometries. These critical dimensions are given in Table I along with the corresponding critical dimensions for ^{239}Pu -water and $^{239}\text{PuO}_2$ -water systems at theoretical densities of 4.16 and 3.46 g/cm^3 , respectively.

Fig. 1. Experimental data from reflected PuO_2 -polystyrene critical assemblies.

1. C. R. RICHEY, "Theoretical Analyses of Homogeneous Plutonium Critical Experiments," *Nucl. Sci. Eng.*, 31, 32 (1967).
2. C. R. RICHEY, J. D. WHITE, R. C. LLOYD, and E. D. CLAYTON, "Homogeneous Critical Experiments with Plutonium/Plastic Mixtures in the Intermediate Energy Range," *Trans. Am. Nucl. Soc.*, 7, 81 (1964).
3. C. R. RICHEY, J. D. WHITE, E. D. CLAYTON, and R. C. LLOYD, "Criticality of Homogeneous Plutonium Oxide-Plastic Compacts at H:Pu = 15," *Nucl. Sci. Eng.*, 23, 150 (1965).

TABLE I
Critical Dimensions in Spherical, Cylindrical, Cubic, and Slab Geometries at 5 H:Pu Atomic Ratio

Geometry	PuO ₂ -Polystyrene 2.302 g Pu/cm ³ 11.46 wt% ²⁴⁰ Pu ^a		²³⁹ Pu-Water 4.158 g Pu/cm ³ 0.0 wt% ²⁴⁰ Pu	²³⁹ PuO ₂ -Water 3.464 g Pu/cm ³ 0.0 wt% ²⁴⁰ Pu
	λ cm	x^b cm	x^b cm	x^b cm
Bare Assemblies				
Infinite Slab	2.85 ± 0.2	16.83 ± 0.31	9.30 ± 0.17	10.40 ± 0.19
Sphere	2.84 ± 0.2	19.68 ± 0.23	11.29 ± 0.13	12.39 ± 0.14
Infinite Cylinder	2.84 ± 0.2	14.40 ± 0.22	8.15 ± 0.12	9.00 ± 0.14
Cube	2.98 ± 0.2	33.05 ± 0.35	----	----
Reflected Assemblies ^c				
Infinite Slab	8.27 ± 0.18	5.98 ± 0.10	3.06 ± 0.05	3.64 ± 0.06
Sphere	8.37 ± 0.35	14.15 ± 0.37	8.39 ± 0.13	9.29 ± 0.14
Infinite Cylinder	8.31 ± 0.35	8.93 ± 0.36	5.12 ± 0.21	5.75 ± 0.23
Cube	8.31 ± 0.35	22.39 ± 0.54	----	----

^aIsotopic concentration.

^bCritical thickness of slab or cube and critical radius of sphere or cylinder.

^cPolystyrene assemblies reflected with Plexiglas; water systems reflected with water.

Criticality of Plutonium Nitrate Solutions in Slab Geometry, R. C. Lloyd, E. D. Clayton, L. E. Hansen (BNW)

Critical thicknesses of infinite slabs of plutonium nitrate solution have been derived from data on spheres and cylinders, but no data appear in the open literature on the direct measurement of critical thicknesses for thin slabs of plutonium nitrate solution.¹ Experiments have been conducted at the Critical Mass Laboratory of the Pacific Northwest Laboratory using a variable-thickness slab-type vessel to determine critical thicknesses of bare and reflected infinite slabs of plutonium nitrate solution.² The experiments provide data for nuclear criticality safety applications in handling and processing plutonium and for checking computational methods.

The criticality vessel shown in the figure is of unique design. It is a slab-shaped vessel with length and height of 42 in., whose thickness can be varied from 3 to 9 in. The variable thickness is made possible by use of a welded stainless-steel bellows around the entire periphery of the slab.

Critical experiments were performed with the 42-in. vessel bare, partially and fully reflected, from which critical thicknesses for the infinite slab could be derived. Assuming constancy of material buckling for a given solution, values for critical bucklings and effective extrapolation lengths were deduced. The effects of the vessel walls, supports, and hood windows were determined experimentally and corrections applied to the data to obtain critical dimensions of clean systems, i.e., bare or water-reflected solutions in the absence of vessel walls, etc.

The results from experiments with dilute plutonium solutions at 58 g Pu/liter (4.6 wt% ²⁴⁰Pu) at excess nitric

Fig. 1. Variable-thickness slab-type vessel.

TABLE I
Criticality Data for Plutonium Solution Slabs
58 g Pu/liter

Description	Experimental Results			Calculated Values	
	Critical Thickness cm	λ^a cm	Buckling ^a m^{-2}	k_{eff} (HFN)	Critical Thickness HFN (cm)
Free Acid Molarity of Solution = 2.3					
Bare ∞ Slab	15.4			0.930	16.91
Bare ∞ Slab/SS	14.7			0.935	16.11
Finite Bare Slab Width = 106.7 cm Height = 98.12 cm	15.75	4.2	185.3	0.949	17.02
Refl ∞ Slab	9.8			1.027	9.16
Refl ∞ Slab/SS	10.2			1.013	9.90
Finite Refl Slab Width = 106.7 cm Height = 101.37 cm	11.18	6.4 ^b	186.4	1.012	10.90
Free Acid Molarity of Solution = 5.0					
Bare ∞ Slab	16.5			0.929	18.18
Bare ∞ Slab/SS	15.8			0.933	17.37
Finite Bare Slab Width = 106.7 cm Height = 96.04 cm	17.02	4.1	171.6	0.946	18.50
Refl ∞ Slab	10.5			1.019	10.08
Refl ∞ Slab/SS	10.9			1.005	10.78
Finite Refl Slab Width = 106.7 cm Height = 87.88 cm	12.20	6.6 ^b	170.2	1.003	12.12
1 in. Lucite Refl ∞ Slab	12.8	5.64	170.5		

^a λ was determined for minimum buckling differences for several critical experimental geometries. Buckling values are averages of those derived.

^b λ for the top of the solution, where an air gap exists above the fuel, was determined to be 5.2 cm to give minimum buckling variation from the several geometries measured.

acid molarities of 2.3 and 5.0 are presented in the table. The change in nitrate molarity from 2.3 to 5.0 is seen to increase the critical slab thicknesses by 1.1 and 0.7 cm for the bare and water-reflected systems, respectively.

The concentration of the ²⁴⁰Pu isotope produces a pronounced effect on the critical thickness of the slab. Experiments currently being conducted with plutonium nitrate solutions at a concentration of 282 g Pu/liter containing 23.3 wt% ²⁴⁰Pu indicate the critical thickness for an infinite water-reflected slab will be nearly double that for ²³⁹Pu at the same concentration.

The HFN multigroup diffusion theory code,³ previously used to predict with good accuracy the criticality of plutonium solutions in spheres,⁴ also was used to compute criticality for the bare and water-reflected slabs at 58 g Pu/liter. The 18-energy group constants used in these calculations were obtained from the GAMTEC-II code.⁵ Computed values of k_{eff} for the measured and deduced critical dimensions are given in the table. The calculations agree quite well for water-reflected slabs, but are nonconservative in predicting the critical thicknesses for bare slabs. In the bare case, k_{eff} is computed to be less than unity, with the largest difference from unity being about 7%. The reasons for the apparent discrepancy are under study, but may be due in part to room reflection.

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2. R. C. LLOYD et al., "Basic Criticality Experiments With Plutonium Nitrate Solutions in Slab Geometry," *Physics Research Quarterly Report January, February, March 1967*, BNWL-472, Battelle-Northwest Labs. (December 1967).
3. J. R. LILLEY, "Computer Code HFN-Multigroup, Multiregion Neutron Diffusion Theory In One Space Dimension," HW-71545, Hanford Works (November 1961).
4. R. C. LLOYD et al., "Criticality Studies of Plutonium Solutions," *Nucl. Sci. Eng.*, 25, 165 (June 1966).
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Nuclear Criticality Safety—II

Critical Experiments at H:²³⁵U Ratios from 0 to 5, D. W. Magnuson (ORNL)

A series of critical experiments was performed at Oak Ridge National Laboratory to investigate the dependence of the critical parameters for small values of H:²³⁵U ratio. The primary motivation for this effort was nuclear criticality safety so that measurements of critical mass would be completed for all ranges of moderation. However, the experiments were done in simple rectangular geometry without other materials or large voids and are ideally suited for comparison to calculations to check various reactor codes.

The unreflected critical experiments were performed for H:²³⁵U ratios from 0 to 5 using layers of uranium metal enriched to 93.15 wt% ²³⁵U and polyethylene or Plexiglas moderators. Polyethylene-moderated and -reflected experiments have also been completed for the same range of moderation. In addition, some unreflected Teflon-moderated experiments were performed. Base dimensions of either 25.4 × 25.4 or 12.7 × 25.4 cm were used and a preselected uranium layer arrangement was made critical by adding or removing moderator. Calculations of the neutron multiplication factor k_{eff} , using the KENO Monte Carlo code¹ with revised Hansen-Roach 16-group cross sections,² were within ± 0.01 for the bare assemblies. Calculations for the polyethylene reflector and moderated assemblies are between 1.5 and 2.0% higher than experiment. Other reflected experiments have shown similar discrepancies.³

An additional comparison of the data to calculated critical masses was made in spherical geometry. Radii of equivalent spheres were obtained by equating bucklings in spherical and the experimental rectangular geometries in which appropriate values of the extrapolation distances and reflector savings were used. The masses derived

from the radii were corrected for small differences in actual and theoretical densities and for small deviations of the experiments from criticality and are then compared in Fig. 1 to critical masses calculated by the one-dimensional multigroup transport code⁴ ANISN-S₈. Revised Hansen-Roach 16-group cross sections were also used for these calculations.

Both experiments and KENO code calculations have shown that the neutron multiplication factor for the heterogeneous assemblies were no more than 1% greater than k_{eff} for homogenized materials. These experiments will be described more fully in an ORNL report.⁵

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The Effect of Steel-Water Reflectors on the Criticality of Low-Enriched Uranyl Fluoride Solution, E. B. Johnson, C. E. Newlon (ORNL)

The effect of composite reflectors of steel and water on the reactivity of single cylinders of aqueous solution of low-enriched uranium has been studied. The results are applicable in the evaluation of the nuclear criticality safety of shipping containers and provide bases for the verification of calculational models.

The solution was contained in each of two stainless-steel cylinders having 0.079-cm-thick walls; the solution diameters were 33.02 and 39.09 cm. The thickness of lateral steel reflectors adjacent to the solution ranged up to 5.08 cm and was surrounded by an infinitely thick water reflector. There was no top or bottom reflector. Similar experiments using U(93.3)O₂(NO₃)₂ have been reported.¹ Table I summarizes the data.

It was found that the presence of 1.27-cm-thick steel outside either container resulted in the maximum decrease in reactivity from that with water alone; this reactivity change was approximately \$2.8 and \$1.8 for the 33.02- and 39.09-cm-diam cylinders, respectively. Further increase in steel thickness up to 5.08 cm increased the reactivity, but not to that existing with water alone. Interposition of 0.081-cm-thick cadmium between the 39.09-cm-diam solution cylinder and the steel decreased the reactivity of the cylindrical volume from that without the cadmium; however, the presence of cadmium between the 5.08-cm-thick steel reflector and the water resulted in no reactivity change. These latter measurements emphasize the importance of the relative locations of neutron-absorbing materials in the evaluation of the nuclear criticality safety of water-reflected containers.

Fig. 1. Comparison of spherical critical masses to values calculated by ANISN-S₈ transport code.

TABLE I

Experimental Critical Heights and Computed k_{eff} of Cylinders of $U(4.98)O_2F_2$ Solution with Lateral Composite Steel-Water Reflectors*

	Reflector Steel Thickness, cm						
	0	0.64	1.27	1.90	2.54	3.81	5.08
33.02-cm-diam Cylinder							
Composite steel-water reflector ^{a,b}							
Critical height, cm	84.60	158.85	201.60	172.80	143.00	109.10	91.95
k_{eff}^c	1.009	0.999	0.995	0.998	0.998	1.001	1.003
39.09-cm-diam Cylinder							
Steel reflector (no water) ^d							
Critical height, cm	98.80	74.65	64.75	59.30	55.70	50.90	48.20
k_{eff}^c	1.000	1.003	1.007	1.009	1.011	1.013	1.016
Composite steel-water reflector ^{a,d}							
Critical height, cm	44.70	48.70	49.40	48.95	48.20	46.50	45.10
k_{eff}^c	1.012	1.004	1.003	1.004	1.005	1.006	1.006
Composite 0.081-cm-thick cadmium-steel-water reflector ^{d,e}							
Critical height, cm	64.90				54.45		50.80
k_{eff}^c	1.002				1.011		1.008

*The solution concentration was 909.1 g U/liter ($H^{235}U = 488$). The cylindrical container was 0.079-cm-thick type-304 stainless steel with no top or bottom reflector.

^aThe steel was adjacent to the solution cylinder; the thickness of the water was effectively infinite.

^bThe steel reflector and the water had a common height of 243.9 cm measured from the same reference as the solution height.

^cValues computed by ANISN Code.

^dThe steel reflector and the water had a common height of 106.2 cm measured from the same reference as the solution height.

^eThe cadmium was between the solution container and the steel.

A one-dimensional transport theory code ANISN² with the Hansen-Roach cross-section set³ and a buckling approximation of the form DB^2 for the finite longitudinal cylinder dimension, were used for the analysis of the experimental data. The largest difference between computed and experimental values of k_{eff} was 0.016, as shown in the table. The code also permits calculation of the neutron current densities at the fuel-reflector interface and, therefore, provides an indication of the number and energy of the neutrons that are returned to the fuel region. The computations thus indicate that the changes in reactivity of the solution cylinders induced by the steel reflector are primarily the result of the competition between the effective decrease in the reflection of the thermal component and the effective increase in the reflection of the epithermal component of the neutron flux at the fuel-reflector interface with increasing thicknesses of steel.

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Criticality Calculations for Fuel Storage Arrays in Water, J. K. Fox (Id Nucl Corp)

Spent fuel elements from power reactors are stored under water. Criticality calculations on the storage arrays must assume an infinite lattice since large numbers of elements are usually present. About 12 to 14 in. of water between fuel elements reduces neutron interaction currents to zero.¹ In this case the array is critical only if individual elements are also critical. Appreciable interaction currents are required for array criticality when the individual elements are far from critical alone. This suggests that safe lattice separations can be expressed in terms of the k_{eff} of isolated fuel elements.

Infinite array system k_{eff} values were obtained by solution of a unit cell problem. Four-group diffusion theory was used in the study. Five fuels were chosen representing a variety of assays. Constants for the three fast groups were obtained using a B_1 approximation in PHROG.² Thermal range constants were obtained with the INCITE³ code which uses the integral B_1 approximation. For the unit cell calculations, it was assumed that the fuel was in an hexagonal array and consisted of cylindrical elements surrounded by thin aluminum cans and then water. The true unit hexagonal cell was made into a cylinder of equal area. This allows one to use FOG,⁴ a one-dimensional code for k_{eff} calculations.

Fig. 1. Safe element separations in an infinite lattice vs isolated element k_{eff} values for several fuels.

The determination of safe lattice separations as a function of the k_{eff} of the individual elements when isolated required two series of preliminary calculations for each fuel. First, unit cell problems were run to determine separation as a function of cylinder diameter for a given array k_{eff} , using a dimension search option in FOG. Second, the k_{eff} of individual elements surrounded by infinite water were determined as a function of diameter. From cross-plots of these data, the curves shown in Fig. 1 were constructed.

If one limits the calculated array k_{eff} to 0.90, for criticality safety, then curves of safe spacing vs the isolated element k_{eff} approach 0.90 instead of unity at large separations. This is expected since the interaction approaches zero. The safe spacing is strongly dependent upon the element reactivity. Most power reactors require large numbers of elements for criticality; hence, the k_{eff} of their isolated elements is small and safe spacings may be much less than 12 in.

A description of the five fuels is as follows:

- Fuel #1 - 93% enriched UO_2F_2 in water at $H/^{235}U = 45$.¹
- Fuel #2 - 93% enriched UO_2F_2 in water at $H/^{235}U = 331$.¹
- Fuel #3 - 93% enriched SPERT D fuel elements with 0.307 kg ^{235}U in a 3- x 3-in. element 24-in. long.¹
- Fuel #4 - 5% enriched UO_2F_2 in water at $H/^{235}U = 496$.⁶
- Fuel #5 - 2% enriched UF_4 dispersed in paraffin at $H/^{235}U = 195$.⁷

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Monte Carlo Calculations of Thick-Graphite-Reflected Uranium-Metal Critical Assemblies, J. T. Mihalcz (ORNL)

The neutron multiplication factor for a variety of uranium-metal annuli and cylinders surrounded with thick carbon reflectors has been measured and calculated by a Monte Carlo method. The material within the annuli was carbon or air. The thickness on the side, top, and bottom reflector was equal in any one assembly. The density of the individual pieces of Type-HLM graphite varied between 1.688 and 1.783 g/cm³. The uranium metal was enriched to 93.2% in ^{235}U and has been described previously.¹

The multiplication factors were calculated using the 16-group cross sections of Hansen and Roach² for uranium and carbon and the KENO code³ which is a multigroup Monte Carlo method with isotropic scattering and with neutron slowing down treated by a transfer matrix. Twenty-one thousand neutron histories (105 batches of 200 source neutrons each) were calculated requiring a time of ~20 min on the IBM 360-75 for each problem. The source neutrons were put in the uranium uniformly and the first five batches were discarded.

The neutron weight below which neutrons were killed by Russian Roulette was a function of the distance in the reflector from the core. This function was inversely proportional to the adjoint flux, obtained from S_N transport theory calculations of a uranium-metal sphere surrounded by an 18-in. carbon reflector. This function was used for all assemblies. The standard deviations on the multiplication factors are between 0.9 and 1.3%.

These assemblies all have one small uranium dimension and would require a high-order S_N calculation for a good transport theory solution. The running time for S_N transport theory calculations using the DOT code⁴ for these assemblies is ~90 min.

The results of the experiments and calculations are given in Table I. Since the calculated multiplication factors are, on the average, about 0.6% higher than the experimental ones, a slight adjustment of the cross sections may be required for exact agreement with experiment.

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TABLE I
Carbon-Reflected Uranium-Metal Annuli and Cylinders

Uranium o.d. (in.)	i.d. (in.)	Reflector Thickness (in.)	Material Within the Annulus	Height of Fuel for Radial Increments of ^a (in.)					Average Density (g/cm ³)		Multiplication Factor	
				0-3.5	3.5-4.5	4.5-5.5	5.5-6.5	6.5-7.5	Uranium	Carbon	Measured ^b	Calculated
15	0	15	---	0.749	0.687	0.691	0.689	0.685	18.734	1.756	1.000	1.018 ± 0.012
15	11	15	Carbon	--	--	--	1.695	1.696	18.722	1.760	1.000	0.999 ± 0.011
15	9	15	Carbon	--	--	1.170	1.191	1.172	18.717	1.762	1.001	1.011 ± 0.011
13	9	16	Carbon	--	--	1.756	1.755	--	18.727	1.757	1.002	1.005 ± 0.011
11	9	17	Air	--	--	4.019	--	--	18.706	1.757	1.002	0.987 ± 0.009
9	0	18	---	1.256	1.288	--	--	--	18.678	1.757	1.000	1.018 ± 0.009
9	7	18	Air	--	4.386	--	--	--	18.727	1.757	1.000	1.012 ± 0.013
9	7	18	Carbon	--	4.069	--	--	--	18.745	1.757	1.001	1.009 ± 0.011

^aThese dimensions show the unevenness of the upper surface of the assembly.

^bCorrected for the effect of support structure. This correction was less than 3¢ in reactivity.

Monte Carlo Calculated Criticality of Sub-critical ²³³U-Metal Spheres in Cubic Geometry, J. T. Thomas (ORNL)

The criticality of individually spaced subcritical quantities of ²³³U metal is being investigated currently by a calculational program supported by the Criticality Data Center of the Oak Ridge National Laboratory. The information is necessary to safety specifications for handling large but realistic quantities of the isotope and is to be incorporated in the revision of the Nuclear Safety Guide.¹

Since no experiments have been performed with spaced quantities of ²³³U metal, reliance must be based upon validated calculational techniques. The demonstrated² ability of Monte Carlo codes to properly handle the geometry of air-spaced components of various materials makes them particularly suited to this problem. The only experiments with ²³³U metal were performed at Los Alamos³ on a single sphere with various neutron reflector conditions. The experiment determining the criticality of an unreflected metal sphere is used to satisfy the condition that the code utilized in the study be capable of computing a critical configuration of the material under consideration.

Calculations by the KENO Monte Carlo code⁴ and the Hansen-Roach 16-group cross-section set⁵ gave 1.0017 as

TABLE I
Calculated Critical Unreflected Arrays of ²³³U-Metal Spheres

Number of Spheres ^a in Array, N	Sphere		Center-to- Center Separation (cm)	Array Uranium Density ^b (g/cm ³)	k _{eff} from KENO Code ^c
	Radius cm	Mass kg U			
64	4.672	7.88	18.080	1.333	0.995
216	4.672	7.88	22.661	0.677	1.002
8000	4.672	7.88	42.315	0.104	1.008
64	4.404	6.60	15.620	1.732	0.995
216	4.404	6.60	19.518	0.888	0.994
349	4.404	6.60	21.185	0.694	1.000
64	3.998	4.94	12.514	2.514	0.996
216	3.998	4.94	15.626	1.294	1.001
512	3.998	4.94	18.182	0.822	1.011
64	3.573	3.53	9.952	3.576	1.002
216	3.573	3.53	12.511	1.800	1.006
1000	3.573	3.53	16.417	0.797	1.005
216	3.386	3.0	11.290	2.085	1.000
64	3.186	2.5	7.967	4.944	1.003
64	2.958	2.0	6.985	5.868	1.007

^aN ¹/₂ spheres arranged along each edge of array.

^bUranium density in unit cell of the lattice.

^cNeutron multiplication factors have a standard deviation of ± 0.005.

the ratio of computed-to-experimental critical radii of an unreflected ²³³U sphere. This may be compared to 0.9973 obtained from the DSN transport code with an S₈ approximation⁶ and the same cross sections. The uranium had a density of 18.45 g/cm³ and contained 98 wt% ²³³U.

The separation of sphere centers required for the criticality of various numbers of spheres arranged in a cubical array was established utilizing the KENO code. The results are given in Table I for several sphere radii along with the determined multiplication factor.

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Monte Carlo Calculations for Arrays of Plutonium Units,* N. Pruvost, G. Pefley (UCLRL-Livermore)

The ability of the Monte Carlo technique to calculate critical configurations of arrays of air-spaced units is of fundamental importance for many problems that arise in criticality safety. However, the reliability of the technique has not been demonstrated for arrays whose units are themselves quite complicated in geometry and materials. The purpose of this paper is to show that for some arrays with units moderated and partially decoupled by interstitial material, a detailed Monte Carlo calculation is capable of predicting critical configurations.

*Sponsor: Oscar C. Kolar.

TABLE I
05R Calculations of k_{eff} for Eight $4 \times 4 \times 4$ Plutonium Arrays*

Plutonium Unit Cell Mass (kg)	Moderator Cell		Horizontal Center-to-Center Critical Spacing (cm)	Vertical Center-to-Center Critical Spacing (cm)	Calculated k_{eff} (05R)
	Thickness (cm)	Mass (g)			
3	0	0	12.51 ± 0.01	7.86 ± 0.01	0.97 ± 0.01
3	1.27	604 ± 7	14.19 ± 0.01	9.63 ± 0.01	0.99 ± 0.01
3	2.54	1663 ± 10	14.55 ± 0.01	13.64 ± 0.01	1.01 ± 0.01
3	3.81	3354 ± 30	14.62 ± 0.35^a	13.63 ± 0.01	1.01 ± 0.01
6	0	0	17.28 ± 0.01	17.12 ± 0.02	0.98 ± 0.01
6	1.27	931 ± 11	17.50 ± 0.01	25.79 ± 0.02	1.00 ± 0.01
6	2.54	2401 ± 9	21.24 ± 0.01	25.82 ± 0.02	0.98 ± 0.01
6	3.81	4614 ± 26	24.52 ± 0.02	25.82 ± 0.02	1.00 ± 0.01

*Vertical center-to-center spacing of the two cylinders making up the 6-kg unit was 5.12 ± 0.02 cm. The gap width between cylinders was 0.48 ± 0.02 cm.

^aExtrapolated estimate. Array closure was terminated at 15.25-cm horizontal center-to-center spacing when moderator cells touched.

The values of k_{eff} have been calculated for eight nearly cubical $4 \times 4 \times 4$ experimentally determined critical arrays, using the 05R neutron Monte Carlo code.¹ Each array is composed of identical units containing either 3- or 6-kg plutonium parts, unmoderated or surrounded by moderating material typical of the type likely to be encountered in dealing with arrays of fissile material. The basic array part is a 3-kg mass of α -phase plutonium (93.6 wt% ^{239}Pu) in the shape of a right circular cylinder with height of 4.63 cm and diameter of 6.53 cm. Plutonium density is 19.5 g cm^{-3} . A 6-kg array part is constructed by placing two 3-kg cylinders end to end to form a cylinder with twice the height-to-diameter ratio of a single part. Array units are composed of 3- or 6-kg plutonium parts, unmoderated or surrounded by cylindrical moderator jackets of thicknesses 1.27, 2.54, or 3.81 cm. Moderator composition is 30.6 wt% C, 3.1 wt% H, 31.6 wt% N, and 34.7 wt% O, with a density of 1.54 g cm^{-3} .

Calculated results for eight unmoderated and moderated $4 \times 4 \times 4$ arrays are presented in Table I. Cross-section data for these calculations are taken from the Howerton Evaluated Library and are represented in the calculation by 750 energy groups between the energies of 10 MeV and 1 eV. A simple one-energy-group model is used for neutrons with energies below 1 eV. Inelastic scattering is described by an evaporation model with the scattered neutron emitted isotropically in the center-of-mass system. Cross-section data for elastically scattered neutrons were handled by expansion through the 16th Legendre polynomial. Since, in a given array, each array unit is identical and regularly spaced with respect to its neighbors, the array geometry is simulated in 05R by specifying that a single cell, with periodic boundary conditions, is to be repeated four times in each direction. The IBM-7094 machine time used to obtain a 1% calculated deviation in k_{eff} ranges between 10 min for the 3-kg unmoderated array to 60 min for the 3-kg array with 3.81-cm-thick jackets.

Approximately 26% of the neutrons born live to reach 1 eV in the most thermal of these arrays (3-kg, 3.81-cm moderator jacket). These thermal neutrons in turn cause about 13% of the fissions. The error in the calculated k_{eff} is therefore strongly dependent on the accuracy of the simple thermal treatment.

Auxiliary calculations estimate the change in k_{eff} per change in cell volume to be of the order $\Delta k_{eff}/\Delta V_{cell} \approx$

-10^{-4} cm^{-3} for these systems. Since for small changes in critical cell shape the critical cell volume remains about constant, an estimate of the error in calculated critical spacing is $\Delta V_{cell} = \Delta k_{eff}/|-10^{-4}| \approx 10^{-2}/10^{-4} = 10^2 \text{ cm}^3$.

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The Application of the HAMMER Computer Code to Ex-Reactor Criticality Calculations, Ronald A. Chitwood (Douglas United Nucl)

The determination of the criticality parameters of fissile material arrays is continually needed to establish criticality safety limits for fabricating, handling, and storing that material. Often the criticality condition of a system may be determined by using the experimental data that have been generated for assemblies of similar size and enrichment. Seldom does a new fuel design exactly match anything that has been done experimentally, and subsequent conservative approximations lead to unnecessary restrictions in the process. Sometimes an alternate to an experiment would be the use of proven calculational techniques.

The HAMMER system,¹ a computer code developed by J. E. Suich and H. C. Honeck, has been employed with some success on arrays of slightly enriched uranium tubes in light water.² The agreement between calculations and benchmark experiments has been good enough to justify using the calculations as the basis for some criticality safety limits at Douglas United Nuclear. At this stage of development it is probably premature to extend the calculations to fuel designs that differ greatly from experimental check points. However, the criticality parameters generated from HAMMER calculations have been conservative in all systems examined to date.

Table I shows the results of both the calculations and experiments over a rather wide geometry change. While optimum lattice spacing for each fuel geometry has been established, only maximum buckling points are used here for comparison. Triangular patterns were used for the fuel arrays because a lower critical mass is attainable with them. One significant anomaly is the poor agree-

TABLE I

Fuel Geometry	Fuel Enrichment and Dimensions	Calculations	Experiment
1.95 w/o U-235 OD 7.2 in. ID 2.6 in.	Calculated Critical Cylinder at Optimum Pitch Using Hammer Flog Option Yielded an Effective Critical Radius of 32.3 cm.	Experimental Critical Radius at Optimum Pitch had an Effective Critical Radius of 32.5 cm. (6)*
3.0 w/o U-235 OD 3.0 in.	$V_{H_2O}/V_u = 1.01$ $B_m^2 = 9436 \mu b$ $\lambda = 6.0 \text{ cm}$	$V_{H_2O}/V_u = 1.0$ $B_m^2 = 9670 \mu b$ $\lambda = 5.6 \text{ cm}$ (7)*
Outer/Inner 1.25 w/o/0.95 w/o OD Clad 2.406/ 1.259 in. OD Fuel 2.351/ 1.168 in. ID Fuel 1.815/ 0.482 in. ID Clad 1.763/0.438 in	$V_{H_2O}/V_u = 1.94$ $B_m^2 = 4486 \mu b$ $\lambda = 6.4 \text{ cm}$	$V_{H_2O}/V_u = 1.80$ $B_m^2 = 3900 \mu b$ $\lambda = 7.25 \text{ cm}$ (8)*
Outer 1.25 w/o U-235 OD Clad 2.406 in. OD Fuel 2.351 in. ID Fuel 1.815 in. ID Clad 1.763 in.	$V_{H_2O}/V_u = 2.67$ $B_m^2 = 4894 \mu b$ $\lambda = 6.3 \text{ cm}$	$V_{H_2O}/V_u = 2.25$ $B_m^2 = 4680 \mu b$ $\lambda = 7.93 \text{ cm}$ (8)*
Driver 2.1 w/o U-235 OD Clad 2.397 in. OD Fuel 2.327 in. ID Fuel 1.768 in. ID Clad 1.718 in. Core $Li_2O-Al_2O_3$ OD 1.075 in.	$V_{H_2O}/V_u = 2.63$ $B_m^2 = 2646 \mu b$ $\lambda = 6.9 \text{ cm}$	$V_{H_2O}/V_u = 2.61$ $B_m^2 = 1490 \mu b$ $\lambda = 9.08 \text{ cm}$ (2)*
Driver 2.1 w/o U-235 OD Clad 2.397 in. OD Fuel 2.327 in. ID Fuel 1.768 in. ID Core 1.718 in.	$V_{H_2O}/V_u = 3.12$ $B_m^2 = 10,334 \mu b$ $\lambda = 6.17 \text{ cm}$	$V_{H_2O}/V_u = 2.6$ $B_m^2 = 9050 \mu b$ $\lambda = 7.76 \text{ cm}$ (2)*

* Numbers Refer to Reference Report.

ment between the experiment and the calculations for the 2.1 wt% ^{235}U tube with the $Li_2O-Al_2O_3$ core. This year Douglas United Nuclear is supporting an experiment with this lattice in the Physical Constants Test Reactor operated by Battelle Northwest Laboratories. This program should aid in resolving the problem.

Calculations using the HAMMER system have extended to more complicated problems where HAMMER was used in conjunction with other codes to obtain a satisfactory solution. A specific example is the New Production

Reactor fuel shipping cask that has a center void divided into four sections by a boron steel cruciform. Three poisoned canisters, each containing 12 fuel elements, are stacked into each cask section for a total of 144 elements in the cask. Previous experimental² results had indicated that criticality could be achieved under optimum conditions (without poison) with approximately 25 of the 2.1 wt% ^{235}U fuel elements. Using HAMMER/Head,³ four-group, region cross sections were obtained and then used in a two-dimensional transport theory code 2DF.⁴ The results from 2DF predicted a k_{eff} for the system of

0.84. A subsequent experiment was performed to confirm the calculation. The experiment established that the cask was subcritical, to the extent that no meaningful neutron multiplication data were observed during the loading.⁵ Other similar calculations and experiments were performed with this cask with the same results. While the experimental results were limited, they did show that calculational techniques may prove adequate to determine the criticality condition of even nonuniform poisoned lattices.

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Unreflected Plexiglas-Graphite-Uranium Critical Measurements*, J. Carlton Hoogterp, (LASL)

This series of measurements was stimulated by criticality questions associated with the processing and handling of graphite-uranium fuel for Rover reactors. It is directed specifically toward mixtures of this fuel with water or other hydrogenous material, in the hope of reducing large uncertainties attached to computed critical masses that had been used for safety guidance.

To simulate near-homogeneous mixtures of Rover fuel and water, unreflected critical parallelepipeds of U(93.16) moderated with Plexiglas or Plexiglas-graphite combinations were set up within the aluminum matrix of a split-table assembly machine. Initially, U(93.16) foils (0.002- to 0.012-in. thick) were interleaved with 1/16-in.-thick Plexiglas plates to yield H:²³⁵U ratios of ~6, 12, and 36. Graphite plates (0.180- or 0.280-in. thick) were added later to obtain overall C:²³⁵U ratios in the neighborhood of 24, 48, and 98. In some cases, effects of density changes were investigated, and in others, various shapes were intercompared.

Measured reactivity contributions of the various materials provided the basis for minor corrections of composition, and for eliminating effects of control-rod perturbations. Observed critical volumes were corrected to spherical shape for comparison with one-dimensional transport calculations using Hansen-Roach cross sections.¹ The shape conversions made use of Stratton's empirical expression for extrapolation distance²

$$\delta_1(\text{cm}) = 1.17 + 0.22 \log l^3,$$

where l is a dimension, such as length or diameter, in centimeters.

Table I lists the observed critical parameters and the critical volumes of equivalent spheres. These spherical volumes, after correction to "standard" composition, are compared with computed values in Table II. The

*Sponsor: Hugh C. Paxton

TABLE I
Critical Parameters of Plexiglas-Graphite-U(93) Assemblies

	$\rho(^{235}\text{U})$	Atomic ^a Ratio		Dimension (in.)			Critical Volume (liters)	
	(g/cm ³)	H: ²³⁵ U	C: ²³⁵ U	L	H	W	Observed	Equivalent Sphere
1	2.303	6.0	3.76	23.5	12	9.26	42.8	25.8
2	2.096	6.0	3.74	23.5	12	10.10	50.8	33.3
3	1.317	12.1	7.6	15	12	11.38	33.6	27.6
4	0.480	35.1	21.9	15	12	12	35.4	29.3
5	0.917	6.0	24.3	23.5	18	18	124.8	100.1
6	0.521	6.0	48.5	23.5	24	27.53	254.4	207.5
7	0.258	12.3	98.7	23.5	28.5	28.70	315.0	254.8
8	0.258	12.4	98.2	32	24.75	24.65	319.8	253.0
9	0.258	12.4	98.2	32	13.61R ^b		305.1	262.7
10	0.336	35.2	48.2	15	15	16.69	61.5	51.5
11	0.223	35.5	99.4	15	21	21	108.4	84.6

Honeycomb Al matrix throughout core and as incidental reflector: $\bar{\rho}(\text{Al}) = 0.165 \text{ g/cm}^3$.

^aHydrogen content obtained from Plexiglas (C₅H₈O₂).

^bPseudo-cylinder.

TABLE II
Comparison of Adjusted Sphere Data and Computed Critical Volumes

Adjusted Parameters			Critical Sphere Volume (liters)	
$\rho(^{235}\text{U})$ (g/cm ³)	H: ²³⁵ U	C: ²³⁵ U	Adjusted Value	Computed ^a
2.3	6	3.75	26.0	26.5
2.1	6	3.75	33.0	34.0
1.3	12	7.5	29.0	28.8
0.48	36	22.5	28.2	28.0
0.925	6	24	99.9	106.3
0.52	6	49	208.0	216.2
0.25	12	98	2,67.1, 265.3, 274.7	272.7
0.33	36	48	52.2	53.9
0.23	36	98	81.7	80.9

^aAluminum from assembly machine and oxygen from Plexiglas (C₅H₈O₂) included.

average difference between the compared critical volumes, about 2½%, is less than one-tenth the uncertainty that had been attached to computed values.

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Fast Critical Measurements and Analyses

Fluxes Intrastructure in a Large, Plate-Type Heterogeneous UO_2 Fast Core (Assembly-6, ZPR-6), R. A. Karam, J. E. Marshall, T. Nakamura* (ANL)

The spatial intrastructure of the fluxes in a unit cell of the plate-type heterogeneous criticals is used to average the broad-group cross sections¹ over space to account for heterogeneity effects. The adequacy of this procedure was tested by measuring the fission rates in ^{235}U and ^{238}U as a function of position in a unit cell at the center of Assembly-6 of ZPR-6.²

Depleted uranium foils (0.2% enrichment, 5-mils thick by $\frac{1}{8}$ -in. diam) and ^{235}U foils (93% enrichment, 4.5-mils thick by $\frac{1}{8}$ -in. diam) were placed at corresponding locations about the center of the two halves of the ZPR-6 machine.³ Foils were placed among the adjoining surfaces of the plates. In another experiments a $2 \times 2 \times \frac{1}{8}$ -in. plate of 93% enriched uranium and a $2 \times 2 \times \frac{1}{8}$ -in. plate of depleted uranium (0.2% enriched) that are normally in the core were replaced by packets of foils equivalent in weight and enrichment to the two plates. All foils were irradiated for one hour at a power level of about 50 W and subsequently counted for fission fragments.⁴ In the case of the packets, only $\frac{1}{8}$ -in.-diam foils punched out from the

center of the $2 \times 2 \times 0.005$ -in. foils comprising the packets, were counted. This procedure eliminated any requirement for edge effect corrections. A few foils were radiochemically analyzed for absolute fissions. In another experiment the reactivity worth of ^{235}U as a function of sample thickness was measured.

The results are given in Figs. 1 and 2. The solid and dashed lines were calculated with the SNARG code⁵ with SN order of 32 using the modified Gaussian quadratures⁶ and a 22-group set generated for Assembly-6 with the MC² code⁷ from ENDF/B data. The results show: 1) the peaking of the high-energy fluxes are measured by ^{235}U fission rates and the depression of the low-energy fluxes in the ^{235}U plate combine to give net negative heterogeneity effect (associated with ^{235}U plates only) on the system; 2) the fission rate in ^{235}U within the plate is flat (Fig. 2); 3) general agreement between calculated and measured spatial distribution of the fission rates; and 4) the calculated fission rate in ^{235}U per fission in ^{238}U is about 20% lower than the measured value regardless of the position within the cell. Independent measurements also showed that the reactivity worth of ^{235}U decreased by 10% as the thickness is increased from 5 mils to that of a $\frac{1}{8}$ -in.-thick plate.

Fig. 1. Fission rate intrastructure in cell of Assembly-6 of ZPR-6.

Fig. 2. Fission reaction rates in depleted and enriched uranium foil packets simulating plates in Assembly-6 of ZPR-6.

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Analysis of Fast Critical Assemblies and Large Fast Power Reactors with Group-Constant Sets Recently Evaluated at Karlsruhe, E. Eiseemann, H. Kuesters, M. Metzenroth, J. J. Schmidt, K. E. Schroeter, D. Thiem (KFZ-Karlsruhe)

This report deals with three topics: 1) Are the conclusions^{1,2} in calculating ^{235}U -fueled fast critical assemblies confirmed by further experiments?; 2) What are the results obtained for Pu-fueled criticals if one includes the recent higher Pu- α values?; and 3) What are the consequences for large power reactors?

The bases of our investigations are the 26-group sets KFK-SNEAK(NAP), H20PMB(NAPPMB), and H20PMB+ α (NAPPMB+ α). The two sequences SNEAK and NAP differ by the weighting spectrum corresponding to steam- or sodium-cooled fast reactors in calculating the group constants. Detailed information is given by Huschke et al.³ The Pu- α values were chosen as lower limits of the results of Schomberg⁴ and are included in the H20PMB+ α and NAPPMB+ α -sets.

TABLE I

The α -Values for ^{239}Pu as Introduced in the " α -Sets"

Groups [keV]	0.465-1.0	1.0-2.15	2.15-4.65	4.65-10.0	10.0-21.5
$\bar{\alpha}$	0.95	0.99	0.98	0.88	0.64

In Table II the results are shown for k_{eff} obtained with the appropriate sets for several fast critical facilities. (For more detailed information see Eisemann et al.⁵)

TABLE II
 k_{eff} for Several Fast Criticals

FAC \ Set	SNEAK NAP	H20PMB NAPPMB	H20PMB+ α NAPPMB+ α	Remarks
SNEAK 3A-1	0.995	0.989	- - -	H content 0.035 g/cm ³ } ^{235}U fueled H content 0.07 g/cm ³ }
SNEAK 3A-2	0.989	0.986	- - -	
SNEAK 3B-2	0.983	0.989	0.974	H content 0.07 g/cm ³ Pu/ ^{235}U fueled Pu fueled { These values are calculated with the NAP-sequence sets.
ZPR 3/48	0.979	0.990	0.974	
ZEBRA VIa	0.970	0.988	0.965	

The σ_a -background concept for calculating resonance self-shielding of the capture and fission resonances may be in error by ~ 0.1 to 0.2% in k_{eff} ; the errors due to an improved correction for anisotropic down scattering are about as large. These relatively small corrections cannot account for the large discrepancies observed for Pu-fueled assemblies. The recent and probably more reliable results⁶ for α -Pu will give only a small improvement.

From the comparisons touched in this abstract we are going to reevaluate our basic data for the capture, fission, and inelastic scattering of the heavy isotopes.

For sodium, steam- and gas-cooled fast reactors of 1000 and 300 MW(e), the most important integral quantities have been calculated with the different sets and are discussed in detail⁵ especially with respect to consequences for the fast breeder development. Here we note only that the loss in breeding ratio by ~ 0.08 , 0.09 , and 0.07 for the three types, respectively, is partially compensated if one takes into account the corresponding quasi-stationary Pu composition as has been shown recently.¹⁰

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The values in Table II are obtained with two-dimensional diffusion calculations corrected for heterogeneity⁶ and transport effects, the effective macroscopic elastic removal cross section being determined by method B from Kuesters et al.¹ The overall sufficient agreement with experiment for the SNEAK 3A-1 assembly, previously reported,^{1,2} is no longer preserved and becomes increasingly worse with increasing softening of the spectrum which is also confirmed by preliminary results in SNEAK 3A-3 with an even higher hydrogen content. This is one of several indications that the capture of ^{235}U below 10 keV is too high in our group sets in agreement with results of recent measurements.⁷ No clear decision can be made whether the PMB-data⁸ have to be favored.

It is known that $\bar{\sigma}_f(^{238}\text{U})/\bar{\sigma}_f(^{235}\text{U})$ is underpredicted theoretically. To a large part, this is due to the fact that in the broad-group structure the group constant $\sigma_f(^{238}\text{U})$ between 0.8 and 1.4 MeV is very sensitive to the weighting spectrum. Using the measured neutron spectrum of 3A-1, there results an $\sim 25\%$ higher group constant, giving an increase in k_{eff} by $\sim 0.4\%$ for 3A-1.

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Monte Carlo Calculations of Two-Core Delayed Critical Assemblies, J. T. Mihalczo (UCC-Y12)

To determine the adequacy of the assumption of loose coupling,¹ i.e., the use of properties measured in isolated cores to describe the behavior of coupled assemblies, the results of Monte Carlo² calculations of two-core delayed critical assemblies of enriched uranium (93.2 wt% ^{235}U) are compared with experiment.³ In a loosely coupled two-core assembly the multiplication factor of a

TABLE I
Monte Carlo Calculations of Two-Core Delayed Critical Assemblies.

Uranium		Multipli- cation Factor ^a	Mean Time to ^b		Coupling Reactivity ^c Δk	Number of Neutrons Crossing per Source Neutron	Probability that Neutron Incident on One Core Will Be Reflected to the Other
Thickness (cm)	Separation (cm)		Fission	Loss (nsec)			
20.32- × 25.40-cm slabs							
4.77	0.30	0.988	6.43	6.85	0.272 ± 0.005	0.83	0.22 ± 0.03
7.64	9.85	0.995	7.50	9.06	0.096 ± 0.003	0.37	0.98 ± 0.01
17.77-cm diam cylinders							
6.68	0.31	1.007	6.37	6.87	0.216 ± 0.004	0.63	0.20 ± 0.02
8.26	1.92	0.995	6.75	7.38	0.153 ± 0.003	0.46	0.15 ± 0.02
27.93-cm-diam cylinders							
4.44	0.31	1.004	6.56	6.99	0.286 ± 0.004	0.88	0.23 ± 0.03
5.40	2.88	0.979	6.82	7.67	0.217 ± 0.004	0.67	0.17 ± 0.02
7.31	12.95	0.996	7.63	9.87	0.088 ± 0.003	0.36	0.08 ± 0.01
8.43	51.10	0.995	6.87	10.64	0.013 ± 0.003	0.11	(6 ± 3) × 10 ⁻⁴
38.09-cm-diam cylinders							
5.71	9.61	1.004	8.92	10.76	0.163 ± 0.003	0.55	0.12 ± 0.01
7.63	130.8	0.999	7.69	12.50	0.031 ± 0.011	---	(2.5 ± 1.6) × 10 ⁻³

^aSeventy batches of 400 source neutrons were calculated except for the system with thickness of 8.43 and 7.63 where 125 and 200 batches were used. Standard deviations on the multiplication factors were between 0.003 and 0.005.

^bStandard deviations were between 0.004 and 0.06.

^cProbability that a neutron born in one core will produce a neutron in the other core.

Fig. 1. Axial neutron production density for two-core delayed critical assemblies of 27.93-cm-diam uranium cylinders.

core equals that of a single isolated core and thus implies an equality between the subcritical reactivity of an isolated core and the coupling reactivity in the two-core assembly. This equality is a measure of the adequacy of the assumption of loose coupling.

Calculations were performed for 3 delayed critical assemblies of 2 slabs⁴ in which the large, adjacent, flat 20.32- by 25.40-cm faces were separated and parallel and the thin sides were coplanar, and 14 delayed critical assemblies of 2 coaxial cylinders⁵ of 17.77-, 27.93-, or 38.09-cm diam with their flat faces separated. In one assembly the separation was zero.⁶ The neutron multiplication factor and some other calculated properties are given in Table I for some assemblies.

A core was divided into ten equal-thickness horizontal layers and the neutron production density was calculated in each layer as shown in the figure for the assemblies of 27.93-cm-diam cylinders.

The calculated coupling reactivity⁷ and the subcritical reactivity⁷ of an individual core obtained from prompt-neutron decay measurements on isolated cores, given in the inset to the figure, agree very well for reactivities < 4\$, although for values as great as 13\$, the calculated values are only 4% more. For the larger reactivities, the

calculated coupling reactivities are only ~10% more than those from measurements on isolated cores.

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Nuclear Criticality Safety—Analysis and Measurements

Analysis of Reflected Oralloy Critical Experiments, S. C. Cohen (GGA)

A series of calculations was performed in an attempt to predict the multiplication of Oralloy critical assemblies ($\rho = 18.8 \text{ g/cm}^3$; 93.5% enriched in ^{235}U) surrounded by light reflectors. The clean critical experiments, performed several years ago at Los Alamos Scientific Laboratory,¹ featured Oralloy cores surrounded by four inches of beryllium, beryllium oxide, graphite, and H_2O . In every case, the results revealed a significant overprediction of the multiplication factor.

The criticality calculations were carried out using the DTF-IV code² in a P_1 , S_4 approximation. Multigroup cross sections were obtained in an 18-group format (13 groups above and 5 groups below 0.414 eV) using the GGC-3 cross-section preparation code.³ The results, including references to the cross-section data sources, are given in Table I.

The very large discrepancy in the prediction for the the BeO-reflected core, noted earlier by other authors,⁴ prompted a more sophisticated analysis of this assembly. The effects on the predicted multiplication factor of variations in the basic cross-section data, the treatment of cross-section data by the multigroup cross-section preparation code, the order of the quadratures for the angular flux and the scattering matrix, and the number of energy groups, were explored. Multigroup cross sections for much of this refined analysis were obtained using the recently developed GAF/GAR/GAND code system.⁵ The magnitude of the effect on the predicted multiplication factor for each of the variations in cross sections and methods is given in Table II. It is interesting, if not disappointing, to note that the net effect of the refined analysis does not resolve the large discrepancy presented in Table I.

The reason for the consistent overprediction of the multiplication factor for these assemblies is not clear. It is felt that the ^{235}U cross sections are unlikely candidates for the source of the discrepancy. This conjecture is based upon the accurate prediction of the critical mass for Assembly No. 1 of the Thermionic Critical Experiment,⁶ which is a bare Oralloy-polyethylene assembly ($\text{NH}/\text{N}^{235}\text{U} \approx 0.8$) whose spectrum was closely tailored to that of a thick BeO-reflected core.

Barring experimental errors, then, either the treatment of the core-reflector interaction or the high-energy reflector cross sections are inaccurate. The refined analysis of the BeO-reflected assembly reduces the probability of the former explanation. Therefore, a careful reevaluation of the high-energy cross sections of these light reflector materials may be warranted.

TABLE I
Results of Analysis for Reflected Oralloy Assemblies

Reflector (4-in. thick)	Reference for the Reflector Cross Sections	Critical Mass (kg)	Critical Radius (cm)	Calculated k_{eff}
Beryllium (QMV)	ENDF/B	14.1	5.76	1.029
Beryllium Oxide	Beryllium (ENDF/B) Oxygen (Ref. 7)	15.5	5.95	1.059
Graphite (CS-312)	ENDF/B	24.2	6.90	1.026
H_2O	Hydrogen (Ref. 8) Oxygen (ENDF/B)	22.9	6.76	1.024

The sources for the core uranium cross sections have been given.^{8,10}

TABLE II
Results of Refined Analysis of Beryllium-Oxide/
Reflected Oralloy Assembly

Variation in Cross Sections and Methods	Effect on Predicted Multiplication Factor ($\delta k/k$)
Use ENDF/B ^{235}U Cross Sections	+0.002
Use ENDF/B Oxygen Cross Sections (infinite dilution)	-0.004
Self-Shield Oxygen and Beryllium Cross Sections	-0.003
Use P_3 Scattering Matrices	+0.009
Use S_8 Angular Flux Representation	-0.009
Use 27 Fast Groups (0.5 lethargy widths to 61 eV)	+0.004

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Critical Parameters of Plutonium Systems, L. E. Hansen, E. D. Clayton (BNW)

A number of clean critical experiments have been performed in recent years on homogeneous plutonium systems.¹⁻⁵ These experiments show that much of the criticality data that have been used as a guide for criticality safety,⁶ which were based on early experiments and calculations, should be revised. Particularly, the analysis of recent data shows that the values given for critical dimensions and masses can be generally in-

creased in the intermediate neutron energy range ($1 < H/Pu < 250$). These more recent data likewise provide a basis for extending the existing criticality data to higher ^{240}Pu contents and other moderation ratios.

To correct, and extend, the existing widely used criticality data, critical parameters were computed, using calculational techniques confirmed by experiments, for unreflected and water reflected spheres, infinite length cylinders, and semi-infinite slabs. The calculations were made using the HFN multigroup diffusion theory code and the DTF-IV multigroup transport theory code, both utilizing multigroup constants obtained from the GAMTEC-II code. These calculations, in conjunction with the available criticality data, were used to derive best estimates of the critical parameters of Pu(metal)-water mixtures over the entire range of moderation ratios. The mixtures contain up to 20 isotopic percent ^{240}Pu .

The validity of the calculations for ^{240}Pu contents of $> 5\%$, in well-moderated systems, was examined by computing eigenvalues for the measured critical Pu(NO₃)₄ solution systems reported in England¹ for cylinders containing 13.51 wt% ^{240}Pu , and for slabs containing 4.6 and 23.2 wt% ^{240}Pu , reported in the United States.³ The results of these calculations, when compared with results obtained by Richey,⁷ show that changing the ^{240}Pu content does not change the accuracy of the calculations.

Criticality data for undermoderated systems have been reported at $H/Pu = 15$ (2.2 and 8.08 wt% ^{240}Pu) by Richey et al.,⁴ and at $H/Pu = 5$ (11.46 wt% ^{240}Pu) by Bierman et al.⁵ The calculational agreement obtained was reasonably good for these experimental data, and was improved by correcting the experimental data for the size of the PuO₂ particles ($\leq 150 \mu$). The particle size corrections indicate that if, when setting criticality safety specifications, allowances are made for the ^{240}Pu content, then consideration should also be given to the self-shielding of the plutonium-bearing particle, since the critical mass will decrease when the fuel is lumped.

Fig. 1. Water reflected spherical critical radii of Pu(metal)-water mixtures.

Figure 1 shows the best estimates of the water-reflected spherical critical radii of Pu(metal)-water mixtures, and compares them to values⁹ taken from TID-7028. It can be seen that for systems containing 5 wt% ^{240}Pu , or more, one size vessel can be made critical at three different plutonium concentrations.

The effect of ^{240}Pu was examined as a function of geometry also, and it was found that semi-infinite slabs are the most sensitive to changes in ^{240}Pu content. Minimum critical masses were also computed, and were found to vary exponentially with the isotopic ^{240}Pu content.

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Monte Carlo Studies of the Use of Borosilicate-Glass Raschig Rings as a Fixed Neutron Absorber in Solutions of Fissile Material, J. Wallace Webster (ORNL)

The use of fixed neutron absorbers for criticality control dates back to 1958. In 1965 a proposed standard having the above title¹ was prepared by Work Group 3 of Subcommittee ANS-8 of the Standards Committee of ANS, and it was circulated for comments.

Since 1965, the Monte Carlo codes, of which the 05R code is the most generalized in its potential applications, have become fully operational and well-tested methods in criticality calculations. This paper describes various 05R studies relevant to Tables I-IV of Ref. 1.

To ensure that the total computing time for the project would be reasonable, all cases were computed with the glass in the form of regular arrays of parallel tubes, the glass volume fraction being the same as for the random arrays of Raschig rings. The first task was thus to determine a factor to apply to k_{∞} to correct for the random orientation. Table I presents these results.

For the glass fraction range, 0.24 to 0.30 (the range of interest for the standard), no significant effect on the correction factor was calculated for change in tube (ring) diameter.

Since the random arrays of Raschig rings should have a slightly smaller correction factor than the mutually perpendicular arrays actually calculated, a figure of 5% in k_{∞} was chosen as the universal correction factor, this figure being slightly smaller than the last two entries of Table I above.

Figures 1 and 2 present the results of the computations.

Fig. 1. k_{∞} vs g U/liter for $^{235}\text{UO}_2\text{F}_2\text{-H}_2\text{O}$ solution.

Fig. 2. k_{eff} for vessels of various shape, size, and reflector condition, containing a $^{235}\text{UO}_2\text{F}_2\text{-H}_2\text{O}$ solution with 1000 g ^{235}U /liter concentration and 30 vol% borosilicate-glass Raschig rings.

TABLE I

Change in k_{∞} Due to Substitution of Mutually Perpendicular Raschig Rings for Parallel Pipes

Vol% Glass	Percent Decrease in k_{∞} for Pipes to Rings
13	1.3
18.7	4.7
22.8	5.3
30.0	6.0

1. "Use of Borosilicate-Glass Raschig Rings as a Fixed Neutron Absorber in Solutions of Fissile Material," *Nucl. Eng. Bull.*, Vol. 3-3 (Nov. 1965).

Criticality Hazards Control for ZPPR Fuel, P. I. Amundson, R. G. Matlock, R. J. Forrester, J. C. Young, J. M. Gasidlo (ANL-Idaho)

The storage and use of the three metric tons of plutonium fuel used in the Zero Power Plutonium Reactor (ZPPR) has posed unique problems in the history of zero-power, fast-critical-facility operation. Similar problems may be encountered in the future, however, with the several tons of metallic or ceramic elements that will fuel individual, large, fast power reactors.

The basic ZPPR fuel consists of rectangular plates of metallic Pu-U-Mo alloy canned in stainless steel. The alloy contains 28.2 wt% Pu, 69.3 wt% depleted U, and 2.5 wt% Mo. The plutonium contains 11.5 at.% ^{240}Pu . Approximately 3000 kg of plutonium are available in this form. This quantity is contained in ~18 000 plates.

The problems arising in criticality hazards control are complicated uniquely by the considerable fast-neutron hazard associated with the spontaneous fission in the ^{240}Pu present, and by the gamma-ray hazard associated with both the unirradiated fuel and the additional gamma-ray activity after use in the critical facility. These neutron and gamma-ray hazards necessitated the use of storage facilities which provided considerable shielding, and this eliminated the use of conventional, air-spaced, birdcage arrays of storage racks, since dose rates of the order of several hundred millirem per hour would be obtained in the vicinity of such arrays.

The storage chosen consists of cast concrete modules, each with outside dimensions ~76 x 32 in. x 27 ft, with a high water content and distributed boron frit in the concrete. Twenty-four storage positions are provided in each module and a maximum of 6.2 kg of plutonium is permitted per module. Calculations of the multiplication in a closed-packed, infinite array of such modules were made using the SNARG transport theory code in one-dimensional cylindrical geometry with the S8 transport approximation. The Russian Bondarenko fast reactor cross-section set was used since it contained complete energy coverage for 10 MeV to thermal and permitted anisotropic treatment of scattering in hydrogen. In a close-packed array, the calculated k_{eff} was 0.33, and in an open array which corresponded to a practical module spacing, this reduced to 0.22. In view of the uncertainty in the calculations and the comparatively high calculated k_{eff} , careful multiplication measurements were made during storage of fuel in the vault. Because of the high

neutron source even in the absence of any multiplication, an experimental assessment of the source background was first made by using a small quantity of fuel which was placed in turn in various locations in the vault. The measurements confirmed the anticipated safety of the storage array.

Further problems are associated with criticality hazards control during movement of the fuel from storage, inspection, loading into drawers with other materials for use in ZPPR, transport to the split-table critical assembly machine, and loading. Many potential criticality hazards are present because of the large quantity of fuel present, the large number of individual fuel pieces present, and the frequent rearrangement and movement of fuel that is an essential feature of the physics studies. These problems are solved by the vigorous application of the usual techniques of administrative control which limit the quantity of fuel present at every stage in operations.

Although the logistic problems involved with ~18 000 fuel pieces and 600 000 pieces of depleted uranium, canned sodium, iron oxide, uranium oxide, etc., are significant, extension of many years experience with the smaller facility Zero Power Reactor 3 (ZPR-3) indicates that safe and efficient operation of ZPPR can be achieved.

Effect of Unit Cell Shape Variation in a Critical Array of Plutonium Metal Parts, H. F. Finn, G. A. Pierce, N. L. Pruvost, G. V. Pefley, Jr. (UCLRL-Livermore)

Six measurements have been performed at the Lawrence Radiation Laboratory to investigate systematic geometry changes in a critical, regularly spaced $4 \times 4 \times 4$ array of plutonium parts. Contrary to expectation, these measurements show that the smallest unit cell volume occurs for a nearly cubical array.

The basic plutonium part is a 3.03-kg mass of α -phase plutonium (93.6 wt% ^{239}Pu) in the shape of a right circular cylinder, 4.63-cm high $\times$ 6.53-cm diam. Plutonium density is 19.54 g/cm^3 . The array part contains 6.05 kg of plutonium, and is constructed by placing two 3.03-kg cylinders end to end to form a cylinder with nearly twice the height-to-diameter ratio of a single part (see Fig. 1). The vertical dimension of the array was first stretched and then compressed from a nearly cubical configuration. The critical state was achieved at each vertical spacing by making the appropriate changes in horizontal spacings. The final measurement, a $4 \times 4 \times 1$ planar array, was constructed as the limiting case of a large vertical separation. The experimental apparatus and techniques used for these measurements have been reported.¹ The experimental results are given in Table I.

TABLE I
Summary of Critical Parameters, $4 \times 4 \times 4$
6-kg Array Shape Variation

Vertical Center-to-Center Critical Spacing (cm)	Horizontal Center-to-Center Critical Spacing (cm)	Unit Cell Volume (liters)
13.12 $\pm$ 0.02	20.19 $\pm$ 0.02	5.349
17.12 $\pm$ 0.02	17.28 $\pm$ 0.01	5.111
22.12 $\pm$ 0.02	15.23 $\pm$ 0.01	5.131
32.13 $\pm$ 0.02	13.09 $\pm$ 0.01	5.505
47.12 $\pm$ 0.02	11.93 $\pm$ 0.01	6.707
∞^a	10.91 $\pm$ 0.02	∞

^a $4 \times 4 \times 1$ planar array.

Fig. 1. Array unit with support equipment.

One might have been tempted to guess that a deviation from a cubical array is a departure from optimum conditions that would demand a more compact array and, hence, a decrease in unit cell volume. However, the experimental results of Table I show that any such deviation in shape results in an immediate increase in unit cell volume. The limiting cases of array shape deviations are infinitely separated planar arrays or infinitely separated linear arrays which must, of course, have infinite unit cell volumes. Our unit cell volume increases are apparently steps in this direction.

Hence, safety calculations on systems that are significant departures from uniformly spaced systems should be treated with caution. A conservative approach would be to assume that the smallest spacing of a three-dimensional system is also the spacing of the other dimensions. Criticality data from uniform systems could then be applied with confidence.

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Critical Lattices of U(4.89) Rods in Water and in Aqueous Boron Solution, E. G. Johnson (UCC-Y12)

Earlier measurements¹ of critical water-moderated and -reflected lattices of 4.89% ^{235}U -enriched uranium [U(4.89)] rods 1.31, 2.07, and 2.49 cm in diameter, which provided reference for calculation of the criticality of heterogeneous assemblies of uranium of this enrichment, have been extended to rods 0.76 cm in diameter. The influence of the lattice pitch on the critical dimensions was determined by constructing several lattices in both square and triangular patterns. The data were then transformed, by equating bucklings using an extrapolation distance of 6 cm, to equivalent spheres for comparison with Clark's² calculations. The results are shown in Fig. 1 where critical masses are plotted as a function of the ^{235}U concentration. Shown for comparison are the earlier data for the 1.31-cm-diam rods. It is noted that the calculated masses of the 0.76-cm-diam rods are less than the experimental values at all concentrations.

Clark² predicted that the smallest critical mass of zero-diameter rods would contain 1.77 kg of ^{235}U at a concentration of 49.7 g/liter. This point was experimentally investigated with an aqueous $\text{U(4.98)O}_2\text{F}_2$ solution containing 45.3 g/liter enclosed in a sphere of 0.05-cm-thick stainless steel, 44 cm in diameter, based on Monte

Fig. 1. Comparison of calculated and experimental critical spherical lattices of U(4.89) rods in water.

Carlo calculations by Webster.³ In the experiment, the water-reflected critical volume, at a temperature of 24°C, filled the container to 99% of its capacity. Consideration of the reactivity of this void and of structural materials, evaluated by substitution experiments, shows that k_{eff} of the solution sphere was 1.006. The mass of ^{235}U in the sphere was 2.01 kg, to be compared with 1.78 kg from Clark's calculations for the concentration of the experiment.

To provide guidance for the dissolution of rods of this ^{235}U enrichment in chemical processes, a limited number of experiments was performed with an aqueous solution containing 0.140 g of boron per liter as the moderator and reflector. A lattice of rods of each of the above diameters was constructed in the boron solution at the pitch that had resulted in the minimum measured critical mass in the water-moderated and -reflected experiments. Typical results are compared in Table I with those from water-moderated and -reflected lattices.

TABLE I
Criticality of Rods Latticed in Water and in an Aqueous Boron Solution
Boron Concentration: 0.140 g/liter

Description of Lattice ^a				Critical Mass of ^{235}U (kg)	
Diameter (cm)	Pitch (cm)	Concentration (g/liter)	Lattice Height (cm)	Water	Boron Solution ^b
2.49	4.72	234.0	30	6.16	7.58
			60	8.42	10.02
2.07	4.45	180.6	30	5.15	6.87
			60	6.97	8.91
1.31	3.21	140.6	30	3.58	4.59
			60	4.89	6.02
0.76	2.20	103.8	30	2.53	3.30
			60	3.47	4.33

^aEach Lattice was in a triangular pattern.

^bThe ^{10}B content of the boron was 19.81 at. %.

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2. H. K. CLARK, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory (1966).
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Approach-to-Critical by Modal Analysis for Plutonium-Fueled Systems, Arne P. Olson, R. G. Palmer (ANL-Idaho)

The large built-in ^{240}Pu spontaneous fission source creates special problems during an approach-to-critical for zero-power plutonium-fueled fast reactors. Generally, the approach to critical is monitored by several detectors whose inverse count rates are empirically correlated in numerous and sometimes arbitrary ways with fissile mass or core size to predict the critical size.

This paper describes an analytical study of two-region reactors to provide a physical basis for selection of empirical correlations, and to suggest new correlations better able to predict the critical size from subcritical measurements. The analysis we have performed compares flux and k_{eff} calculated by one-group modal analysis with multigroup diffusion and transport theory results.

The one-group flux in a cylindrical reactor of height H with core radius a and a zero flux boundary b having a flat distributed source in the core can be expanded in modes as follows:

$$\phi_1(r, z) = \sum_{m=1}^{\infty} \sum_{n=1}^{\infty} A_{mn} \cos(\alpha_m z) J_0(\beta_n r), \quad 0 \leq r \leq a;$$

$$\phi_2(r, z) = \sum_{m=1}^{\infty} C_m \cos(\alpha_m z) \left[K_0(\gamma_m r) - \frac{K_0(\gamma_m b)}{I_0(\gamma_m b)} I_0(\gamma_m r) \right], \quad a \leq r \leq b,$$

$$\text{where } \alpha_m = \frac{m\pi}{H} \text{ (m odd)}, \gamma_m^2 = \alpha_m^2 + \kappa^2.$$

The expansion coefficients A_{mn} and C_m are

$$A_{mn} = \frac{(-1)^{\frac{m+1}{2}}}{(B^2 - \alpha_m^2 - \beta_n^2)} \cdot \frac{8 S J_1(\beta_n a)}{m \pi a D_1 \beta_n [J_0^2(\beta_n a) + J_1^2(\beta_n a)]};$$

$$C_m = \sum_{n=1}^{\infty} A_{mn} J_0(\beta_n a) / K_0(\beta_n a).$$

The core buckling is B^2 , with κ^2 the blanket Laplacian. D_1 and D_2 are core and blanket diffusion coefficients. The β_n are roots of

$$\beta_n \frac{D_1 J_1(\beta_n a)}{D_2 J_0(\beta_n a)} - \frac{\gamma_m \left[K_1(\gamma_m a) + \frac{K_0(\gamma_m b)}{I_0(\gamma_m b)} I_1(\gamma_m a) \right]}{K_0(\gamma_m a) - \frac{K_0(\gamma_m b)}{I_0(\gamma_m b)} I_0(\gamma_m a)} = 0.$$

k_{eff} is given by the relation

$$k_{eff} \approx 1 - \frac{(B_g^2 - B^2) L^2}{1 + L^2 B_g^2},$$

where B_g^2 is the geometrical buckling and L the diffusion length.

The 22-group calculations were performed using ANL Cross-Section Set 224 in the diffusion and transport theory codes MACH 1¹ and SNARG² to give reference flux shapes and reactivities for a range of core sizes with k_{eff} as small as 0.4, and to give group-collapsed parameters for one-group modal analyses. Two large (~450-liter spherical core) fast-reactor compositions were chosen to represent hard and soft spectra. Geometries studied included cylinders with axial extrapolated lengths of 100, 120, and 140 cm, plus axially infinite system. Excellent agreement was obtained with reactivities computed by the one-group theory approximation.

Figure 1 illustrates typical soft-spectrum results. Hard-spectrum results are essentially the same. The correlation of $R^{5/2}$ /count rate vs R is linear to very small values of k_{eff} . Part of the nonlinearity at small core

Fig. 1. Correlation of approach-to-critical.

radii may be due to truncating the modal expansions after 15 radial modes for each of 15 axial modes. Studies of this possibility are continuing. Figure 1 also shows results obtained³ for Assembly 47. Again a nearly linear curve results. Work is in progress to apply the same correlation to the analysis of various loadings of ZPR-3.

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Neutron Age and Infinite-Medium Neutron Multiplication Factor for UF₄-Paraffin Mixtures with Low ²³⁵U Enrichment, S. J. Raffety, (ORNL), J. T. Mihalcz (UCC-Y12)

The infinite-medium neutron multiplication factor k_{∞} and the neutron age to thermal energy for UF₄-paraffin mixtures have been inferred from critical experiments with 2 and 3% ²³⁵U-enriched uranium. The volume fraction of paraffin was varied so that the H:²³⁵U atomic ratio was between 195 and 972 for the 2% enrichment and was 133 and 277 for the 3% enrichment.

The infinite-medium multiplication factor is given as

$$k_{\infty} = [(1 - \beta) K_p(B) + (\beta/\epsilon) K_d(B)]^{-1} \quad (1)$$

where $K_p(B)$ and $K_d(B)$, the nonleakage probabilities for prompt and delayed neutrons, respectively, are given by

$$K_d(p)(B) = \{(1 + \tau_d p) B^2 (1 + L^2 B^2)\}^{-1} \quad (2)$$

The change in reactivity with a change in buckling is

$$\frac{\Delta \rho}{\Delta B^2} \approx \frac{\epsilon(1 - \beta)(1 + \tau_d B^2)(L^2 + \tau_p + 2L^2 \tau_p B^2)}{(1 + \tau_p B^2)^2 (1 + L^2 B^2)} + \frac{\beta(L^2 + \tau_d + 2L^2 \tau_d B^2)}{(1 + \tau_d B^2)(1 + L^2 B^2)} \quad (3)$$

The fast fission factor ϵ , the material buckling B^2 , and the quantity $\Delta \rho / \Delta B^2$ were measured in critical assemblies of each fuel mixture. The thermal-neutron diffusion area L^2 was calculated from thermal-neutron cross sections.⁴ The delayed-neutron fraction β was determined from the delayed-neutron fractions for ²³⁵U and ²³⁸U measured by Keepin⁵ and the ²³⁸U-to-²³⁵U fission ratio measured in the critical assemblies.

The age to thermal energy of delayed neutrons τ_d was obtained from the second moment of the flux distribution in 199-cm-radius spheres calculated by S_{32} transport theory using the ANISN³ code, the energy distribution of delayed neutrons as the source spectrum, and a modification by Knight⁴ of the Hansen-Roach⁵ 16-group cross sections. This modification is a reduction of the ²³⁸U absorption cross sections in the resonance region such that agreement is obtained between calculations of k_{∞} and experimental results from the Physical Constants Testing Reactor for low-enrichment materials. The sphere size, the order of S_N , and the convergence criteria used in the calculations were determined by comparing the calculated age of prompt fission neutrons in water to the average of four experimental values given before.⁶

With all the other parameters obtained as described above, Eq. (3) can then be solved for the age of prompt fission neutrons to thermal energy τ_p . The nonleakage probabilities $K_p(B)$ and $K_d(B)$ are then calculated by Eq. (2) and these quantities are used in Eq. (1) to obtain k_{∞} . The results, along with the parameters used in the calculation, are given in Table I.

The age of prompt neutrons to thermal energy, calculated by the same method as that for the delayed neutrons, and values of k_{∞} obtained from transport theory are shown in Table I. The values of k_{∞} agree very well with experiment.

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TABLE I
Neutron Age and Infinite-Medium Neutron Multiplication Factor for
UF₄-Paraffin Mixtures with 2 and 3% ²³⁵U Enrichment

	2% ²³⁵ U						3% ²³⁵ U	
²³⁵ U atomic density (10 ²⁰ atoms/cm ³)	1.581	1.330	1.119	0.992	0.867	0.623	2.349	1.671
H: ²³⁵ U atomic ratio	195 ± 3	294 ± 5	406 ± 7	496 ± 8	614 ± 10	972 ± 17	133 ± 2	277 ± 5
β, delayed fraction of fission neutrons (10 ⁻³)	7.06 ± 0.31	6.96 ± 0.29	6.89 ± 0.28	6.85 ± 0.27	6.81 ± 0.26	6.72 ± 0.24	7.10 ± 0.31	6.88 ± 0.28
ε, fast fission factor, measured	1.042 ± 0.007	1.034 ± 0.006	1.029 ± 0.005	1.026 ± 0.004	1.022 ± 0.004	1.016 ± 0.002	1.045 ± 0.008	1.028 ± 0.005
L ² , thermal-neutron diffusion area, calculated (cm ²)	1.69 ± 0.08	1.67 ± 0.08	1.72 ± 0.09	1.79 ± 0.09	1.86 ± 0.09	2.11 ± 0.11	1.15 ± 0.06	1.22 ± 0.06
B ² , average critical buckling, measured (10 ⁻³ cm ⁻²)	4.393 ± 0.059	5.892 ± 0.118	6.568 ± 0.122	6.651 ± 0.116	6.205 ± 0.073	3.616 ± 0.030	6.561 ± 0.051	10.209 ± 0.122
$\frac{\Delta\rho}{\Delta B^2}$, change in reactivity per unit change in buckling, measured (cm ²)	-36.80 ± 0.52		-26.69 ± 0.37	-24.85 ± 0.35	-23.65 ± 0.33	-24.05 ± 0.34	-32.89 ± 0.46	-23.07 ± 0.32
Multiplier for Hansen-Roach ²³⁵ U absorption cross sections for groups 8 through 12	0.74	0.78	0.83	0.87	0.93	1.00	0.74	0.83
τ _d , average age of delayed neutrons to thermal energy, calculated (cm ²)	27.0	21.5	18.6	17.3	16.0	14.5	25.7	17.5
τ _p , age of prompt fission neutrons to thermal energy (cm ²)								
Measured	42.6 ± 3.2		31.8 ± 2.5	28.8 ± 2.0	26.4 ± 1.8	24.4 ± 1.6	42.8 ± 4.0	32.0 ± 3.1
Calculated	45.5	37.9	33.9	32.0	30.1	27.5	44.1	32.6
k _∞ , infinite-medium neutron multiplication factor								
Measured	1.196 ± 0.014		1.222 ± 0.018	1.205 ± 0.014	1.177 ± 0.011	1.096 ± 0.006	1.290 ± 0.027	1.343 ± 0.032
Calculated	1.204	1.229	1.217	1.202	1.170	1.086	1.302	1.341

Reviews of Recent Criticality Experiments and Plans for Obtaining Nuclear Criticality Safety

Recent Experimental Criticality Safety Data Obtained in France—Correlation with Calculations—Planned Experimental Program, P. R. Lécorché (CEA-France)

Nuclear criticality safety data have been obtained from experiments performed at the "station de Criticité" at VALDUC.

The effect of ^{240}Pu on criticality has been studied using plutonium solutions ranging in concentration from 25 up to 355 g Pu/liter. Five different variations in ^{240}Pu content were used: 1.5, 3.11, 4.23, 6.93, and 9.95 wt%; some of the results have been reported.¹

Heterogeneous poisoning of plutonium nitrate solutions, using either Pyrex glass pipes in a lattice or Pyrex glass Raschig rings at random, has also been studied. The minimum critical concentration at 2.3 acid molarity was found to be 183 g/liter ± 10 using Raschig rings of dimensions 3-cm o.d., 2.6-cm i.d., and 3 cm in height. The B_2O_3 content was 13% and the volume fraction occupied by the Raschig rings was 16.2%. The ^{240}Pu content was 4.23 wt%.

A program has been started on interaction studies, utilizing either 90 wt% enriched uranyl nitrate solutions or plutonium nitrate solutions.

Isolating properties of several materials have been determined by placing a wall made of the material between two vessels and comparing the critical height of one vessel reflected by the wall with the common critical height of the two vessels interacting through the wall. The degree of isolation is determined from the ratio $P = (H_{ci}/H_{cr})$, which approaches unity when the wall's thickness provides effective isolation. Allowing for some interaction through reflection by the floor, it has been found that isolation was accomplished ($P = 0.998$) for thicknesses of 20 cm of parafin wax, 17.5 cm of parafin wax plated with 0.85-mm cadmium on each side, 18 cm of borated permali, and 20 cm of borated concrete. With 30 cm of ordinary concrete, P was only 0.993.

Experiments have been performed using mixtures of liquid 93 wt% enriched UF_6 and HF in fully reflected spherical geometry. Spheres were made of Monel with a 4-mm wall thickness. For a temperature of 75°C , the minimum critical volume was 66 liters at a H/U ratio of 32, corresponding to a mass of 19 kg of uranium. S_{∞} calculations, using the Hansen and Roach 16-group cross sections, were found to be in good agreement with the experiments. Critical masses were then computed for UO_2F_2 aqueous solutions in the undermoderated range. Results are: $\text{H/U} = 20-8$ kg, $\text{H/U} = 10-20$ kg, $\text{H/U} = 5-41$ kg, $\text{H/U} = 2-74$ kg.

Some preliminary experiments in the scope of CRAC program (see below) have been performed with a solution of 90 wt% enriched uranyl nitrate at a concentration of 100 g/liter in a 29.6-cm-i.d. stainless-steel bare cylinder. Excess reactivities of 235×10^{-5} , 345×10^{-5} , and 420×10^{-5} above delayed criticality were introduced at a rate of $\sim 10 \times 10^{-5} \text{ sec}^{-1}$. The free evolution of the reactor was recorded over intervals up to 8 h. In the last

experiment which lasted 7 h, 10 min. 7.38×10^{16} fissions occurred (radiochemical determination) and 4 peaks were observed. Maximum fission rate of the first peak was between 6.10^{14} and 10^{15} fissions/sec with a period of 5.3 sec. Calculations made with RTS and RTR codes appear to be in good agreement with experiments.

Planned experimental programs include, in addition to further interaction studies, experiments on low enriched (4 wt%) uranium solutions, but the major program is directed toward an experimental study of criticality accidents with fissile solutions. This program, called CRAC (Criticality Accidents or Radiological Consequences of Criticality Accident) will begin with the study of 93 wt% enriched uranyl nitrate solutions in vessels ranging from 30-cm i.d. to 80-cm i.d. Rates of solution addition shall range between 150 l/h and 1500 l/h. Besides the studies of the neutronic behavior, radiological and biological studies will be conducted by specialized departments. These studies will include calibration of criticality dosimeters, testing of criticality alarm systems, irradiation of phantoms, and probably, in a later stage, irradiation of animals.

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The Development of Criticality Data for Nuclear Safety Purposes, J. H. Chalmers (UKAEA-Winfrith)

It has been established practice to base criticality safety guidance on experimentally derived data wherever possible. The growing complexity of many processing plants and even individual items of equipment, however, gives rise to a situation where direct reference to experiment is not always possible. Unless experiments are to be designed to match this complexity, and this would be extremely costly even if practicable, it follows that an increased reliance upon calculational methods becomes inevitable. Methods of calculation are available that are capable of true representation of plant geometry and arrangement, and there is distinct advantage in this approach, if they can, with confidence, be applied directly to the solution of nuclear safety evaluation problems.

This, in fact, is the approach favored in the UKAEA influenced by the development in the GEM program in recent years. Although development for this code is now at an end, effort along similar lines is being continued in the formulation of the program MONK, which more or less builds in the foundation of GEM. It will have a more powerful geometry capability and has the further advantage that multigroup constants such as are associated with diffusion and transport calculations will be available routinely. The flexibility of the Hole routine concept will be increased and much that now requires this treatment will be possible with standard geometries. The potential of this type calculation can be exploited only if there is a sufficient experimental program oriented toward prov-

Fig. 1. Unreflected minimum critical masses for PuO_2/U (NAT) $\text{O}_2/\text{H}_2\text{O}$ mixtures.

ing the adequacy of the nuclear data used in them. This paper gives some examples of checks made to this end on recent, and on some not so recent, experiments.

There have been few reported experiments on mixed plutonium and uranium systems, but data were obtained for $\text{PuO}_2/\text{UO}_2/\text{CH}_2$ polythene reflected cores¹ and for 6 wt% PuO_2/UO_2 fuel rods² in water. In the former case, cores were constructed from compacts having H/Pu and Pu/U atomic ratios of 18.6 and 0.335, respectively. Each compact measured $4.29 \text{ cm}^2 \times 2.157 \text{ cm}$ and contained 170.5 g total oxide. The calculated k_{eff} for a stack $30 \text{ cm}^2 \times 24.4\text{-cm}$ high containing 94 kg oxide was 1.0591 (this represents ~95% of the measured mass) using one set of ^{239}Pu data and 1.022 using another. With the second data set, the calculated k_{eff} for a $34.3\text{-cm}^2 \times 20.74\text{-cm}$ high core containing 105 kg oxide was 1.0282.

Comparison with the measured square pitched lattices of 0.3374-in.-diam rods is given in Table I.

TABLE I

Rod Pitch (in.)	Critical No. Rods	Critical H ₂ O Height (in.)	Calculated K_{eff}
0.56	19 × 19	31.8	1.048 ^(a)
0.56 ^(b)	21 × 21	34.669	1.025
0.735	13 × 13	26.933	1.0106
0.792	12 × 12	30.22	1.007
1.04	11 × 11	31.299	1.0015

(a) For $K_{\text{eff}} = 1$ calculated critical $\text{H}_2\text{O} = 31.0$ in.

(b) 337 ppm boron in H_2O .

Previously reported work³ on the water-reflected critical parameters of $\text{PuO}_2/\text{UO}_2/\text{H}_2\text{O}$ systems has been extended to include the unreflected condition and the results are shown in Fig. 1.

The paper also includes some further checks on individual uranium and plutonium systems experiments and investigates the limiting concentrations for $k_{\infty} = 1$ for mixed fuels and for variations of ^{240}Pu content in plutonium systems.

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Criticality Research at Pacific Northwest Laboratory, S. R. Bierman, E. D. Clayton (BNW)

The research programs at the Pacific Northwest Laboratory's Critical Mass Laboratory have been primarily concerned with the criticality of plutonium and its various isotopes. However, experiments and calculational studies are performed on uranium-fueled systems, and experiments are now planned which involve ^{238}Pu - ^{239}Pu mixtures and plutonium-uranium mixtures. Also under consideration is the criticality evaluation of ^{238}Pu and

some of the more promising lesser-known actinide elements such as neptunium, curium, and americium. An integral part of each of the experimental studies is the development of reliable calculational techniques for use in plant criticality safety—thus reducing the amount of experimental data needed.

The plutonium criticality studies are currently centered around two types of experiments. One is concerned with the criticality of plutonium compounds in the undermoderated range. The other is concerned with slightly undermoderated to overmoderated plutonium solutions. The objective of these programs is to provide, over the full range of neutron moderation, well-defined experimental data points that can be used both in establishing plutonium criticality safety limits and as a series of benchmarks against which nuclear constants and calculational techniques can be compared. The primary parameter of interest in these experiments is the critical mass of both bare and reflected systems and the effect that such variables as geometry, moderation, ^{240}Pu isotopic concentration, nitrate concentration, plutonium density, and neutron poisons have on criticality. However, neutron kinetic measurements in the form of prompt-neutron decay constants and neutron spectra measurements in the form of reaction rates are also made.

The experiments with undermoderated plutonium have an H/Pu atomic range typical of that found in the wet powders, precipitates, slurries, and polymers encountered in the reprocessing of irradiated reactor fuels. Consequently, these experiments are carried out on a remote split-table machine with plutonium oxide in the form of solid cubes. Experiments are currently planned with homogeneous mixture of PuO_2 -polystyrene having H/Pu atomic ratios from about 1 to around 55 and ^{240}Pu isotopic concentrations up to about 25%. To date, experiments have been completed on mixtures having H/Pu atomic ratios of 15 (2.2 and 8.08 wt% ^{240}Pu) and 5 (11.46 wt% ^{240}Pu). Experiments are now planned with PuO_2 at an H/Pu ratio of about 1 and a ^{240}Pu isotopic concentration of about 18 wt%.

The experiments covering the slightly undermoderated to overmoderated range of the neutron spectrum are performed with plutonium nitrate solutions having H/Pu atomic ratios ranging from about 1100 down to 55. These experiments are carried out in a specially designed system in which solution is transferred, by remote control, to an experimental vessel which is itself interchangeable to permit studying the effects of geometry on criticality. The experimental vessel in which measurements are currently being made is an expandable slab tank, 42 in. on a side having a thickness range of from 3 to 9 in. Previously, measurements were performed on thin stainless-steel spheres, 11.5-, 14.1-, and 15.2-in. diam, as the experimental vessels. Experiments are planned with dilute plutonium nitrate solutions in large cylindrical geometry in conjunction with a series of measurements involving fixed and soluble neutron poisons.

The results from the plutonium experiments to date indicate the values for critical sizes and masses (given in references such as TID-7028) can be increased in the intermediate neutron energy range ($1 < \text{H/Pu} < 250$). Also, calculations, confirmed by experiments, indicate that consideration should be given to the fuel particle size in mixtures if proper allowances are to be made for the effect of ^{240}Pu on criticality.

For complex geometries and interacting arrays, the GEM¹ code is being used in establishing criticality safety limits and in correlating experiments with theory. However, the calculational technique being developed concurrently with the experiments on easily defined clean geometries uses the computer code GAMTEC-II² to generate multigroup physics constants for use in either the diffusion theory code HFN³ or the transport theory code DTF-IV.⁴ Using Sher-Felberbaum⁵ normalized ^{239}Pu cross sections, these calculational models can reproduce, with one or two exceptions, the critical assemblies over the full range of neutron moderation, nitrate concentration, and ^{240}Pu isotopic concentration covered to date. Calculated k_{eff} values for some of the critical assemblies are shown as being typical (Table I).

TABLE I

Pu(NO ₃) ₄ Solutions - 4.6 wt% ²⁴⁰ Pu					
Critical Assembly	Reflector	Pu, g/liter	Calculated k _{eff}		
			Diffusion Theory	Transport Theory	
14-in. sphere	water	33.0	0.999	1.015	
11.5-in. sphere	water	268.7	1.001	1.018	
15.2-in. sphere	water	24.4	1.000	1.014	
15.2-in. sphere	48-mil steel	39.0	0.997	1.006	
42- × 38.63- × 6.20-in. slab	none)	58.0	0.949	--	
42- × 39.91- × 4.40-in. slab	water	58.0	1.012	--	
PuO ₂ -Polystyrene					
Critical Assembly	Reflector	²⁴⁰ Pu, wt%	Pu g/cm ³	Calculated k _{eff}	
				Diffusion Theory	Transport Theory
35.84- × 35.84- × 28.82 cm	none	11.46	2.302	--	0.997
5.88-cm infinite slab	Plexiglas	11.46	2.302	--	0.966
7.38-cm infinite slab	Plexiglas	8.08	1.05	0.982	0.979
5.99-cm infinite slab	Plexiglas	2.2	1.12	0.993	0.991

These same calculational models reproduce pulsed-neutron and reactor noise measured values of β /liter for the thermal systems; however, the calculated values for the undermoderated systems are about 40% higher than measurements.

Comparisons between calculations and experiments will be covered in greater detail in the paper. Also, additional results from recent experiments with plutonium solutions in slab geometry will be presented.

The critical experiments planned with ^{238}Pu - ^{239}Pu fuel mixtures, although limited to well-moderated systems, will provide data needed for establishing criticality safety limits in the production of ^{238}Pu and for checking computational methods and cross-section sets. Survey-type calculations are in progress on the criticality of Pu-U mixtures typical of the LMFBR fuels and on those actinide elements which may be produced in large quantities in the not-so-distant future. The results of these calculations will serve as a guide in planning critical experiments with these materials and for planning purposes in their processing until sufficient experimental data become available.

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Fast Reactor Lattice Experiments and Analysis

A Benchmark Study of ZPR-3 Assembly-48 Using ENDF/B Cross Sections, R. Archibald, A. Baxter, D. Sargis (GGA)

A benchmark analysis of ZPR-3 Assembly-48 has been performed using ENDF/B data and the GAF/GAR/GAND series of fast reactor codes.¹ Only minor alterations of the ENDF/B data were made to conform with the formats required in the cross-section generating programs. The calculated reactivity worths and other integral quantities should therefore be useful, in conjunction with experimental results, in identifying possible cross-section discrepancies in the present microscopic ENDF/B data.

A number of reactivity worths were also computed using the previous GAM² cross-section data in use at Gulf General Atomic for comparison with the ENDF/B results.

Detailed neutronics calculations were performed in spherical geometry using the transport code DTF-IV³ in the P₁-S₄ approximation with 27 broad energy groups. A space-independent perturbation cross section was also computed at the center of the system for comparison with measured results. The first-order transport perturbation theory program GAPER⁴ was used in evaluating central material worths as well as Doppler and sodium-void reactivity coefficients.

In Table I, a comparison of calculated and measured central reactivity worths is presented in units of lh/kg using the conversion factor $1\% \Delta k/k = 949.67$ lh for $\beta_{eff} = 0.00358$. The calculated reactivity worths using the GAM data tapes for Fe, Cr, Ni, Al, and C are given in the parentheses in Table I. Experimental and calculated central fission and capture ratios for various materials relative to the ²³⁵U fission rate are also presented in Table I.

In general, the calculated ENDF/B and measured reactivity worths, as well as the neutron spectra, compare reasonably well. However, central worths for several ENDF/B materials exhibit substantial deviation from measured results and a reevaluation of the cross-section data may be justified. Recently, detailed comparisons of calculated ENDF/B and GAM data with measured results in the Thermionic Critical Experiment⁵ corroborate many of the large discrepancies found in the present study.

A number of conclusions regarding the status of the ENDF/B data for Mo, O, Cn, Na, and ²³⁸U may be made. The ENDF/B molybdenum infinite dilution cross sections are 70 to 90% larger than the older GAM infinite dilution data and reactivity worths are in even poorer agreement with experiment than the older cross sections. A set of unresolved resonance parameters for Mo should be added to the ENDF/B data so that more meaningful resonance cross sections may be obtained. The ENDF/B oxygen cross sections are in slightly better agreement with experiment than the older data, but still deviate from measured reactivity worths by more than 25%. The new chromium data are a considerable improvement over the old cross-section set, although very large discrepancies, of the order of 60% with regard to experiment, still exist. The calculated sodium-void and central reactivity worths deviate by a factor of 4 from measured results and warrant possible modification of the data. The calculated

capture-to-fission ratio α for ²³⁸U is 10 to 15% higher than the measured result and may account for the calculated discrepancies in the ZPR critical mass and ²³⁸U Doppler calculations.

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TABLE I
Comparison of Measured and Calculated Central Worths and Reaction Ratios

Material	Reactivity Worth (lh/kg)		Reaction Ratios		
	Experiment	Calculation	Ratio	Experiment	Calculation
²³⁵ U	339	379	σ_f^0/σ_f^0	0.976	0.936
²³⁸ U	- 24.8	- 34.3	σ_c^0/σ_f^0	0.243	0.232
²³⁹ Pu	445	507	σ_f^0/σ_f^0	---	1.294
²⁴⁰ Pu	81	79.4	σ_f^0/σ_f^0	---	---
¹⁰ B	-8926	-8427	σ_f^0/σ_f^0	1.480	1.427
Fe	- 12.3	- 14.9 (-13.9)	σ_f^0/σ_f^0	0.204	0.198
Cr	- 9.4	- 16.8 (-23.6)	σ_f^0/σ_f^0	0.067	0.0676
Ni	- 18.2	- 17.0 (-16.5)	σ_f^0/σ_f^0	0.0307	0.0294
Al	- 15.7	- 16.2 (-14.9)	σ_f^0/σ_f^0	0.138	0.148
Na	- 6.3	- 1.77	σ_c^0/σ_f^0	---	1.289
C	- 4.6	- 5.29 (-4.57)	σ_c^0/σ_f^0	---	---
O	- 6.8	- 9.20	σ_c^0/σ_f^0	---	---
Ta	- 164	- 168	σ_c^0/σ_f^0	---	---
Mo	- 43.4	- 93.6	σ_c^0/σ_f^0	---	---

FTR Critical Experiments Analysis, W. R. Young, R. A. Bennett (BNW)

ZPR-III Assemblies-51 and -52a through -52f were constructed by Argonne National Laboratory for Battelle Northwest as a part of the early effort in support of the design of the Fast Test Reactor (FTR).¹ Material compositions approximated those of the PuO₂-UO₂ driver zone and nickel-sodium reflectors of the reference design of the FTR.

Experiments carried out in Assembly-51 were devised to check computational models and to measure parameters related to the design and safe operation of the FTR. These included critical mass, spatial activation and small sample reactivity distributions, sodium voiding, spatial dependence of neutron spectra, central reactivity worth measurements, fuel compaction, and reactivity control worths.

The 52 series of assemblies was specified for investigation of the effect of fuel-free core zones, i.e., simulated test zones, on critical mass and peak-to-average flux. The ~80% NA-20% steel test zones were centered on the assembly axis and approximated either cylindrical holes or rectangular slots.

Analyses of these experiments were done with the diffusion, transport, and perturbation computer codes, FCC IV,² DTF V,³ 2DF,⁴ 2DB,⁵ and PERT IV.⁶ Space-energy-averaged group cross-section sets were used and were obtained from a highly modified 26-group Bondarenko set.⁷ Experiment-calculational comparisons are listed in Table I.

Critical mass calculations are in agreement with experiments within the Estimated Current Accuracy (ECA)⁸ of $\pm 6\%$. The masses are, however, with only one exception, consistently overestimated by ~6%. Therefore, the peak flux was correspondingly underestimated by ~6%. The exception indicated is for a system with large axial rectangular fuel-free zone through the core, where axial buckling averaging problems are tentatively blamed for this larger 10% discrepancy.

Sodium was progressively removed from a central zone in Assembly-51. The zone radii and heights were selected to allow an approximate identification of the zone over which the sodium worth is positive. Calculations of the worths of sodium removed are less negative or more positive than measured, as is shown in the table. While the discrepancies between calculations and experiments are large, these results and, hence, early conceptual design values for loss of sodium, are conservative.

Central perturbation worths of ^{239}Pu and ^{238}U are overcalculated by ~30% here, as has been the case in analysis of many earlier experiments. While no explanation of these discrepancies between perturbation calculations and experiments can be given yet, it is noted that significantly better agreement is obtained for larger core samples whose worths are primarily due to ^{239}Pu and ^{238}U .

Fuel compaction experiments and large-sample perturbation experiments were performed to support safety and fuel management studies and design activities. The correlations are mostly between 0.85 and 1.09 ($\pm 15\%$) compared with ECA values of $\pm 40\%$.

Fission-rate ratio correlations provide some measure of the adequacy of calculated neutron spectra and are, as in earlier experiments, very good.

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TABLE I
ZPR-III Assemblies-51 and -52-Results and Analysis

Experiment	Experimental ^a Results ^a	Calculational Results	e/c
ZPR-II Assembly-51			
Critical Mass, kg	212	225	0.941 ^a
Central Perturbation Worths, lb/kg			
^{239}Pu	+ 579.7	+ 761	0.761
^{238}U	- 23.1	- 35	0.65
^{235}U	- 9675.0	- 9645	1.003
Uniform Sodium Removal 11.54 kg over the core, Δk	- 202	- 192	1.05
Progressive Sodium Removal, Δk			
Voided Volume	Voided		
Height, in.	Volume, liter		
14	9.8	+ 9.1	+ 12
34	23.9	- 18.6	- 4
14	22.9	+ 10.4	+ 24
Axial Movement of Core in Central Column, Δk			
7-in. Slug Position	0-7 in.	- 92	- 92
	5-12 in.	- 206.8	- 235
	10-17 in.		
Fuel Compaction, Δk	+ 125.4	+ 146.2	0.86
U + Pu Worth in Central Column, Δk	- 752	- 863	0.87
Core Out - Na in Central Column, Δk	- 763	- 891	0.86
Central Rod Worths, Δk			
B,C	905	832	1.09
Ta	573	538	1.07
Peripheral Rod Worths, Δk			
B,C	213	213	1.00
Ta	120	147	0.82
Fission Ratios			
$^{239}\text{Pu}/^{238}\text{U}$	0.031	0.033	0.94
$^{239}\text{Pu}/^{235}\text{U}$	1.003	1.012	0.99
$^{238}\text{U}/^{235}\text{U}$	0.240	0.240	1.00
ZPR-III Assemblies-52			
Critical Mass, kg			
52a cylinder	199	213	0.934
52b 24-liter central hole	236	251	0.940
52c 66-liter central hole	280	292	0.958
52f 98-liter central hole	309	325	0.951
52d 88-liter rectangular hole	275	305	0.902
52e 119-liter slotted core	282	301	0.937

^aAll reactivity in Δk , 1% $\Delta k/k = 1036 \Delta k$.

Studies of Fast-Neutron Cross Sections Through Integral Experiments, J. M. Kallfelz, W. P. Poenitz, B. R. Sehgal, B. A. Zolotar (ANL)

Calculations of integral parameters in ten fast critical assemblies have been performed with several neutron cross-section sets to test the agreement with experimental integral values. These critical assemblies are listed in Table I in approximately decreasing order of spectrum hardness. They and their specifications for one-dimensional homogeneous calculations have been selected by Davey and Hess¹ for the next phase of ENDF/B data testing.

TABLE I
Percent Deviation of Calculations from Experiment

Assembly (Type)	k_{eff}			$^{238}\text{U}-c/^{235}\text{U}-f$ Experimental Accuracy 4-5%			$^{239}\text{Pu}-f/^{235}\text{U}-f$ Experimental Accuracy 1-2%			$^{238}\text{U}-f/^{235}\text{U}-f$ Experimental Accuracy 2-3%		
	ENDF/B	Set 1	Set 2	ENDF/B	Set 1	Set 2	ENDF/B	Set 1	Set 2	ENDF/B	Set 1	Set 2
Jezebel (Pu)	+0.7	-1.0	-5.5	--	--	--	-7	-6	-6	-4	-6	-3
Godiva (94% enriched U)	+2.5	+0.6	-3.7	+14	+22	+19	-5	-4	-4	+1	-1	+1
Vera 11-A (Pu)	-2.0	-2.8	-6.5	--	--	--	+5	+6	+4	+13	+9	+9
Vera 1-B (94% enriched U)	+0.5	-0.5	-3.0	-1	+1	+3	-1	0	-1	+9	+5	+4
ZPR-III Assembly-6F (47% enriched U)	+0.9	-0.6	-3.8	+6	+9	+10	--	--	--	+3	0	0
ZPR-III Assembly-12 (21% enriched U)	-0.1	-1.7	-3.9	+5	+8	+10	-5	--	--	-4	-9	-9
Zebra 3 (Pu)	-4.6	-5.3	-8.5	+7	+9	+10	-7	-5	-6	-9	-15	-15
ZPR-III Assembly-11 (12% enriched U)	-2.6	-4.1	-6.4	+7	+8	+10	-7	-5	-6	-12	-16	-16
Zebra 2 (14% enriched U)	-2.5	-3.7	-5.2	+3	+4	+6	-5	-5	-6	-8	-12	-14
ZPR-III Assembly-48 (Pu)	-1.3	-2.1	-4.8	+2	+3	+4	-6	-6	-7	-1	-5	-7

In addition to the ENDF/B data set, calculations have been carried out for two other "consistent" data sets in which only the σ_f and σ_c of ^{235}U , ^{238}U , ^{239}Pu , and ^{240}Pu for $E > 10$ keV are different from the ENDF/B values. The consistent nature of these two sets is distinguished by the effort to base these cross sections on the same standard (in this case the σ_f of ^{235}U) and to use the experimentally determined values of cross-section ratios. Set 1 uses White's² measured σ_f of ^{235}U , while Set 2 uses the measurements of Poenitz.³ Set 1 has about the same cross sections as ENDF/B, except above 2 MeV where they are approximately 15% lower. Set 2 has up to 15% lower values for σ_f and σ_c in a fairly large intermediate-energy range.

For each assembly, 26-group cross sections were generated⁴ with the code MC². The assembly k_{eff} and central reaction ratios were calculated by the 1-D S_N transport code SNARG⁵ using the transport approximation. The results are shown in Table I. Within the limitations of such calculations, the following observations can be made:

1. The lowering of the ENDF/B ^{235}U σ_f above 2 MeV (Set 1) provides reasonable k_{eff} values for the hard assemblies.

2. The calculated central $^{238}\text{U}-c/^{235}\text{U}-f$ ratio is generally about 5% high. It can also be seen that, for the uranium assemblies, the k_{eff} values become progressively lower as ^{238}U becomes more important. This is especially evident in the sequence of assemblies 6F, 12, and 11. They have similar spectra, but increasing ^{238}U content. These facts suggest that the ^{238}U $\sigma_c/^{235}\text{U}$ σ_f ratio may be too high.

3. The calculated central $^{239}\text{Pu}-f/^{235}\text{U}-f$ ratios are about 5% low. An increase in the differential ratio used for finding ^{239}Pu σ_f would also improve the low k_{eff} values for Zebra 3 and Assembly-48.

4. While using Set 2 ^{238}U σ_c with Set 1 fission values would give better overall agreement, this would be incompatible with the experimental differential cross-section ratios.⁶ It should be noted that errors giving consistently soft calculated spectra (e.g., ^{238}U σ_n too high) would help explain the discrepancies in paragraphs (2) and (3).

5. A comparison of the k_{eff} values for Sets 1 and 2 indicates that the normalization of Set 2 may be too low.

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Reactivity Effects In Thermal Reactors

Comparison of Experimentally Measured and Analytically Predicted Material Reactivity Worths in the Flexible Critical Experiment (FCX), V. S. Oblock, S. Salah, W. E. Freidhof (W-Astro)

To validate the analytical methods used for NERVA nuclear rocket reactor design calculations, the spatially dependent material reactivity worths of carbon, uranium, and polyethylene were measured in the Flexible Critical Experiment (FCX) reactor¹ at the Westinghouse Astro-nuclear Laboratory.

This assembly consists of two concentric regions of approximately cylindrical geometry. The central region consists of graphite-enriched uranium NERVA fuel with an outer graphite reflector region. For these measurements, all control rods were totally withdrawn leaving fueled followers in place.

Full core length mass changes of U, C, or polyethylene were made as a function of radial position. A boron-lined ion chamber was used to determine the reactor period (converted to reactivity by use of the inhour equation) before and after the known mass perturbation to obtain the material reactivity coefficients. The hydrogen worths were obtained by subtracting the carbon contribution from the polyethylene.

For the measurement of the axial worth of polyethylene, 12 two-inch-long rods, each placed at a constant radius from the axial centerline, were remotely moved internal to the core during slightly supercritical operation. A Pace PC-12 analog computer² calculated the instantaneous reactivity changes caused by the measured movement of these rods.

Material reactivity worths were calculated using the linear first-order perturbation codes, PERT-1D (modified to include leakage corrections for C and H), PERT-2D,³ and the one-dimensional discrete ordinate S_N transport code ANISN.⁴ The radial and axial forward and adjoint angular fluxes used in the PERT-1D and PERT-2D codes were obtained with ANISN⁴ and DOT,⁵ respectively.

Fig. 1. Comparison of the experimentally measured and analytically calculated hydrogen reactivity worths in FCX reactor.

Figure 1 presents a comparison of the measured and calculated core length hydrogen reactivity worths (normalized to PERT-1D calculation near the center of the core) as a function of relative core radius. Table I summarizes the calculated and measured reactivity worths of carbon, uranium, and polyethylene. The agreement between the calculations and the experiment is good.

For all of the analytical methods and materials employed, the measured and calculated reactivity worths agree within $\pm 10\%$.

TABLE I
Comparison of Experimental and Analytical Reactivity Coefficients for
Uranium, Carbon, and Polyethylene in FCX (¢/kg)

Position	Radial Distribution				Axial Distribution	
	Uranium		Carbon		Polyethylene	
	Analytical ^a	Measured	Analytical ^a	Measured	Analytical ^b	Measured
Near the core center	20.6	18.2 $\pm$ 1.1 19.7 $\pm$ 0.7	8.2	8.8 $\pm$ 0.4 8.5 $\pm$ 0.4	291.7	305.0 $\pm$ 5.5
Near the core-reflector interface	8.6	8.8 $\pm$ 0.6	5.1	5.2 $\pm$ 0.4	--	--
Near the center of reflector	17.2	16.0 $\pm$ 0.6	2.5	2.8 $\pm$ 0.4	--	--

^aUsing 52 energy group PERT-1D calculations.

^bUsing 16 energy group PERT-2D calculations.

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Fast-Neutron Spectra and Related Quantities

Fast Spectrum Measurements on a Subcritical Section of the Thermionic Critical Experiment, S. C. Cohen, R. A. Moore, J. L. Russell, Jr. (GGA)

The emergent neutron spectrum from a subcritical section of Assembly No. 1, Thermionic Critical Experiment, was measured using the Gulf General Atomic Linear Accelerator and the time-of-flight method.¹ The subcritical spectrum determination is intended to augment the Thermionic Critical Experiment Program, whose objective is to compare integral neutronic data with analysis in support of the core design of thermionic space power reactors.² Assembly No. 1 consists of a neutronically homogeneous mixture of fully enriched uranium and polyethylene ($H/^{235}U = 0.855$) and a small fraction of stainless steel for structural purposes.³

A plan view of the configuration for the spectrum measurements is shown in Fig. 1. The thickness of the finite slab subcritical ($k_{eff} = 0.64$) assembly was 7.28 cm and the transverse dimensions were each roughly 29.1 cm. A depleted uranium target spaced 14 in. from the center of one face of the assembly provided an isotropic, pulsed photoneutron source spectrum which was measured from 10 MeV to ~100 eV. An extensive combination boron carbide/boric acid shield designed to obviate the room-return problem surrounded the source. The emergent spectrum from the subcritical assembly was extracted through a collimated 50-m flight path positioned at 42° to the normal of a face of the assembly. The high-energy portion of the spectrum (>800 keV) was measured with a liquid scintillator detector and the intermediate- and low-energy (<1 MeV) portions were measured with a lithium glass detector.

The angular flux spectrum was computed using the DTF-IV code⁴ in an S_{10} approximation (Lobatto quadrature) to the transport equation. The GAM-II code⁵ was used to obtain 27-group, P_3 cross sections for the constituents of the subcritical assembly. Energy self-shielding calculations for ^{235}U were performed with the GANDY code⁶ using the unresolved resonance parameters recommended by Schmidt.⁷ The assembly configuration was approximated by infinite-slab geometry, incorporating a DB^2 correction to account for the transverse leakage. Using the measured source spectrum, the neutron source was included in the calculation normal to one face of the slab.

The measured and predicted spectra were normalized to a constant value of integrated fissions and are compared in Fig. 2. The results demonstrate the need for consideration of unresolved resonance self-shielding in ^{235}U , and suggest that the ^{235}U absorption cross-section data may be slightly high at low energies ($100 \text{ eV} < E < 2 \text{ keV}$).

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Thermal Reactor Lattice Experiments and Analysis

Studies of Some Plutonium Exponential Assemblies, Glenn A. Price, Hans Ludwig, Richard Schiller, K. Chandramoleswar* (BNL)

Uncertainties in the value of α for ^{239}Pu in the epithermal energy range have indicated the need for integral experiments to resolve some of the reported discrepancies.¹ The experiments reported in this paper were designed to provide integral data with clean, well-defined lattices of Pu-Al rods, 0.600-in. diam, clad in Zircaloy tubes, 0.668-in. o.d. $\times$ 0.612-in. i.d. The plutonium contains 8.79 wt% ^{240}Pu . Subcritical lattices are composed of fuel rods in a hexagonal lattice, moderated by D_2O , and in tanks with radii either 13.0 or 14.8 in.

Earlier buckling measurements² were reported with similar lattices surrounded by boral and paraffin. The interpretation of those measurements, which used In foil activations, presented some difficulties, presumably due to the boral-paraffin reflector, since it was not possible in every case to establish a unique material buckling.

The data in this paper were obtained using borax in place of paraffin in the reflector region, and using Au and Mn foils in place of In foils. It is found that flux traverses with Au and Mn foils are, in general, more consistent than those made with In foils, possibly due to their sensitivity to higher energy neutrons. In addition, the borax should reflect fewer neutrons than paraffin, thus perturbing the spatial and energy distributions to a lesser degree. These lattices are predominately epithermal; for example, the first lattice in Table II has 73% of its fissions in the energy group from 0.6 to 5530 eV.

The HAMMER³ system of codes, with modifications in the resonance and thermal ranges, was used in the analysis of these lattices. A group structure emphasizing the 0.3-eV resonance in ^{239}Pu was used in the thermal range, and the resonance calculation was carried out using RABBLE.⁴ The thermal library was constructed using ENDF/B data for all isotopes, except for hydrogen and deuterium. In these cases, the Nelkin Kernel was used. The resonance parameters for ^{239}Pu were obtained from measurements by Derrien et al.⁵ In the range above the resonances, the ^{239}Pu cross section was varied by varying α and keeping the absorption cross section constant. Values for α were obtained from measurements by Gwin⁶ in the range 300 eV to 25 keV. In the range above this, α values from an evaluation by Barre et al.⁷ were used. ENDF/B data were also used in the epithermal range for all isotopes except hydrogen and deuterium, which were based on data from BNL-325.

Cadmium ratios and activation ratios for Au, Mn, In, and ^{239}Pu (fission) were measured in two of the lattices and are reported in Table I. Activation ratios, relative to Mn, were normalized by means of exposures in a thermal column. Preliminary calculations of Cd ratios are lower than measured values, in some cases, by as much as 12%.

Table II illustrates the calculated values for k_{eff} at the measured values of B_m^2 using various detector foils. It is observed that the values of B_m^2 are apparently dependent upon the foil material used. This effect may be

TABLE I

Detector	1:1 Pu-Al-D ₂ O (clean)		2:1 Pu-Al-D ₂ O (clean)	
	Activation Ratio	Cd Ratio	Activation Ratio	Cd Ratio
Au	9.02 $\pm$ 0.07	1.024 $\pm$ 0.005	10.06 $\pm$ 0.09	1.04 $\pm$ 0.01
In	7.21 $\pm$ 0.05	1.105 $\pm$ 0.008	8.87 $\pm$ 0.11	1.13 $\pm$ 0.02
Mn	1.00	1.09 $\pm$ 0.01	1.00	1.24 $\pm$ 0.01
Pu	0.358 $\pm$ 0.003	1.797 $\pm$ 0.019	0.654 $\pm$ 0.006	2.77 $\pm$ 0.04

TABLE II

Mod/Fuel Vol Ratio	B ₂ O ₃ Concentration mg/ml	D ₂ O Purity Mole %	Foil Material	B _m ² (m ⁻²)	k _{eff}
0.99	29.8	97.65	Gold	2.5 $\pm$ 0.2	1.0223
0.99	19.0	97.30	Gold	6.5 $\pm$ 0.25	1.0371
0.99	0.0	99.26	Gold	15.5 $\pm$ 0.3	1.0396
0.99	0.0	98.86	Manganese	17.4 $\pm$ 0.2	1.0065
1.49	28.75	98.34	Gold	-3.1 $\pm$ 0.2	1.0013
1.49	0.0	99.05	Gold	19.65 $\pm$ 0.3	1.0127
1.49	0.0	98.96	Manganese	19.8 $\pm$ 0.25	1.0101
1.99	18.9	97.70	Gold	-0.96 $\pm$ 0.2	0.9984
1.99	0.0	98.92	Gold	23.8 $\pm$ 0.2	0.9930
1.99	0.0	98.89	Manganese	23.8 $\pm$ 0.2	0.9930

attributed to the high-energy neutron sensitivity of Mn and the epithermal nature of the neutron spectra.

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Monte Carlo Analysis of Physics Parameters in U-H₂O Lattices, J. Hardy, Jr. (W-BAPL)

The full-range Monte Carlo program RECAP has been used in a consistent analysis of physics parameters measured in seven H₂O-moderated uranium lattices. The motivation has been to test input nuclear data, which constitute the major source of uncertainty. The following parameters are considered: epithermal-to-thermal ratio for ²³⁸U capture (ρ^{28}) and for ²³⁵U fission (δ^{25}), ratio of ²³⁸U-fission-to-²³⁵U fission (δ^{28}), and eigenvalue (λ).

There were four TRX metal rod lattices of aluminum-clad, 0.387-in.-diam uranium metal rods enriched to 1.3% ²³⁵U. These were arranged in hexagonal arrays at water-to-fuel volume ratios of 1.00, 2.35, 4.02, and 8.11. There were, also, three lattices of 1.0-in. natural-uranium slabs in parallel arrays, with water-to-fuel volume ratios of 1.09, 0.50, and 0.16.

The 2.35/1 and 4.02/1 TRX lattices were full; the others were coupled to drivers. Except in the 0.16/1 slab lattice, parameters were measured in regions of asymptotic spectrum and leakage was relatively small. Leakage corrections were obtained from homogenized-lattice calculations with MUFT-MAGMA^{1,2} and with P3MG.³

RECAP described the lattice cell explicitly over the full energy range below 10 MeV. For ²³⁸U, asymmetric Doppler-broadened resolved resonance profiles were described in detail up to 2030 eV, with unresolved resonances to 24,800 eV. The ²³⁸U inelastic scattering and fission cross sections were essentially those of ENDF/B. The ²³⁵U cross sections were based on ENDF/B with resolved single-level resonances up to 61 eV; the fission integral above 0.625 eV was 258 b; the capture integral 136 b. The fission spectrum was that of D. R. Harris. Above 0.625 eV, smooth cross sections, including inelastic scattering transfer matrices, were described in the 54-group MILC⁴ structure. Below 0.625 eV, there were 25 groups, with a P3 Haywood kernel for H₂O. Binding effects were treated approximately up to 10.7 eV.

The comparison of calculated and measured parameters appears in Table I. There is generally good agreement over a wide range of parameter values. Two comments are in order, however.

The calculated δ^{25} results are high in the tighter lattices. A detailed multilevel description (furnished by A. Staub and M. Goldsmith) of the ²³⁵U cross section between 0.625 and 30 eV failed to correct the situation.

A completely straightforward choice of ²³⁸U capture cross sections, with P-wave capture integral of 2 b below 24,800 eV, corresponding to a strength function of 2×10^{-4} , produced distinctly too much epithermal ²³⁸U capture. This led to high ρ^{28} values and noticeably affected δ^{28} and λ . To clarify the situation, the smooth (P-wave) high-energy cross section was reduced in the present calculation. A P-wave capture integral below 24,800 eV of only 0.65 b was required for best ρ^{28} agreement. This is 0.8 b below that needed to reproduce the isolated rod resonance integrals recommended by Hellstrand.⁵ A larger difference has been noted by Askew.⁶

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TABLE I
Comparison of Measured and Calculated Parameters*

Lattice	ρ^{28} (0.625 eV)		δ^{25} (0.625 eV)		δ^{28}		λ
	Experiment	Calc/Exp	Experiment	Calc/Exp	Experiment	Calc/Exp	Calculation
TRX 8.11	0.466 ± 0.010	1.024 ± 0.023	0.0352 ± 0.0004	0.974 ± 0.023	0.0452 ± 0.0007	1.027 ± 0.025	1.0047 ± 0.005
4.02	0.830 ± 0.015	0.995 ± 0.022	0.0608 ± 0.0007	1.013 ± 0.017	0.0667 ± 0.0020	1.001 ± 0.025	0.9991 ± 0.004
2.35	1.311 ± 0.020	1.001 ± 0.019	0.0981 ± 0.001	1.011 ± 0.014	0.0914 ± 0.0020	0.997 ± 0.025	1.000 ± 0.004
1.00	3.01 ± 0.05	0.980 ± 0.018	0.230 ± 0.003	1.030 ± 0.015	0.163 ± 0.004	0.992 ± 0.025	0.9902 ± 0.004
Slab 1.09	1.21 ± 0.03	0.983 ± 0.026	0.124 ± 0.002	1.037 ± 0.020	0.190 ± 0.004	0.979 ± 0.025	0.985 ± 0.009
0.50	2.63 ± 0.06	0.997 ± 0.024	0.267 ± 0.006	1.042 ± 0.024	0.264 ± 0.006	0.957 ± 0.025	0.991 ± 0.018
0.16	12.1 ± 0.2	0.989 ± 0.023	1.10 ± 0.02	1.079 ± 0.020	0.442 ± 0.009	0.950 ± 0.035	0.952 ± 0.036

*²³⁸U P-wave capture integral below 24,800 eV of 0.65 b (see text).

A Computerized Library and Evaluation System for Integral Neutron Experiments, Viktor E. Hampel, Robert J. Howerton, John A. Wade (UCLRL-Livermore)

A computerized library of references to integral neutron experiments has been developed at the Lawrence Radiation Laboratory at Livermore. This library serves as a data base for the systematic retrieval of documents describing diverse critical and bulk nuclear experiments. The evaluation and reduction of the physical parameters of the experiments to a standard numerical format, and their comparison with calculated results using Monte Carlo computer codes and established libraries of neutron cross sections, will make routine appraisal easier, and also make possible improvement of neutron transport calculations.

For several years, evaluated neutron cross sections have been maintained as libraries on magnetic tape.¹⁻⁴ The verification of these stores of neutron data by comparison of neutronic calculations with integral experiments has been of growing concern. This has been expressed most recently during the November 1968 International Conference on the Constructive Uses of Atomic Energy.⁵⁻⁷ There exists indeed a large amount of information on critical assemblies, arrays of fissile materials, and pulsed-neutron experiments. However, their comprehensive evaluation was delayed since most of these experiments conducted during the past decades were designed to be checkpoints for particular projects. Clearly, the advent of fast computers with large memories makes it possible to consider their evaluation now on a systematic and continuing basis.

As in the evolution of independent libraries of neutron cross sections, it is not likely, and may be undesirable, that a single scheme for the task would evolve. Several efforts are in progress.⁸⁻¹⁰ This paper describes the computer system, the objectives, and some results of such an effort at the Lawrence Radiation Laboratory. The program has four aspects: (1) collection and retrieval of pertinent documents; (2) evaluation of specific experiments; (3) neutronics calculation appropriate to specific experiments; and (4) comparison of the results of the calculations with these experiments, followed by reexamination of the models and cross sections used in the calculations. These four aspects will be discussed individually.

1. The LRL library of references to integral experiments now contains more than 1000 citations and their abstracts on magnetic tape. Emphasis was given to experiments with an apparent high degree of accuracy and precision. Experiments directed solely toward nuclear safety or radiation shielding are not included.

A computerized bibliographical retrieval system allows the user to search for the presence of certain terms in the titles, in assigned keywords, and in the full text of abstracts. The user may compose his own search criteria in an unconstrained format, specifying a weighted collection of whole words, roots of words, words with prefixes and suffixes, phrases, and synonyms of any or all of the preceding word classes, in an arbitrary order, and without limitations as to number. This system has also been applied to other types of data files, and is described elsewhere.¹¹

2. The accurate evaluation of experiments is essential. Often it is based on data from different sources since questions arise due to apparent contradictions and insufficiencies which require clarification from the original experimenter(s), and occasionally a reexamination of the original log-books. The collaboration of those concerned with integral experiments is clearly needed in this area and has been gratefully received, notably from D. Callihan, J. H. Chalmers, E. D. Clayton, G. E. Hansen, H. C. Paxton, and J. T. Thomas.

3. The neutronics calculations which simulate the experiments are carried out at LRL with specialized versions of the SORS¹² and O5R¹³ Monte Carlo computer programs. These codes are being linked to a standard magnetic tape library of neutronic descriptions of evaluated experiments. This should improve the precision of calculations and facilitate their repetition as new evaluations of neutron cross sections become available. The computer programs are being written for the CDC-6600 and CDC-7600 computers with future anticipated use of the PHOTOSTORE* memories. Modification of the programs to other software and hardware systems should require minimal effort.

4. The comparison of calculation with experiment is made over a wide range of simple critical configurations with progressively different neutron energy spectra. The strategy consists of systematically investigating individual experiments, with the fissile materials ²³⁵U, ²³⁸U, and ²³⁹Pu being mixed with various diluents of increasing proportions and/or reflected by various materials of different thicknesses. Thus, for any of the fissile species, a conceptual array having three dimensions can be envisioned in which various experimental configurations are arranged by diluent, reflector material, and reflector thickness.

Each direction of such an array accents one particular aspect, i.e., the reaction rates of the fissile species, the non-fissile constituents, and the neutron energy spectrum. The observed trends of deviations between calculations and experiments direct attention to regions in which a reexamination of cross sections and calculational models should be undertaken.

These methods have been tested by neutronics calculations for a group of simple spherical assemblies consisting of a central core of ²³⁹Pu and reflected by 93.2% ²³⁵U, U(nat), W, and BeO. Independent calculations for these evaluated assemblies have been made at the Los Alamos Scientific Laboratory and AWRE, Aldermaston. The results and conclusions of this study are presented.

It is anticipated that the exchange of similar information, and the collaboration with other installations, may lead to a better determination of nuclear constants and to an improvement in the general capability of quantitative numerical predictions for problems in neutron transport.

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Nuclear Criticality Safety

Criticality of Uranium of Low Enrichment in Water, E. B. Johnson (UCC-ND)

As a part of a continuing program investigating the critical dimensions of lattices of uranium rods in aqueous solutions, limited measurements were made with rods containing uranium dioxide, and measurements with unclad metal rods were extended.

In the first series of experiments, the uranium, as UO_2 , was enriched to 3.95% in ^{235}U . The fuel-bearing region of each rod was 1.27 cm in diameter and 138.7 cm long, clad in 0.046-cm-thick Inconel tubing. Because of the limited number of rods, it was possible to make only a few lattices critical in water. However, it was established that the minimum number of these rods that can be made critical, when moderated and reflected by water, is 149, in a square pattern at a pitch of 2.64 cm; this was a 12×13 lattice with seven rods removed from one face and contained 9.5 kg of ^{235}U .

Fig. 1. Comparison of calculated and experimental water-moderated and -reflected minimum critical lattices of U(4.89) rods of several diameters.

Further measurements have been made with U(5) rods 0.234 cm in diameter latticed in water, extending work previously reported.^{1,2} The minimum critical mass in a spherical water-moderated and -reflected lattice is 1.6 kg of ^{235}U . As shown in Fig. 1, which compares the calculated and experimentally determined minimum spherical critical masses, this value is less than those for rods of larger diameter and also less than that for a sphere of aqueous $\text{U(4.98)O}_2\text{F}_2$ solution at a concentration of 45.3 g of ^{235}U /liter, and is greater than that predicted by Clark³ by essentially the same amount as were those reported earlier for rods of large diameters.

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Critical and Near-Critical Graphite-Moderated Arrays of U(93.2) Cylinders, E. C. Crume, J. T. Thomas (UCC-ND)

A series of experiments and complementary calculations using the KENO program¹ on both bare and thick-polyethylene-reflected, graphite-moderated cubic arrays of 8 nominally 15.7 kg U(93.2) metal cylinders, has been started. Data from these experiments supplement that on unmoderated and hydrogenously moderated arrays of the same metal cylinders,² and so will be valuable in the analysis of the effects of array moderation and reflection. Also, the data provide valuable check points for validation of calculational techniques, such as the KENO program, that are used in nuclear criticality safety analysis. This summary presents data and calculational results on the first several experiments in the series.

The arrays consisted of eight units arranged in a cubic lattice. Each unit was constructed of a two-piece graphite block containing a central cavity in which a uranium cylinder was placed. The average diameter, height and mass of the cylinders are 11.494 cm, 8.077 cm, and 15.692 kg, respectively. The Speer Carbon Company Type-873S graphite blocks ($\rho = 1.766 \text{ g/cm}^3$) were stacked in contact with one another on top of a low-density aluminum framework to construct the unreflected arrays. The axes of the uranium cylinders were parallel. In the reflected arrays, pieces of square aluminum tubing were used to separate the blocks from one another and from the 6-in.-thick polyethylene ($\rho = 0.925 \text{ g/cm}^3$) reflector. The inside surface of the reflector was displaced from the surface of the uranium cylinders one-half the surface-to-surface spacing of the cylinders themselves. Pertinent dimensions of the arrays and their respective experimentally determined criticality conditions are given in Table I.

TABLE I
Experimental and Calculated Array Criticality Conditions

Surface-to-Surface Spacing (cm)		U Cylinders ^a	Graphite Blocks	Array U Density (g/cm ³)	k _{eff}	
		Side	End		Experimental ^b	Calculated
25.59	22.91	--	--	0.368	0.996 ± 0.001	0.997 ± 0.005
20.52	17.82	--	--	0.591	0.993 ± 0.001	0.993 ± 0.010
15.42	12.78	--	--	1.038	0.990 ± 0.001	0.994 ± 0.010
24.64	21.95	4.128 ^c	--	0.400	0.9995 ± 0.0004	1.006 ± 0.010
19.55	16.91	4.128 ^c	--	0.852	0.9995 ± 0.0004	1.002 ± 0.010

^aErrors on surface separations are ± 0.02 cm for the unreflected arrays and 0.03 cm for the reflected arrays.

^bBased on assumed $\beta = 0.007$

^cArray had 6-in. polyethylene reflector.

The KENO calculations were done using Hansen-Roach 16-energy-group neutron interaction cross sections.³ The calculated k_{eff} and associated statistical 95% confidence interval bounds of each array are also given in Table I.

Preliminary KENO calculations of unreflected arrays indicated that arrays with uranium surface-to-surface spacings from about 7 cm to > 25 cm should be slightly subcritical. Below 7 cm it is necessary to have the

thickness of the graphite moderator less than the just-critical surface-to-surface spacing of the uranium. Further calculations indicate the possibility of constructing critical unreflected arrays of this type at spacings of the order of 50 cm. The experimental data reported here confirm the existence of a broad range of slightly subcritical spacings for unreflected arrays, although the limits have not yet been defined. The excellent agreement of the KENO calculations using Hansen-Roach cross sections with the experimental results for these arrays is consistent with the excellent agreement with other experimental assemblies found previously.^{4,6}

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The Reactivity Effect of Replacing a Water Reflector with Lead, G. Klotzkin, R. F. Valentine, J. C. Stachew, C. A. Flanagan (W-BAPL)

A series of experiments performed with a critical assembly in the High Temperature Test Facility of the Bettis Atomic Power Laboratory indicate that replacing water by lead in the reflector region of a thermal reactor causes the reactivity of the core to increase. This effect was first noted in a series of Oak Ridge experiments.¹ The HTTF experiments were carried out at room temperature to test the calculational procedures being used to evaluate the effect of a cylindrical lead shipping container on the reactivity of a single PWR Core-2 Seed-2 cluster during the upcoming refueling of the Shippingport Atomic Power Station. The reactivity of a Seed-2 cluster inside the container was calculated to be about 10% higher than the reactivity of the same cluster in an infinite water reflector.

In the experiment, lead was placed in the reflector region of a small water-reflected rectangular core (11 × 10 × 30 in.) and the change in the reactivity was determined as the lead thickness was varied from 0 to 6 in. A stainless-steel (304) container was fabricated to hold the lead and was mounted ~0.25 in. from the core face on one side of the core and extended the entire length of the core.

The experiment consisted of measuring the critical rod-bank height and differential worth of the core alone, the core with the empty container, and with 2, 4, and 6 in. of lead in the container. The rod bump method in con-

junction with a least-square fit to ρ vs h was used to determine the differential worths.

A weighted least-squares method was used to fit a straight line to the inverse cube root of the differential worth vs rod-bank height. The integral of the differential worth from the critical rod-bank configuration to an all-rod-out configuration was then taken to be the excess reactivity of that particular core configuration.

The experiment revealed that the core became more reactive as lead was added to the reflector region, the reactivity increasing by 1.6% $\Delta k/k$ for 6 in. of lead.

Two-dimensional calculations of the experimental configurations were undertaken using the Bettis diffusion theory computer program PDQ07.² The calculations were in good agreement with experiment in predicting the increase in reactivity with increasing lead thickness. These experimental studies confirm the ability of the diffusion theory model to determine the reactivity effect of placing a PWR Core-2 Seed-2 cluster in a lead shipping container. A Seed-2 cluster (~7.4 in. square × 96 in.) contains 19.5 kg of ²³⁵U in UO₂-ZrO₂-CaO in plate form. The keff of a new seed cluster in an infinite water reflector at 68°F, under the double accident assumption of double the nominal fuel loading and that all of the burnable ¹⁰B poison plates were missing, was calculated to be ~0.95 using the PDQ07 code. When this same cluster was placed inside the container (mainly composed of 9.3 in. of lead) and the shipping container and cluster were immersed in water, the keff was calculated to ~1.045. [A one-dimensional P-3 multigroup calculation (P3MG)³ and a two-dimensional Monte Carlo (RECAP) calculation also indicated a similar increase in reactivity.]

The results of both experiment and calculation indicate that the presence of lead in the reflector region of a core may cause a significant increase in its reactivity compared to a water reflector, and should be represented in any calculational description of a system containing lead. It was also found that diffusion theory calculations predicted reasonably well the change in reactivity caused by the lead substitution.

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Characterizing Plutonium Accumulations, J. L. Forstner, J. M. McKibben, D. A. Orth (duPont-SRL)

In large-scale plutonium processing, inadvertent gradual accumulation of plutonium in a potentially unsafe arrangement is a constant hazard. In our experience at the Savannah River Plant, the principal problem has been with filters in vent and vacuum lines in which plutonium entrainment was expected to be insignificant.

When such an accumulation occurs, the method of removing it must not rearrange a hazardous amount of plutonium into a more reactive array. To maintain nuclear safety during removal, information is needed on

the amount of plutonium present, and the degree of moderation. We have used a variety of neutron and gamma measurements to characterize these accumulations, depending on the instruments available at the time.

Usually, the first and simplest measurement is the neutron abundance as determined with a moderated BF₃ tube; this gives a rather wide range for the possible amount of plutonium, since the production rate of neutrons is influenced so greatly by α , n reactions with the light elements in the system. Moderation, interposed tank walls, and multiplication in the source also affect the neutron count. Accordingly, an accurate estimate of an accumulation requires empirical calibration against a known amount of plutonium in as near the same condition and composition as can be devised.

Gamma pulse-height measurements can provide independent estimates of the amount of plutonium, and aid in identifying the reactions that are the source of the neutrons. Scans have been made with a single-channel gamma PHA instrument and, more recently, with a portable 400-channel analyzer, both utilizing NaI crystals. The spectra are analyzed for actinides, for the characteristic gammas from the α , n γ reactions (to aid in interpreting neutron measurements), and for similarities to the distinctive fission gamma spectrum (to estimate the amount of fission that might be occurring).

Preliminary measurements have indicated, on occasion, that the plutonium content either was or could be above the moderated minimum critical mass; i.e., more than 450 to 500 g. In such cases, the amount of moderator has been estimated from neutron energy measurements so that safe handling techniques could be based on the minimum critical mass appropriate to the actual moderator present. The basis of the estimates is that the leakage neutron spectrum from an assembly hardens as the amount of moderator decreases. In the measurements, the neutron spectrum was separated into slow, intermediate, and fast flux ranges with cadmium and boron carbide absorbers. The theoretical distribution between the energy ranges was calculated as a function of moderation, and comparison between the experimental and calculated distribution provided the estimate of moderation.

At the most recent need for assessing an accumulation of plutonium, a portable pulsed-neutron instrument was available. Measurements were able to show that the reactivity of the system in the existing configuration was negligible and that the moderation of the system was low.

We conclude that the best equipment is pulsed-neutron instrumentation and high-resolution gamma spectrometers (lithium-drifted germanium crystals). Lacking these, much can be done with less expensive instruments.

Examples of the applications of these various techniques can be given for three filters which were found to be emitting neutrons. One contained 1100 g Pu, primarily as oxide, in the presence of hydrated salts; another contained 460 g Pu, primarily as fluoride; the last, in a system that had processed ²³⁹Pu, ²³⁸Pu, ²³³U, ²³⁵U, and ²³⁷Np, was determined by gamma PHA and pulsed-neutron measurements to contain essentially no fissile material (confirmed by later analyses).

Criticality Considerations in Basket Design for Irradiated Fuel Shipment, C. E. Newlon (UCC-ND)

Nuclear criticality has become an important consideration in the design of shipping containers for irradiated

fuel. Containers must be shipped relatively long distances despite their size and weight; thus, any limitation on the number of fuel elements per container could result in severe economic penalties.

Performance criteria established by regulatory statutes to ensure that the shipment is safely subcritical must also be met. Nuclear criticality in irradiated fuel shipments can be successfully controlled through the use of $\frac{1}{4}$ -in. boral plates to reduce the excess reactivity of the fuel element array such that k_{∞} , the infinite-medium neutron multiplication factor, is less than unity. This is a requisite for Fissile Class I shipments.

The grid arrangement, shown in Fig. 1, is that of a boral fuel basket designed and fabricated at the Oak Ridge Gaseous Diffusion Plant for shipment of BONUS fuel assemblies in the 30-ton ATCOR cask. Internal cavity dimensions are 36 in. diam $\times$ 116 in. The basket holds 12 superheater or 24 boiler fuel assemblies. Maximum design loading is 26.2 kg ²³⁵U for the boiler fuel, 2.7% ²³⁵U enriched, or 24.6 kg ²³⁵U for the superheater fuel, 3.95% ²³⁵U enriched.

The maximum k_{∞} of 1.455 of the BONUS fuel is reduced by $\frac{1}{4}$ -in. boral, with 50% B₄C core, to a k_{∞} of 0.989 as determined by the ANISN transport code.¹ With the design loading, the corresponding k_{eff} of the cask and fuel basket is 0.905 (Table I). Changing the fuel assembly

Fig. 1. Nuclear criticality safety fuel basket for ATCOR shipping cask.

TABLE I
 k_{∞} and k_{eff} of ATCOR Cask with BONUS
Fuel Basket—ANISN Code

Basket Material	k_{∞} of Fuel Basket ^a	k_{∞} of Cask and Fuel Basket	k_{eff} of Cask and Fuel Basket
0.25-in. Boral (100% B ₄ C Core)	0.960		
0.25-in. Boral (50% B ₄ C Core)	0.989	0.918	0.905
0.25-in. Boral (35% B ₄ C Core)	1.003		
0.125-in. Boral (35% B ₄ C Core)	1.029		

^aCylindrical cell approximation used in determining maximum reactivity k_{∞} of 1.455, of BONUS fuel.

position within the 1-in. clearance of the fuel compartment induced reactivity changes of $< 1\%$.

Three experimental critical square lattices of 0.50-in.-diam $\times$ 54-in. $U(3.95)O_2$ fuel rods, water-reflected and -moderated, provided support data for the code calculations. Edge spacings were $\frac{1}{4}$, $\frac{1}{2}$, and $\frac{3}{4}$ in. between fuel rods. The number of rods in these lattices was 224, 151, and 201, respectively.² The calculations showed a slight positive (conservative) bias of 3 to 4% in k_{eff} . In further criticality tests, using nonirradiated BONUS fuel assemblies, a simulated fuel basket configuration, water-reflected and -moderated, with a $\frac{1}{4}$ - $\times$ 26 $\frac{1}{2}$ - $\times$ 66-in. boron plate, centrally positioned, was shown to have no detectable source neutron multiplication. The basket configuration was critical without the boron plate.

Neutron interaction effects for spent fuel containers are generally small because of the thick cask walls. Again, from Table I, it is noted that the incremental reactivity increase due to interaction,

$$\Delta k_{int} = (k_{\infty} - k_{eff}) ATCOR \text{ cask}, \quad (1)$$

is only about 2%. Interstitial water in an array of such containers would serve only to attenuate the neutron flux and further reduce the small interaction effect.

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Increasing Maximum Enrichment Processing Capability Through Criticality Calculations, D. L. Dunaway (Nat'l Lead-Ohio), W. A. Johnson (USAEC-ORO)

Enriched-uranium residues containing $< 5.0\%$ ^{235}U are recovered at the National Lead Company of Ohio in a plant originally designed for processing natural-uranium ores and concentrates. These enriched residues are dissolved in nitric acid in safe mass batches. The product, uranyl nitrate is blended to a lower enrichment for safe processing throughout the remainder of the plant. Following isotopic blending, the uranyl nitrate is purified by liquid-liquid extraction and converted to uranium trioxide by thermal denitration. Denitration takes place in 500-gal, agitated, gas-fired pots. The pots are cylindrical with concave bottoms and can process ~ 3600 lb U /batch.

Formerly, the maximum enrichment approved for processing in this equipment was 1.25% ^{235}U . At that time nuclear criticality safety was based on mass control. In this study, we sought to define the enrichment limitation by calculating the effects of nitrogen, oxygen, and hydrogen on neutron multiplication during the conversion from $UO_2(NO_3)_2 \cdot xH_2O$ to UO_3 . Samples were taken from the pots throughout the denitration cycle to determine the H/U , N/U , and O/U atomic ratios and uranium density characteristics of the material.¹

Using the ANISN code,² the pot was considered a 500-gal sphere, with a 1-in. steel reflector. KENO³ code calculations included the effects of pot geometry, interaction with adjacent units, and neutron reflection from the concrete floor. k_{eff} values were calculated for the 2.0, 2.1, and 2.2% ^{235}U enrichments. ANISN code calculations indicated $k_{eff} = 1$ at 2.12% ^{235}U ; KENO at 2.23% ^{235}U . An independent study at Idaho Test Station produced similar results.⁴

The most significant relationship is N/U vs H/U , since it is indicative of the amount of neutron absorber present to neutron moderator. Higher O/U ratios yield only slightly higher k_{eff} values. Maximum k_{eff} occurs at $H/U \approx 6$ and drops off sharply at lower and higher H/U ratios.

One pot was picked for the higher enrichment processing; six separate denitrations were sampled and confidence limits were established. Sample data at the 99% confidence level gave $k_{eff} = 0.947$ at 2.0% ^{235}U using ANISN. This code was demonstrated to be conservative (1 to 4%) when ANISN results were compared with experimental values at 3.04, 2.14, and 2.26% ^{235}U uranyl nitrate.^{5,6} Therefore, 2.0% ^{235}U was deemed safe.

This limit requires that the moderators, nitric acid, water, and oil, be excluded from the pot.

The greatest potential source of moderation was the scrubber above the pot. Nitric acid used to scrub nitrogen oxide gases could have entered the pot through some malfunction. This risk was eliminated by venting to the scrubber of an adjacent, inoperative pot so no moderator could reach the process pot.

The gear box above the pot is limited to 12 gal of oil, which is insufficient moderation for criticality.

A HNO_3 wash line was disconnected and the uranyl nitrate feed line was modified so it can be disconnected immediately after charging the pot. Safety showers in the immediate area were enclosed. A dry chemical extinguisher was provided in case of fire. Water hoses were removed from the area. These changes create a dry environment; they allow processing 2.0% ^{235}U in a single pot with adjacent equipment shutdown. If additional denitration requires use of adjacent equipment, and enclosure for each pot operating at 2.0% ^{235}U would be needed.

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Criticality Control in Large Spent-Fuel Dissolvers Using Homogeneous Poison, Richard L. Brunnenmeyer (Bechtel-San Francisco)

The dissolution of spent fuel from light-water reactors in a high-capacity, solvent-extraction fuel processing

plant may require equipment too large for purely geometric control. Soluble neutron poison in the dissolver solution as a primary means of control should be considered. This paper describes a method of evaluating poison requirements and applies it to a typical dissolver configuration.

Fig. 1. Comparison between experimental and calculated critical bucklings as functions of boron concentrations in water moderator. Lattice description: 3.745% enriched UO_2 rods, 0.7544 cm diam, pitch = 1.2522 cm (square), clad thickness 0.0406 cm, poison is boric acid.

The GAMTEC-II Computer Code,¹ which calculates lattice parameters for heterogeneous mixtures of fuel rods and moderator, is the principal tool used in this analysis. To evaluate known deficiencies in the theoretical treatment, a series of comparisons with experimental

data were made.^{2,3} A typical comparison, shown in Fig. 1, reveals a bias toward high values of the calculated buckling.

This error is attributable to two sources: The Fermi age calculation in metal-water mixtures by the moments method appears to give values lower than experimentally observed⁴; and calculations of thermal utilization by the one-group P_0 method results in high values.⁵ Both effects contribute in varying amounts to the overestimate of buckling; however, the trends are predictable and the overall calculation is sufficiently accurate for criticality control purposes.

Evaluating a particular dissolution scheme depends on identifying maximum reactivity combinations of fuel lump size, moderator-to-fuel ratio, homogeneous poison and dissolved fuel concentration, and other parameters. The effects of fuel lump size and fuel in solution have been reported in papers by Richey et al.^{6,7} This work allows a simplified approach involving only the most important parameters. As shown in Fig. 2, the optimum buckling for given materials shifts to lower moderator-fuel ratios as boron content is increased.

Fig. 2. Calculated critical buckling, 0.05-in.-diam UO_2 rods, 3.5 wt% ^{235}U enrichment. Boron dissolved in moderator.

Cross-plotting these data shows buckling as functions of poison concentration for fixed moderator-fuel ratios. Calculation of geometric bucklings for simple dissolver configurations, using empirical extrapolation lengths, enables the designer to quickly determine systems that will be subcritical under optimum reactivity conditions and to specify the required poison concentration.

Using the fuel described above in a cylindrical, fully reflected dissolver basket, 30 in. in diameter, a poison requirement of about 4.5 g/liter of boron, or the equivalent, is indicated as the solution concentration for which the maximum k_{eff} is between 0.95 and 0.98. To this, a working margin may be added to allow for instrument inaccuracy, operator errors, and so on. Many practical dissolver arrangements will require poison concentrations of this magnitude. These are within the solubility range of such poisons as boron and a cadmium.

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Experimental Reactor Physics

 β_{eff} Ratio Measurements, S. G. Carpenter, S. Ramchandran (ANL-Idaho)

There have been a number of suggestions that systematic errors exist in the measurement or calculation of reactivity changes produced by a small sample in fast critical experiments. For example, Little and Hardie¹ indicate approximately 30% difference in certain cores.

An investigation of the possible experimental errors in measuring $\Delta k/k$ has been started as well as measurements of other quantities which must be calculated correctly if one expects to calculate the reactivity of small samples. The first results are determinations of the ratio of β_{eff} for ^{235}U and the fissile materials in recent ZPPR or ZPR-3 cores.

The method used is outlined by Carpenter.² The power history is used to calculate

$$\frac{\Delta k_e}{k_e \beta_{\text{eff}}} + \frac{1}{k_e \beta_{\text{eff}}} = 1 + \frac{1}{n k_e} \times \left\{ \frac{1}{\beta_{\text{eff}}} \frac{dn}{dt} - \sum a_i \exp(-\lambda_i t) [n_0 k_0 + \lambda_i \int_0^t k_{\text{en}} \exp(\lambda_i t) dt] \right\}.$$

After a rod drop, Δk_e and S_e are constants and n is known to a scale factor. The right side of the equation can be calculated since it is insensitive to 1 , β_{eff} , and k_e . If incorrect values for a_i 's and λ_i 's are entered, Δk_e and S_e

TABLE I
ZPPR Core I Measurement

$^{235}\text{U}/^{239}\text{Pu} \beta_{\text{eff}}$	Relative Error ^a	Average, ϕ ^b
+30% ^c	1.036	187.2
+20%	0.811	187.3
+10%	0.586	187.4
As calculated	0.413	187.5
-10%	0.425	187.9
-20%	0.661	188.1
-30%	0.999	188.4

^a Δk_e was picked for each case to minimize the variations in S_e to satisfy the equation at each data point. The listed value is proportional to the RMS error at that minimum.

^bThis value is the subcriticality of the reactor mentioned in a.

^cIt was assumed that the β_{eff} of ^{235}U was higher or lower than the MACH-1 calculation by the % indicated.

will not be constants. It has been assumed that the relative isotopic delayed precursor fractions a_i and λ_i are as given by Keepin³ and only the fraction due to each isotope is unknown. The proper split is found by trial so that the constants are constant. The minor contributions to the delayed neutron from the other isotopes are left as calculated by a MACH-1 diffusion theory calculation which was used to generate the required constants.

Table I shows results from ZPPR Core I. The rod drop of about 188¢ occurred in about 200 msec. Data were analyzed in 0.1-sec intervals for the next 20 sec. The ratio, $(^{235}\text{U}/\text{Fissile})\beta_{\text{effmeas}}/(^{235}\text{U}/\text{Fissile})\beta_{\text{effcalc}}$, was 0.96 ± 0.02 . Measurements were also made on ZPR-3-56

which was similar in composition giving a result of 0.92 ± 0.02 and on ZPR-3-57 with a result of 0.88 ± 0.06 . The latter was a ^{235}U core with only a small amount of ^{238}U . These measurements will continue on other cores.

These initial results show that there is not a relative error in the assumed value for β_{eff} of ^{235}U to cause the errors noted in reactivity. The worth of the drop in ρ is almost independent of the choice of delayed-neutron parameters for results near the drop. For the ZPR-3 Assembly 56, the result is within 1% of 132¢ for any reasonable constants ($\pm 30\%$ in assumed ^{235}U fission rate). This value can be used as a standard for other relative measurements so that only a small uncertainty is inherent in reactivity measurements expressed in cents. This would also be true if the isotopic values for a_i 's and λ_i 's are all allowed to vary with reasonable limits.

If there is any large experimental error in the measurement of $\Delta k/k$, it is in the assumed total β_{eff} , either in the value of β itself or the conversion to β_{eff} . β_{eff} will also be measured and can be used to correct previous values if it is found to be in error.

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An Evaluation of the Fission Neutron Spectrum of ^{235}U , N. M. Steen (W-BAPL)

The objective here is to provide an assessment of both the shape and mean energy of the fission neutron spectrum of ^{235}U for slow-neutron fission in the Laboratory coordinate system.

The mean energy of the ^{235}U spectrum has been established by consideration of calculated reactivity bias in the ^{235}U aqueous nitrate Oak Ridge cylinders as reported by Conley.¹ It is to be noted that calculated reactivity is strongly dependent on the mean energy but is only weakly sensitive to the higher moments of the spectra. Conley reported a set of highly detailed Monte Carlo computations of the reactivity of these assemblies, for which the fast leakage fraction ranged from ~ 2 to $\sim 40\%$. These calculations utilized a ^{235}U spectrum representation which was based on the work of Harris² and has a mean energy reported to be $\bar{E}_{\text{eq}} = 1.9976$ MeV.

These calculations indicated a reactivity bias $\delta\lambda$, that is an increasing function of fast leakage. An approximate theoretical relation which follows this pattern may be obtained from the criticality equation if an error $\delta\tau$ is assumed in the calculated age. The result is $\delta\lambda = \delta\tau/\tau \ln(1-L)$, where L is the fast leakage fraction. Least-squares analysis of the calculated bias indicates that such a relation gives good correlation with the data and that the bias coefficient is $\delta\tau/\tau = 0.0337 \pm 0.0061$.

A review of the differential data used by Conley indicates that the bias can only partially be accounted for on the basis of the reaction cross sections involved. After correcting for the effects of deviant reaction cross sections and calculational approximations, the remaining bias was used to establish the mean energy of the ^{233}U equilibrium fission neutron spectrum to be $\bar{E}_{\text{eq}}^{233\text{U}} = 2.012 \pm 0.010$ MeV. Taking $\bar{E}_d = 0.45$ MeV, implies that the mean energy of the prompt spectrum is $\bar{E}_p^{233\text{U}} = 2.016 \pm 0.011$ MeV.

To determine the shape of the ^{233}U spectrum it is necessary to consider the shape of the prompt fission neutron spectrum of ^{235}U . A sensitive measure of the adequacy of the ^{235}U spectrum at high energy is the experimental determination of the spectrum-averaged $^{16}\text{O}(n,p)^{15}\text{N}$ reaction cross section of 22.6 ± 4 μb obtained by Roy and Shure.³ (The value here has been revised by Shure to account for recent values of yields and decay constants.) The differential data of Stooksbury et al.⁴ indicate a strong resonance at 11.8 MeV, just above the threshold energy (10 MeV). Based on these data, the Harris,² Cranberg,⁵ and Watt⁶ spectra appear to fall off too rapidly at high energy while the McElroy⁷ spectrum or a Maxwellian with $\bar{E} \approx 2.00$ produce too high a neutron population above 10 MeV. If a single Maxwellian is assumed to represent the spectrum, preservation of the $^{16}\text{O}(n,p)^{15}\text{N}$ reaction would require a temperature $T = 1.283 \pm 0.02$ MeV ($\bar{E} = 1.925 \pm 0.03$ MeV), which is incompatible with the value $\bar{E} = 2.028 \pm 0.01$ MeV inferred from reactivity bias or $\bar{E} = 2.00$ MeV obtained by Watt.

Consequently a more general shape function, patterned after recent forms⁸⁻¹¹ suggested in the literature,⁸⁻¹¹ was used. The basic assumptions are:

1. Neutron emission is isotropic in the CM (neutron-fragment) coordinate system
2. The CM emission spectra is of the evaporation form
3. 10% of the neutrons are emitted before scission, the remaining 90% are emitted from the two fragments, each having a separate distribution.

The parameters involved were interrelated, in the case of ^{235}U , to simultaneously preserve the mean energy $\bar{E} = 2.028$ MeV, the $^{16}\text{O}(n,p)^{15}\text{N}$ reaction and to be consistent with the available differential data below 10 MeV.

A similar interrelation was made for the ^{233}U spectrum. In this instance, the interrelation was required to preserve the mean energy, $\bar{E} = 2.016$ MeV, match the spectrum produced by a Maxwellian with temperature $T = 1.283 \times 1.016 = 1.305$ MeV at 11.8 MeV, consistent with the work of Grundl,¹² and produce a mean energy in the Bramblett spheres^{13,14} consistent with $\bar{E} = 2.016$ MeV.

It is of interest that, while the interrelation of the temperatures produces only one independent variable, the uncertainty on the mean energy inferred from the Bramblett spheres is $\sim 1/2$ that obtained if either a Maxwellian or a Watt form, with fixed fragment energy, is used. Also, the ratio of $N(E)^{233\text{U}}/N(E)^{235\text{U}}$ is similar to that observed by Smith et al.¹⁵

The resulting representation of the ^{233}U spectrum has been used, together with a recently evaluated set of ^{233}U cross sections and ENDF/B data to calculate criticality of the ^{233}U aqueous nitrate critical experiments of Gwin and Magnuson and the unreported high leakage experiments of Thomas. In all cases, the calculated reactivity agreed with experiment to within the limits ($\pm 0.25\%$) due to inventory uncertainties. The average error in calculated reactivity for the 11 assemblies is $\delta\lambda = 0.06\%$.

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Reactivity Effects in Critical Facilities Due to Fissile Isotope Decay, R. G. Matlock, R. E. Kaiser, J. M. Gasidlo (ANL-Idaho)

A plutonium critical experiments program, in support of the FTR Development, has been underway at Argonne National Laboratory's ZPR-3 and ZPPR for the past two years.

During the course of this program, it has been necessary to reestablish reference core configurations as

starting points for successive experiments. Over a two-month period of ZPR-3 operation, a comparison of reference cores showed a reactivity loss of about 20 lh ($2 \times 10^{-4} \Delta k/k$). At the time of these measurements, this reactivity change was attributed to temperature changes in both the reactor core and structural components.

Subsequent measurements on ZPPR also showed a systematic reactivity loss, with time, which could not be attributed to temperature effects or other known experimental uncertainties.

A preliminary investigation of reactivity changes due to the loss of radioactive plutonium isotopes led to a more detailed analysis of the effects of ^{241}Pu decay.

First-order perturbation calculations of the respective worths of ^{241}Pu and ^{241}Am were performed using the MACH-1 diffusion code.¹ The problem was run in spherical geometry and ENDF/B cross sections, obtained from an MC² calculation² with FTR-1 core composition. Calculations were performed for uniform changes in concentrations from 1.25×10^{17} atoms/cm³ to 5.0×10^{17} atoms/cm³ for both ^{241}Pu and ^{241}Am . This range of concentrations was sufficient to provide data for decay times ranging from 0 to 140 days. The ^{241}Pu loss contributed 97% of the total calculated reactivity effect, with only 3% from the buildup of ^{241}Am . The calculated effect of ^{241}Am was negative, and therefore added to the effect of loss of ^{241}Pu .

TABLE I

Comparison of Measured and Calculated Reactivity Losses in ZPR-3 and ZPPR

ZPR-3			ZPPR		
Time (days)	Meas (lh)	Calc (lh)	Time (days)	Meas (lh)	Calc (lh)
5	2.57	1.18	6	1.1	1.42
57	23.05	13.52	7.5	1.86	1.78
61.5	23.72	14.58	26	6.65	6.16

The results of the calculations, compared with observed reactivity changes in both ZPR-3 and ZPPR versions of the resumed Phase B FTR-1 core, are presented in Table I. The errors associated with the measured values are on the order of 1 lh or less. In the ZPR-3 core, the ^{241}Pu decay appears to account for only about 60% of the observed effect, whereas in ZPPR, it is entirely possible that the entire effect could be accounted for in this manner. The discrepancy in the ZPR-3 calculations is believed to be caused by day-to-day cell temperature variations and long-term changes in the temperature distribution in the reactor. Such effects would be minimized in ZPPR, where cell temperature is controlled to within $\pm 0.5^\circ\text{F}$.

The problem of fissile isotope decay in a critical facility may not be significant in relating measurements made within several hours of each other, since the reactivity loss is < 0.5 lh/day. However, in a sequence of measurements performed over several days or longer, or in experiments where very high precision is otherwise obtainable, some correction for this effect may well be necessary. The rate of reactivity loss would depend on the type of fuel, higher concentrations of ^{241}Pu leading to larger reactivity effects.

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Fast Reactor Physics—II

Integral Measurements of Fast-Neutron Spectra in Some Uranium- and Plutonium-Fueled Fast Reactor Cores, R. Sher (Stanford)

Fast-neutron spectra in several uranium- and plutonium-fueled cores in the Swedish FRO critical assembly have been studied using the threshold reactions

Each of these detectors is available as metal foils of high purity, permitting easy fabrication, irradiation, and counting. A NaI scintillation counter was used to detect the 20-keV conversion x ray in ^{103m}Rh (57 min), 335-keV gamma ray in ^{115m}In (4.5 h), and annihilation radiation in ^{64}Cu (12.8 h).

Table I shows the compositions of the cores studied, denoted as core 5, Pu A-5, Pu B-5, and Pu C-5; The latter three cores had central zones containing plutonium surrounded by a core 5 driver region. All the cores were surrounded by a thick copper reflector.

All the central spectra were calculated with the SPENG code; spectra as a function of position in the various cores were calculated by SPENG-MONDAY.

For the experimental spectra, a simple three-parameter unfolding scheme was employed; as reference spectrum, the SPENG spectrum at the center of the Pu-A zone was used. The scheme minimizes somewhat the sensitivity of the results to cross-section uncertainties, which are particularly severe for the $^{64}\text{Zn}(n,p)^{64}\text{Cu}$ reaction.

Figure 1 shows the unfolded and calculated spectra at the center of core 5; the agreement is typical of all

Fig. 1. Unfolded spectrum core 5 center.

the central spectrum results. Near the Cu reflector, the unfolded spectra tend to be a little harder than the calculated spectra.

Validity of Fast-Critical Experiments As Integral Tests of ENDF/B, W. G. Davey, A. L. Hess (ANL-Idaho)

The task of the Cross Section Evaluation Working Group (CSEWG) in establishing an Evaluated Nuclear Data File-B (ENDF/B) for use in reactor calculations includes the testing of the data in predictions of integral measurements. Ten benchmark fast criticals have been recommended by the authors¹ for ENDF/B-based analyses and two cases have been extensively studied by several CSEWG participants using multigroup cross sections generated from the July 1968 edition of ENDF/B. The results of such studies are indicated by the calculated central reactivity coefficients for the plutonium-fueled Assembly 48 shown in Table I. These data were selected from a summary of the numerous analyses performed by the CSEWG members. The average calculated k-value for this system (S_4 transport theory) was 0.986.

Table I shows that the measured worths must be increased by 24% to give agreement with the average calculated value for ^{239}Pu and that when this factor is applied to all measured values there is good agreement for many materials.

This reactivity "discrepancy" of 24% is similar to those observed for other fast assemblies, and it has been speculated² that the measurements were incorrect due to errors in the assumed delayed-neutron data. While such a source of error cannot be completely discounted, other experiments³ do not support this conclusion, at least to the degree required to explain the discrepancy, and the authors wish to encourage caution in the discussion of discrepancies.

For example, the benchmark models were generated by the authors specifically to permit the use of homogeneous computations. The critical size was adjusted to allow for the reactivity-advantage from heterogeneity, and thus the calculated k should be unity if the multigroup cross sections used are correct. However, this is the only parameter where the correlation of calculation against experiment is certain even under these ideal

TABLE I
Core Atomic Densities*

atoms/cm ³ × 10 ²²	Core 5	PuA 5	PuB 5	PuC 5
^{238}U	1.96	1.73	-	-
^{235}U	0.498	0.0126	-	-
^{239}Pu	-	0.397	0.662	0.530
^{240}Pu	-	0.0205	0.0342	0.0274
^{241}Pu	-	0.0015	0.0025	0.0020
Fe	0.408	0.707	0.908	0.808
Cr	0.096	0.178	0.233	0.206
Ni	0.048	0.082	0.107	0.095
Cu	-	0.120	0.200	0.160
H	0.604	0.604	0.755	0.151
C	2.77	2.15	4.08	5.02
Al	-	~0.12	~0.12	~0.10

*For further details about these cores, see L. I. Tirén, "Preliminary Programme for Doppler Measurements in Pu-Loaded FRO Cores," TPM-FFX-104 (September 1968); T. L. Andersson et al., "Experimental and Theoretical Work at the Zero-Energy Fast Reactor FRO," ANL-7320, Argonne National Lab. (1966).

TABLE I

Comparison of Reactivity Coefficients Measured at the Center of Assembly 48
and Calculated Using ENDF/B-Based Cross Sections

Material	Measured Worth, (10^{-5} Δk /mole)	Range of Calculated Worths, (10^{-5} Δk /mole)			Measured Worths Renormalized to Av Calculated ^{239}Pu Worth
		Lower Extreme	Average Value	Upper Extreme	
^{239}Pu	106 $\pm$ 1	126	131	134	131
^{240}Pu	19 $\pm$ 5	20.1	21.2	22.6	24
^{238}U	-5.9 $\pm$ 0.1	-7.04	-7.48	-7.88	-7.3
^{235}U	79 $\pm$ 1	92.8	98.9	103.4	98
^{10}B	-89 $\pm$ 1	-90	-96.4	-101	-110
C	-0.05 $\pm$ 0.03	-0.015	-0.11	-0.23	-0.06
Na	-0.14 $\pm$ 0.03	-0.05	-0.15	-0.29	-0.17
Al	-0.42 $\pm$ 0.03	-0.42	-0.49	-0.56	-0.52
Fe	-0.69 $\pm$ 0.04	-0.80	-0.87	-0.98	-0.85
Cr	-0.49 $\pm$ 0.04	-0.66	-1.07	-1.22	-0.60
Ni	-1.07 $\pm$ 0.04	-1.04	-1.20	-1.33	-1.32
Mn	-1.18 $\pm$ 0.04	-1.78	-2.37	-3.67	-1.46
Mo	-4.15 $\pm$ 0.05	-8.42	-9.15	-9.96	-5.12

conditions. The reactivity coefficients were measured in a spectrum generated in a heterogeneous reactor and have not been corrected (as has the critical mass) to allow for the homogeneity of the calculations. It is not known if the difference is sufficient to explain the discrepancy, but caution is essential. Even the good agreement between relative reactivities could be fortuitous since there is no reason to expect that heterogeneous-environment effects would be identical for all types of materials.

Thus, at present, integral testing of the cross sections using fast-critical data may only reveal general areas of poor microscopic data and gross errors or misentries in the files. However, criticality calculations for a series of assemblies may indicate discrepancies dependent on specific materials composing the cores.

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Critical Mass Sensitivity to the Treatment of Heterogeneity Effects in a Large UC Fast Core, Assembly 5 of ZPR-6, R. A. Karam, J. E. Marshall (ANL)

Two main types of heterogeneity effects are associated with fast criticals loaded in plate geometry: fast and resonance. The fast effect depends on the physical shape of the fissile materials, i.e., thicknesses and lengths of columns, and on the physical separation between the fissile and fertile plates. It gives a positive contribution to the reactivity of the system.

Resonance neutrons are readily absorbed at the plate surface resulting in fewer such neutrons reaching the plate interior and thus lowering the average reaction rates. Resonance self-shielding contributes negatively through the fission positively through the capture processes. It contributes either positively or negatively through scattering events; the sign is determined by the relative importance of the neutron before and after scattering.

The critical mass of Assembly 5 of ZPR-6¹ was calculated with various cross-section sets (25 groups) generated from ENDF/B data to evaluate the magnitude of the various types of heterogeneity effects. The cross-section sets were used with 1-D and 2-D diffusion codes and with 1-D transport code. The results, along with descriptions of the various sets, are listed in Table I.

These results show that the equivalence theory correction for spatial self-shielding in ^{238}U resonances decreases the true homogeneous critical mass by 7.7% and increases k_{eff} by 0.47%. Weighting Set B with the 1-D fine structure of the real flux in the unit cell, decreases the critical mass by 5.6% and increases k_{eff} by 0.43%. Bilinear weighting of Set B with the spatial distribution of the real and adjoint fluxes reduced the critical mass by another 5%. Spatial averaging with 2-D unit cell fine structure of the real flux yields similar results to 1-D spatial averaging, indicating that the isotropic leakage assumption made in the 1-D cell calculations is adequate. The difference between transport and diffusion theory calculations is about 0.6% in k_{eff} . The lowest calculated critical mass (1775 kg) would have been reduced further had bilinear weighting of the cross section been used in the transport calculation.

Experimental evidence^{2,3} indicates that the fast heterogeneity effect as inferred from the measured spatial distribution of the ^{238}U fission rates within the unit cell is accounted for reasonably well. However, the calculated central reactivity difference between a homogeneous sample (Set A) and a plate-type heterogeneous sample (Set C) was about a factor of 4 lower than the experimental value. The measured reactivity worth of a nominally 2-in. cube, plate-type heterogeneous sample was 20% (or 0.464 ± 0.004 lh) more than an equivalent homogeneous sample.⁴ Both samples had the core composition of Assembly 5. It appears, therefore, that equivalence theory is not adequate for the treatment of the resonance

TABLE I
Critical Mass Calculations of Assembly 5 of ZPR-6

Cross Section Set Used	Code Used	Critical Mass (kg ^{235}U)	k_{eff} for as-Built ^a System	Remarks
Set A	1-D Diff.	2326	0.976196	Set A generated with MC ² code ⁵ from ENDF/B data for homogeneous composition of Assembly 5
Set B	1-D Diff.	2147	0.980954	^{238}U resonance-region cross sections in Set A corrected for spatial self-shielding via equivalence theory
Set C	1-D Diff.	2026	0.985200	Cross sections in Set B spatially averaged with one-dimensional fine structure of real flux in unit cell ³
Set D	1-D Diff.	2038	---	Cross sections in Set B spatially averaged with two-dimensional fine structure of real flux ⁴
Set E	1-D Diff.	1922	---	Cross sections in Set B bilinearly averaged with real and adjoint fluxes obtained from 1-D transport calculation
Set C	2-D Diff.	1981	---	Two-dimensional diffusion calculation using Set C
Set C	1-D Trans.	1859	---	One-dimensional transport calculation
Set C	1-D Trans.	1755	0.990948	One-dimensional transport calculation using measured extrapolated axial height

The Measured Critical Mass was 1585 kg of ^{235}U .

^aThe k_{eff} calculations were done with the measured extrapolated height of 180.0 cm. Unless otherwise indicated, a calculated extrapolated height (171.4) was used in all calculations.

heterogeneity effect in the complex plate geometry. Consequently, the reasonable agreement between the calculated (1755 kg) and measured (1585 kg) critical mass should not be interpreted as an indication of the accuracy of ENDF/B data and/or the validity of the methods for treating heterogeneity effects. Further refinement and validation of the calculational methods are needed.

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Criticality Data and Experiments Applicable to Nuclear Safety Guidance

Surface Multiplication, R. E. Wilson, R. J. Hall, E. D. Clayton (PNL)

A new criticality parameter, surface multiplication (M_s), was reported on in 1961 by Woodcock.¹ This parameter is defined as the number of neutrons that leave a core boundary as a result of a single entering neutron. Although no measured values of surface multiplication have been previously reported, Monte Carlo calculations^{2,3} have shown it to be a useful and valid index of criticality. Monte Carlo calculations have shown that, at least for fast-spectrum systems, $1/M_s$ varies in an approximately linear manner with linear dimension from zero size through the point of criticality. If verified, this property could allow the prediction of critical size from measurements on systems far subcritical.

The parameter M_s was determined experimentally for systems consisting of a bare cube of plutonium-polystyrene with a beam of neutrons incident on one face. The H/Pu ratio was 15, and 2.2% of the Pu was ²⁴⁰Pu. The beam was that portion of the leakage of an intermediate-spectrum critical assembly which streamed through a 2-in. square hole in a shadow shield. Two independent measurements were made on each cube. The values of $1/M_s$ determined from these measurements are shown in Fig. 1. The measurements were made on systems ranging from 0.44 to 28% of a critical mass.

The reaction rates, which were measured with indium-covered indium foils, were spectrum weighted using the calculated leakage flux on each face of the cubes. The angular distribution of the neutrons, cover foil shielding, and self-shielding were accounted for in the effective indium cross-section calculations. M_s was then determined as the ratio of the corrected reaction rates of the neutrons out to neutrons in.

Fig. 1. Inverse surface multiplication.

An analytical mockup of the experimental system was made and the GEM code² (Monte Carlo) was used to calculate M_s for a series of plutonium-polystyrene cubes. Calculations were made using two simulated beams whose spectra are assumed to bracket the experimental beam spectrum. The peak of the spectra for C and D are at 1.8 and 2.2 MeV, respectively. The code inserts source neutrons perpendicular to the front face of the cube and tracks a neutron until it is lost by leakage or absorption. All neutron progeny are similarly treated. The results are also shown in Fig. 1.

The error limits shown for the experimental points result primarily from uncertainties in the neutron spectra, the indium cross sections, and mapping techniques. In all cases, the two independent measurements on a particular cube agree to within the experimental error. In general, the experimental points are in agreement with the GEM-calculated curves.

These initial experiments are the first reported measurements of surface multiplication and, although of limited accuracy, have established the feasibility of experimental measurements of the parameter.

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Criticality of PuO₂-UO₂ Aqueous Mixtures, L. E. Hansen, S. R. Bierman, R. C. Lloyd, E. D. Clayton (PNL)

Associated with the development of Liquid-Metal Fast Breeder Reactors will be the need to fabricate, process, and recover large quantities of mixed PuO₂-UO₂ fuels. There is, however, a lack of homogeneous experimental criticality data that can be used to provide guidance for establishing nuclear criticality safety specifications for fuels typical of those to be encountered in the LMFBR program. Consequently, one must resort to computed values for guidance.

To provide an interim check on the accuracy of computed criticality data for homogeneous, aqueous mixtures of PuO₂ and UO₂, critical experiments were performed using depleted uranium (0.222 wt% ²³⁵U) layers interspersed within PuO₂-polystyrene systems at H/Pu atomic ratios of 5 and 15. Criticality data for the experimental arrays, as well as multiplication factors computed for the experimental assemblies using the GEM-III Monte Carlo code, are summarized in Table I. The computed multiplication factors are quite consistent, and reasonably accurate, for the various heterogeneous arrangements and

TABLE I
Criticality of Heterogeneous Pu-U Mixtures

System-Averaged Pu/Pu + U Atom Ratios for Region Containing Depleted U Layers	H/Pu Ratio for region Containing U	Layers of Depleted U in Core	Nominal Thickness of U Layers (cm)	Critical Length Containing U Layers (cm)	Critical Quantity of Pu in Region Containing U (kg)	Monte Carlo Calculations of k_{eff} (GEM-III)
I - Two-Region Rectangular Parallelepiped: Uranium-free region of PuO_2 -polystyrene, 2.2 wt% ^{240}Pu , H/Pu = 15; Uranium-containing region of PuO_2 -polystyrene, 11.46 wt% ^{240}Pu , H/Pu = 5; cross-sectional dimensions 31.2×30.8 cm.						
1.0	5	0	0	23.30	49.68	1.0029 ± 0.0065
0.465	5	7	0.61	38.02	70.20	0.9923 ± 0.0047
0.554	5	6	0.46	33.66	64.48	0.9877 ± 0.0054
0.631	5	6	0.30	30.30	59.21	0.9926 ± 0.0051
0.748	5	5	0.20	28.31	57.02	0.9931 ± 0.0052
II - Single-Region Rectangular Parallelepiped: Cross-sectional dimensions of 30.9×30.9 cm; 6-in. Lucite reflector on top and bottom; PuO_2 -polystyrene, 2.2 wt% ^{240}Pu , H/Pu = 15.						
1.0	15	0	0	22.79	23.92	0.9928 ± 0.0069
0.361	15	5	0.61	34.88	32.47	0.9956 ± 0.0060
0.462	15	4	0.46	30.53	29.61	0.9973 ± 0.0055
0.412	15	5	0.46	31.59	30.24	0.9957 ± 0.0056
0.553	15	4	0.30	28.64	28.37	0.9996 ± 0.0060
0.541	15	2	0.61	27.32	27.05	0.9961 ± 0.0062
0.700	15	4	0.15	26.35	26.76	0.9911 ± 0.0062

system-averaged Pu/U+Pu atom ratios examined. The GEM-III code was subsequently used to check transport theory calculations for homogeneous PuO_2 - UO_2 aqueous mixtures at Pu/U+Pu, and H/Pu atom ratios equal to those obtained experimentally. These data indicate that the computed critical dimensions, obtained using the DTF-IV transport theory code, may be slightly conservative from a nuclear safety point of view for systems at an H/Pu ratio of 5, and slightly nonconservative at an H/Pu ratio of 15.

Some experimental data have also been obtained for mixed $\text{Pu}(\text{NO}_3)_4$, $\text{UO}_2(\text{NO}_3)_2$ solutions in the United Kingdom.¹ An analysis of those data indicates that the critical parameters of mixed, neutronically thermal, solutions can be predicted quite accurately using the HFN diffusion theory code.

Survey calculations were subsequently performed for systems containing 8, 15, 30, 50, and 100 wt% PuO_2 . The systems examined were water-reflected and unreflected spheres containing $^{239}\text{PuO}_2$, and water-reflected spheres containing PuO_2 with 15, and 25 isotopic percent ^{240}Pu . The moderation ratios examined extend from dry, unmoderated oxide powders, to very dilute, overmoderated, aqueous solutions.

The calculations for mixed-oxide spheres containing 15 and 25 isotopic percent ^{240}Pu show that the effect of ^{240}Pu which results in equal critical volumes occurring at three different concentrations, previously observed for Pu systems,² occurs for mixed PuO_2 - UO_2 systems as well. Also, it is interesting to note that for systems containing $^{239}\text{PuO}_2$, a subcritical vessel containing ~20% of the mass of $^{239}\text{PuO}_2$ required for criticality in an aqueous solution at 1.0 g PuO_2/cm^3 , could, for example, be made critical through simple addition of a natural UO_2 -water mixture at 1.0 g UO_2/cm^3 .

Minimum critical masses for water-reflected spheres containing aqueous mixtures of PuO_2 and UO_2 were also examined. These results show that the effect of 15 wt% ^{240}Pu on the minimum critical mass of ^{239}Pu is greater than the effect caused by mixing the plutonium with natural uranium even with the uranium content of the mixtures as high as 92 wt%.

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Criticality of $\text{Pu}(\text{NO}_3)_4$ Solution Slabs Having High ^{240}Pu Content, R. C. Lloyd, E. D. Clayton, L. E. Hansen (PNL)

Critical thicknesses for infinite slabs of plutonium nitrate solution have previously been computed from experimental data obtained in spherical and cylindrical geometries.^{1,2} Recently,^{3,4} critical experiments in slab geometry have been reported for solutions containing 48 g Pu/liter (4.6 wt% ^{240}Pu) at two acid molarities. Also,⁴ solutions containing $\text{Pu}(\text{NO}_3)_4$ with 23.2 wt% ^{240}Pu , have been examined for two concentrations of plutonium. Additional, more comprehensive experiments have now been performed at the Pacific Northwest Laboratory in slab geometry for determining the critical dimensions of bare and reflected slabs of $\text{Pu}(\text{NO}_3)_4$ solution with high (18.4 wt%) ^{240}Pu content.

The systems examined experimentally were unreflected, partially reflected with one inch of Plexiglas, and fully reflected with water. The experimental data were acquired using a 42-in., square, variable thickness, slab-shaped vessel.³ Variations of the thickness of the slab,

TABLE I
Criticality of $\text{Pu}(\text{NO}_3)_4$ Solutions With 18.4 Wt% ^{240}Pu

Plutonium Concentration	Acid Molarity	Reflector	Critical Thickness of Infinite Slab of Solution Plus System (cm)	Material Buckling (m^{-2})	Effective Extrapolation Length (cm)
66.5	2.4	Unreflected	17.44	147.8	4.20
66.5	2.4	1-in. Plexiglas	14.43	148.4	5.68
66.5	2.4	Water	12.55	151.4	6.49
160.0	3.0	Unreflected	17.54	141.0	4.46
160.0	3.0	1-in. Plexiglas	14.34	136.0	6.30
160.0	3.0	Water	12.27	145.2	6.90
240.8	4.1	Unreflected	18.93	124.9	4.59
240.8	4.1	1-in. Plexiglas	15.49	122.3	6.46
240.8	4.1	Water	12.67	134.9	7.19
412.1	4.2	Water	13.97	114.6	7.69

TABLE II
Criticality of Infinite $\text{Pu}(\text{NO}_3)_4$ Solution Slabs

$\text{Pu}(\text{NO}_3)_4$ Solution Description	Unreflected Slab for System Measured ^a (cm)		Clean-Unreflected Slab (cm)		Water-Reflected Slab for System Measured ^a (cm)		Clean-Water-Reflected Slab	
	Experimental	Calculated	Experimental	Calculated	Experimental	Calculated	Experimental	Calculated
2.36 M HNO_3 , 66.5 g Pu/liter 18.4 wt% ^{240}Pu	17.44	16.92	19.79	19.73	12.55	12.70	11.38	11.57
3.04 M HNO_3 , 160.0 g Pu/liter 18.4 wt% ^{240}Pu	17.54	17.07	20.05	20.07	12.27	12.03	11.21	11.01
4.14 M HNO_3 , 240.8 g Pu/liter 18.4 wt% ^{240}Pu	18.93	18.09	21.48	21.34	12.67	12.76	11.64	11.78
4.16 M HNO_3 , 412.1 g Pu/liter 18.4 wt% ^{240}Pu	---	---	---	23.11	13.97	13.98	13.01	13.07

^aThin-walled stainless-steel criticality vessel (0.062-in. wall thickness with grid reinforcement); experimental data uncorrected for effects of vessel's walls and room-return neutrons.

and hence the height of the solution required for criticality, permitted a number of slab configurations to be examined for each plutonium solution. Assuming a constant material buckling for each plutonium nitrate solution, with specific reflector conditions, permitted the determination of the critical infinite slab thickness, material buckling, and effective extrapolation length for each of the solutions. These values are given in Table I for the plutonium nitrate solutions examined.

Critical thicknesses for clean slabs of the above-mentioned plutonium nitrate solutions were derived from the experimental data. These values compare quite favorably with the values computed using the HFN multigroup diffusion theory code,⁵ with multigroup constants obtained from the GAMTEC-II code,⁶ as shown in Table II.

These data provide guidance for setting nuclear criticality safety specifications for systems with high ^{240}Pu content, and in addition, they provide clean one-dimensional check points to determine the accuracy of calculational methods.

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Plutonium Metal Criticality Measurements, D. C. Hunt, M. R. Boss (Dow-Colo)

The results of a series of critical mass measurements on alpha plutonium metal ($^{238}\text{Pu} = 0.01 \text{ wt}\%$; $^{239}\text{Pu} = 93.58 \text{ wt}\%$; $^{240}\text{Pu} = 5.90 \text{ wt}\%$; $^{241}\text{Pu} = 0.49 \text{ wt}\%$; $^{242}\text{Pu} = 0.02 \text{ wt}\%$)

assemblies reflected by mild steel (SAE 1018) and/or oil (Texaco No. 522) are reported. The experimental apparatus¹ consists of two aluminum tank-reservoir combinations. The assemblies are secured by aluminum mounting fixtures in the smaller of these tanks. This tank is then transported to the test cell where it is placed inside the larger tank. Both tanks are independently and identically plumbed to allow three different liquid filling speeds.

The plutonium assemblies are either spheres or hemispheres built up from nested subassemblies. Spherical assemblies are measured because of the relative

TABLE I
Experimental Critical Masses*

Geometry	Average Fuel Density (g/cm ³)	Steel Reflector Thickness (cm)	Oil Reflected ^a	Mass Difference ^b (grams)	Critical Mass (grams)		
Hemisphere	16.88 18.32	0	Yes	---	7 752 ± 100 6 687 ± 170		
Asymmetric Sphere	17.31 18.56 17.30 17.97 17.32 18.14 17.31 18.16	0 1 3 5	Yes	2 112 ± 114 3 371 ± 64 2 836 ± 112 2 342 ± 129 3 172 ± 10 2 182 ± 28 3 169 ± 2 2 070 ± 95	6 667 ± 192 6 494 ± 180 7 576 ± 135 7 039 ± 101 8 044 ± 202 7 260 ± 57 8 043 ± 212 7 370 ± 88		
	Hemisphere	16.86	0 0.33 0.67 1.0 1.33 1.67 2.0 2.33 2.67 3.0 3.33 3.67 4.0 4.33 4.67	Yes	---	7 637 ± 100 7 717 ± 137 7 860 ± 152 7 885 ± 156 7 921 ± 157 7 930 ± 158 7 934 ± 158 7 956 ± 158 7 961 ± 158 7 979 ± 158 7 974 ± 158 7 964 ± 159 7 957 ± 159 7 953 ± 159 7 944 ± 159	
	Asymmetric Sphere	17.43 17.41 17.17 16.36 17.22 17.48 17.06 16.81 17.95 16.84 17.61 18.72	0 1 3 5	No	12 086 ± 251 9 713 ± 250 4 002 ± 92 11 377 ± 249 7 162 ± 203 1 190 ± 89 8 929 ± 139 5 226 ± 46 932 ± 55 7 866 ± 102 4 040 ± 101 1 766 ± 94	14 128 ± 220 14 455 ± 272 13 064 ± 135 13 294 ± 216 11 842 ± 109 10 401 ± 112 10 973 ± 215 9 836 ± 153 8 559 ± 140 9 876 ± 188 8 811 ± 148 7 849 ± 90	
		Hemisphere	16.90 16.20 16.19 17.06	0 1 3.33 4.67	No	---	14 500 ± 300 12 800 ± 200 11 250 ± 150 11 040 ± 150

*No correction for interstitial grease loading, mounting fixtures, or tie bolts is included in the critical masses reported in this table.

^aThe difference in mass between the two halves of the critical asymmetric spherical assembly.

^bEffectively infinite (~ 30 cm) oil reflection is present for all oil-reflected assemblies.

TABLE II

Comparison of Experimental and Calculated Perfect Sphere Critical Masses

Oil Reflected	Average Fuel Density ^a (g/cm ³)	Steel Thickness (cm)	Observed Critical Mass ^b (grams)	Computed Critical Mass (grams)	Scaled Critical Mass ^c (grams)
Yes	18.37 ± 0.05	0	6 123 ± 200	5 924	5 474
	21.80 ± 0.1	1	4 813 ± 400	4 985	5 662
	19.94 ± 0.1	3	5 838 ± 300	6 025	5 942
	19.74 ± 0.1	5	6 210 ± 350	5 971	6 210
No	17.35 ± 0.25	0	12 709 ± 590	12 954	9 825
	17.71 ± 0.36	1	9 996 ± 330	10 381	8 121
	17.85 ± 0.26	3	8 271 ± 260	8 326	7 043
	18.46 ± 0.21	5	7 356 ± 360	7 260	6 555

^aDeduced from plots of mass difference (see Table I) vs average fuel density.^bDeduced from plots of mass difference (see Table I) vs critical mass. Corrected for effect of mounts and interstitial grease and oil loading.^cScaled from observed critical masses to a density of 19.74 g/cm³ using computed scaling exponents.

ease with which their critical masses may be calculated. Hemispherical geometries are measured since they are easily assembled, and offer greater measurement precision. The steel assemblies are also constructed from nested subassemblies and have an average density of 7.62 g/cm³. When steel is present in the reflector region, it is always in contact with the plutonium assembly. The oil used in the reflector region has been found to be compatible with metallic plutonium and also has a C:H ratio (0.56) which is close to the O:H ratio of water (0.5).

A total of 41 critical masses are identified. These critical masses are shown in Table I. In Table II, the "perfect sphere" critical masses, deduced from the Table I values, are compared with S_8 DTF³ calculations made using Hansen-Roach³ 16-group cross sections.

The critical masses reported here are of interest since, owing to handling difficulties, only a limited number of plutonium-metal critical mass measurements have been reported.⁴⁻⁸ Specific conclusions include: the critical masses of steel-oil reflected assemblies are increased, relative to oil-only reflection; the curve of critical mass variation with steel thickness, for steel-oil reflection of hemispherical assemblies, shows structure which is similar to that found in previous enriched-uranium metal measurements on spherical assemblies.⁹ The structure observed consists of a pair of inflection points at steel thicknesses of 2 and 2½ cm together with a maximum at ~3 cm. This structure has been shown to be associated with the scattering resonances of ⁵⁶Fe in the energy range⁹ 20 to 100 keV.

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Critical Parameters for 0 to 7% PuO₂-U(Nat)O₂ Rods in Water, C. L. Brown, D. R. Oden (PNL)

The HAMMER system¹ has been found to calculate with reasonable accuracy material bucklings measured experimentally for 1.5, 2.0, 4.0, and 6.6 wt% PuO₂/PuO₂-UO₂ rods in water.²⁻⁴ For 36 experimental lattices, the average of the k_{eff} 's calculated is 1.015; and the range, 0.990 to 1.034 (Table I). In all instances, calculated k_{eff} 's at the points of maximum buckling and minimum critical mass are equal to or greater than unity, which is conservative from the viewpoint of criticality safety.

The HAMMER system comprises programs for calculations of infinite lattice parameters by multigroup transport theory, and of composite reactor parameters by few-group diffusion theory. Program THERMOS performs a multigroup calculation of thermal flux distribution ($E < 0.625$ eV) from integral transport theory, followed by multigroup Fourier transform leakage spectrum correction based on homogenized-cell parameters (multigroup flux-volume weighting). Program HAMLET does the same as THERMOS except the energy range is 0.625 eV ≤

TABLE I

Comparison of HAMMER System Calculations With Experimental Measurements

Water-to-Fuel Volume Ratio	Measured Buckling (M ⁻²)	Calculated Buckling (M ⁻²)	Calculated k _{eff}	Water-to-Fuel Volume Ratio	Measured Buckling (M ⁻²)	Calculated Buckling (M ⁻²)	Calculated k _{eff}
1.5 wt% PuO ₂ -UO ₂ Rods; 0.372 in. diam; 0.027-in. Zircaloy Clad; 7.83 wt% ²⁴⁰ Pu ²				23.50 wt% ²⁴⁰ Pu			
1.123	48.0	48.28	1.001	1.434	62.4	59.35	0.990
1.582	65.1	68.19	1.010	2.445	78.9	82.17	1.010
2.730	78.5	85.59	1.021	3.461	77.1	84.60	1.021
3.813	74.9	82.31	1.021	4.332	72.2	78.86	1.019
5.167	60.9	68.43	1.022	6.195	53.6	55.92	1.007
5.605	55.2	62.51	1.021	6.958	44.0	45.47	1.004
2.0 wt% PuO ₂ -UO ₂ Rods; 0.508 in. diam; 0.030-in. Zircaloy Clad; 7.65 wt% ²⁴⁰ Pu ²				4.0 wt% PuO ₂ -UO ₂ Rods; 0.498 in. diam; 0.034-in. Zircaloy Clad; 18.10 wt% ²⁴⁰ Pu ³			
1.434	88.0	87.59	0.999	1.931	94.7	94.56	1.000
2.445	102.9	111.10	1.023	2.566	107.9	111.17	1.009
3.461	103.4	113.68	1.028	3.624	108.7	121.40	1.034
4.332	97.8	107.79	1.027	4.533	107.9	120.81	1.034
6.195	76.8	83.99	1.020	7.271	88.4	96.34	1.021
6.958	70.0	73.20	1.009	10.874	59.5	61.14	1.004
9.835	32.7	31.09	0.995	11.580	41.1	42.97	1.005
16.54 wt% ²⁴⁰ Pu				6.6 wt% PuO ₂ -UO ₂ Rods; 0.337 in. diam; 0.023-in. Zircaloy Clad; 8.57 wt% ²⁴⁰ Pu ⁴			
2.445	88.4	94.56	1.018	1.681	108.8	106.0	0.992
3.461	89.0	97.27	1.023	2.165	121.5 ^a	126.7	1.012
4.332	81.9	91.52	1.027	4.699	159.6	169.2	1.022
6.195	61.8	68.32	1.018	5.673	159.3	170.8	1.026
6.958	55.5	57.79	1.006	10.754	128.4	140.5	1.029

^aAverage of 2 Measurements.

TABLE II

Minimum Critical Parameters for 0 to 7 wt% PuO₂-UO₂ Rods in Water
(5% ²⁴⁰Pu; natural uranium; 95% TD)

Enrichment (wt% PuO ₂)	Mass (kg Oxide)	Volume (liter)	No. Rods ^a	∞ Cylinder Diameter (in.)	∞ Slab Thickness (in.)	Areal Density (kg Oxide/ft ²)
0.3-in.-diam Rods						
0.7	278.0	78.0	742	15.90	8.60	50.00
1.0	164.0	66.0	492	13.89	7.28	36.80
2.0	69.2	35.0	243	11.05	5.45	21.00
3.0	45.8	26.8	170	10.00	4.76	15.80
4.0	35.7	22.9	136	9.42	4.40	13.20
5.0	29.9	20.5	119	9.08	4.17	11.60
7.0	24.0	18.1	98	8.64	3.89	9.95
0.5-in.-diam Rods						
0.7	300.0	98.5	290.0	16.01	8.64	55.2
1.0	185.0	66.5	200.0	13.90	7.31	42.1
2.0	85.5	37.0	108.0	11.28	5.59	26.3
3.0	60.8	29.0	81.5	10.32	4.95	21.0
4.0	49.8	25.3	68.8	9.78	4.59	18.4
5.0	43.8	22.9	61.8	9.43	4.39	16.9
7.0	37.1	20.4	54.2	9.02	4.12	15.1
0.7-in.-diam Rods						
0.7	337.0	101.0	165.0	16.20	8.77	62.0
1.0	213.0	70.0	117.0	14.20	7.47	48.8
2.0	107.0	40.3	68.2	11.62	5.80	32.5
3.0	80.0	32.1	53.8	10.69	5.19	27.0
4.0	67.3	28.1	46.8	10.18	4.84	24.0
5.0	60.2	25.7	42.6	9.85	4.66	22.1
7.0	52.2	23.0	38.0	9.47	4.38	20.2

^aRod infinite length; and in cylindrical geometry.

$E \geq 10$ MeV. Collision probabilities are computed under the assumption of cosine currents crossing region boundaries, and the energy spectrum during moderation is computed as in MUFT. Program FLOG utilizes the four-group constants produced in THERMOS and HAMLET to perform a one-dimensional many-region reactor calculation.

Based on the above correlation, the HAMMER system was used to calculate minimum critical parameters for 0.3-, 0.5-, and 0.7-in.-diam. rods in the enrichment range 0 to 7 wt% $\text{PuO}_2\text{-U(nat)O}_2$ (Table II). Minimum values were obtained by plotting each critical parameter for each rod diameter and enrichment as a function of water-to-fuel volume ratio. This information is needed to help establish reasonable criticality safety limits for handling, storage, and shipping of $\text{PuO}_2\text{-U(nat)O}_2$ fuel rods and assemblies to be utilized in the Plutonium Recycle Program.

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Enriched-Uranium-Metal Criticality Calculations, Douglas C. Hunt, Donald C. Coonfield (Dow-Colo)

The results of a series of criticality calculations on 93.19% enriched-uranium-metal systems is presented. The assemblies considered are spherical shells (effective density 18.14 g/cm^3) which are internally moderated and fully reflected by a pure hydrocarbon oil of density 0.882 g/cm^3 and a C/H ratio 6.8. The shell inner radius is varied from zero (solid sphere) to 12 cm. The calculations are done with the computer code DTF¹ which uses Carlson's S_n method² to approximate solutions of the Boltzmann transport equation by difference techniques. In all calculations, 16 energy groups are used and linearly anisotropic scattering is assumed. The results are compared with the experimental values obtained by Tuck.³

The factors considered in the calculations are the spatial mesh in each material region, the angular mesh, and the cross-section set assumed. The spatial mesh effect is studied by individually varying the number of space points in each region with the number held fixed in the other regions. Uniform meshes are used as the results are simpler to interpret and less ambiguous. It is found that eight mesh points per centimeter in the uranium and one mesh point per centimeter in the moderator region are sufficient to saturate the effect of adding more mesh points.⁴ This is approximately one mesh point every three transport mean-free-paths for thermal neutrons in these regions. In reflector regions, the mesh effect saturates if 15 mesh points are specified between the fuel surface and the reflector thermal flux peak and an additional 15 points between the flux peak and the outer reflector boundary. With regard to angular⁵

TABLE I
Comparison Between Calculational and Experimental Results

Shell Inner Radius (cm)	Observed Critical Radial Thickness (cm)	Computed Critical Radial Thickness (cm)
0	6.85 ± 0.03	6.83
4	3.39 ± 0.04	3.39
8	2.05 ± 0.05	2.03
12	1.75 ± 0.05	1.75
∞	---	1.73

meshes, it was found in all cases that S_{16} calculations were needed to saturate the angular mesh variation effect.

Cross-section sets due to Howerton⁶ and Hansen and Roach⁷ were tried in the calculations. The group cross sections obtained from the Howerton data are found to be unsuitable for this type of calculation, while the Hansen-Roach sets for C, H, and ^{235}U are satisfactory. For ^{238}U , the Hansen-Roach set is used outside of the resolved resonance energy region (3 eV to 3 keV), while in this region, sets obtained from infinite-dilution resonance integrals are used. This change makes the Hansen-Roach ^{235}U and ^{238}U cross-section sets self consistent.

A comparison of the computed and observed critical radial thickness appears in Table I. It is concluded that a proper handling of spatial and angular mesh effects and a suitable cross-section set specification yield excellent agreement between theory and experiment for enriched-uranium-metal systems in which the neutron energy range of importance varies from fission to thermal. The most sensitive factor in the calculations is the angular mesh specification where, for example, an S_4 calculation may underestimate the critical sphere radii by as much as five percent.

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Slab Cylinder-Array Experimental Data, Grover Tuck, Harold E. Clark (Dow-Colo)

Frequently, the nuclear safety engineer is concerned with the criticality safety of systems involving fissile

TABLE I
Experimental Data

[UO₂(NO₃)₂ solution, 490 g U/liter ²³⁵U (93.2%);
cylinder walls ~ 0.25 cm;
(slab tank bottom 0.6 cm)]

Number of Cylinders in Array	Inside Diameter of Cylinders in Array (cm)	Array Height (cm)	Critical Slab Thickness (cm)
Minimum Reflection ^a			
0	0	0	12.8 ^b
1	11.0	100	12.6
4	11.0	100	12.4
9	11.0	100	12.1
16	11.0	25	11.8
		50	11.6
		100	11.5
1	16.3	25	12.3
		50	12.2
		100	12.2
4	16.3	25	12.0
		50	11.8
		100	11.7
9	16.3	25	11.4
		50	10.9
		100	10.4
16	16.3	25	9.5
		50	5.4
		78	array alone critical
1	21.3	25	11.4
		50	11.2
		100	11.2
4	21.3	25	10.7
		50	9.8
		100	8.7
9	21.3	25	7.9
		47	Array alone critical
16	21.3	25	0.9
		26	array alone critical
Box Reflector (10.2 cm thick)			
0	0	0	10.3
9	13.6	25	9.3
		50	8.6
		100	7.3
16	13.6	25	8.2
		50	5.3
		92	array alone critical

TABLE I continued

Box Reflector (5.1 cm thick, top and sides; 10.2 cm thick on bottom)			
16	13.6	25 50 100	8.5 5.9 0.6
Box Reflector (2.5 cm thick, top and sides; 10.2 cm thick on bottom)			
16	13.6	25 50 100	9.0 7.4 4.2
Box Reflector (none)			
16	13.6	25 50 100	11.3 10.7 10.0

^aOnly reflection was walls, floor (137 cm away), and test equipment.

^bAverage of 9 measurements over course of all slab-cylinder measurements.

solution cylinders interacting with solution slabs. Typical examples are process-area floor drains or leaks in one of a group of vessels or pipes in a dry box. Previous work in this geometry has been done where the axes of the cylinders are parallel to the surface of the slab.¹ The measurements reported here are for the case where the axes of the cylinders are perpendicular to the surface of the slab.

The experimentally measured critical parameters given here are for a 490 U/liter uranyl nitrate solution system which consists of an array of vertical cylinders resting on and interacting with a 120-cm-square horizontal slab. The parameters varied were the array height (0 to ~100 cm), the number of cylinders (0 to 16), the cylinder diameter (11.0 to 21.3 cm), and the slab thickness. The slab-array systems, having 9 and 16 cylinders of 13.6-cm diam, were reflected with a 10-cm-thick reflector. All other measurements were essentially unreflected. The critical thickness of the slab alone was also measured.

The cylinders in the arrays were positioned in a square lattice with equal spaces between cylinder centerlines and half this space between the outer cylinder centerline and the edge of the slab. A Plexiglas "box" reflector was used for the reflected measurements. The bottom and sides of the box reflector closely fit the slab tank. The top of the box was 160 cm above the slab tank bottom.

A summary of the data is given in Table I.

One noteworthy observation is the small change in critical slab thickness due to the 11.0-cm-diam, 100-cm-high solution cylinders placed on the slab. One cylinder produces <2% change in critical slab thickness. Sixteen of these cylinders spaced over the slab produced only a 10% change. The effect on critical slab thickness is small because the cylinder arrays alone are well subcritical and hence contribute little to the reactivity of the slab.

The data have been fitted using the equilateral hyperbola equation.^{2,3} The calculated equation fits the

experimental data to within 5%. The asymptotes have been interpreted as the infinite slab critical thickness and the cylinder diameter of each cylinder in the array at an infinite height when the array is critical.

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Hand Assembly Safety Rule Failure, H. F. Finn (UCLRL-Livermore)

Critical experiments with arrays of plutonium cylinders have been performed at the Lawrence Radiation Laboratory, Livermore, in support of its criticality safety efforts since August 1965. The basic plutonium part is a 3-kg mass of α -phase plutonium (93.6 wt% ^{239}Pu) in the shape of a right circular cylinder with a height of 4.63 cm and a diameter of 6.53 cm. Array units are composed of single or double parts, unmoderated or surrounded by cylindrical moderator jackets. Units and internal spacers are loaded into columns which are fixed to a split-table machine for remote final assembly. Since the arrays to be studied vary from a $2 \times 2 \times 2$ configuration to a $5 \times 5 \times 5$, the portion of the array which must be hand-loaded on the assembly table will vary from one-half to as much as two-thirds of the full array. The safety of hand assembly operations then becomes a significant part of the experimental program.

The first safety criterion to be used was the rule which limited hand assembly to those partial arrays which, when reflected by a large polyethylene slab, had multiplications not greater than 10. When this rule proved to be overly restrictive, an effort was made to remove the reflector from the safety criterion. The proposed rule would allow hand assembly of any unreflected system having a multiplication of 10 or less. This was considered to be a reasonable application of the guidance stated in Ref. 1.

Calculations indicated that a small bare plutonium sphere of $k_{\text{eff}} = 0.90$ (normal-mode multiplication of 10) when 2π water reflected increased in k_{eff} to only 0.985; i.e., remained subcritical. Experiments showed that a small, bare multiplication-of-ten system had only a multiplication of 16 when one-side reflected, and 44 when two-side reflected by massive polyethylene slabs. Further experiments with compact (touching) arrays showed that the smaller the system, the larger the reactivity effect due to reflection. These studies were thought to be conclusive proof that a bare Pu array of the units in question, with a multiplication of 10 or less, could not be one-body reflected to critical, and the rule was approved.

In July 1968, two-thirds of a $3 \times 3 \times 3$ array of 6-kg units were hand-loaded in a configuration which had a multiplication of 10. A subsequent experiment showed that this configuration could be reflected into a supercritical state with a single large (60-in.-high $\times$ 14-in.-wide $\times$ 6-in.-thick) polyethylene body simulator placed against one face of the array. Analysis of the experiment showed that of the neutrons escaping from the reflector, more were returning to the array than in earlier configurations, and into regions of greater importance.

In the earlier experiments with compact systems, it was noted that a smaller system has a greater fraction of its leakage neutrons impinging on a body-simulating reflector than a larger one, so that body-reflection effects were thus believed to be inversely related to system size. The few experiments performed seemed to support this hypothesis. It is now apparent that the changes in the system seen by the returning neutrons can more than compensate for this effect.

As a result of these studies, we have modified our procedures to require directly applicable body-reflection safety experiments for all systems containing more than two 3-kg parts. Design studies are also proceeding on equipment modifications that will allow the columns to be loaded remotely onto the assembly table. If there is a lesson to be learned from this experiment, it is that generalizations from a limited class of experiments are to be avoided when personnel criticality safety is involved.

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Criticality Problems of Synthetic Actinide Elements

The Synthetic Actinide Elements—From Discovery to Manufacture, *Glen T. Seaborg (USAEC)*

The first synthetic actinide element, neptunium, was discovered in 1940. The last element of the actinide series, lawrencium, was created for the first time in 1961. These and the nine other intervening elements have added a new dimension to science, technology, industry, medicine, and politics in an extraordinarily short period of time. Each synthetic actinide element from atomic number 93 to atomic number 98 (with the exception of berkelium, atomic number 97) can now be manufactured in essentially any desired quantity, a truly remarkable achievement. The high points of the history of the actinide elements are traced, production methods are described, and a forecast is given of the manufacturing levels to be expected during the next decade. An analysis is presented of the current and near-term applications of the various isotopes of the actinide elements.

Critical Masses of Fissile Transplutonium Isotopes, *H. K. Clark (duPont-SRL)*

Although the nuclides ^{238}Pu , ^{241}Am , ^{243}Am , ^{244}Cm , and ^{252}Cf are not fissile, and hence cannot be made critical in a well-moderated system, they are necessarily accompanied by other nuclides that are fissile. Because the fission cross sections of these other nuclides in some cases are very large, mixtures containing them can become critical when moderated, even though their concentrations are small. Care must therefore be taken when processing the nonfissile nuclides in a moderating medium to maintain the amounts or concentrations of fissile nuclides below subcritical limits.

Critical masses were calculated for the fissile nuclides ^{242m}Am , ^{243}Cm , ^{245}Cm , ^{247}Cm , ^{249}Cf , and ^{251}Cf as a function of the concentration of the nuclide in water-reflected metal-water mixtures. The cross sections used in these calculations were those currently in use at the Savannah River Laboratory.¹ Values of ν were calculated for Am and Cm from the formula of Gordeeva and Smirenkin,² which is said to be valid for $N < 152$, where $N = A - Z$. For ^{251}Cf , the formula is not valid. Comparison² of ν for spontaneous fission of ^{240}Pu , ^{242}Pu , ^{242}Cm , ^{244}Cm , and ^{246}Cf with ν for thermal fission of ^{239}Pu , ^{241}Pu , ^{241}Cm , ^{243}Cm , and ^{245}Cf shows an average difference of about 0.7. On this basis, ν for ^{251}Cf was taken to be $3.78 + 0.7 = 4.48$. The nuclear data are given in Table I.

TABLE I

Nuclear Data for Transplutonium Nuclides

Nuclide	2200 m/sec		Resonance Integral		ν
	σ_f	σ_c	Fission	Capture	
^{242m}Am	6400	1600	1700	300	3.18
^{243}Cm	700	250	1860	200	3.28
^{244}Cm	1.21	13.29	12.6	650.4	3.20
^{245}Cm	2438.5	430.5	600	150	3.30
^{246}Cm	0	8.7	0	103	3.21
^{247}Cm	525.7	63.7	202	33	3.31
^{249}Cf	1735	265	694	106	3.70
^{251}Cf	4606	1839	490	490	4.48

TABLE II

Minimum Critical Masses of Fissile Transplutonium Nuclides

Nuclide	Mass, g		Concentration ^a g/liter
	Critical	Subcritical Limit	
^{242m}Am	23	10	5
^{243}Cm	213	150	40
^{245}Cm	42	25	15
^{247}Cm	159	120	60
^{249}Cf	32	20	20
^{251}Cf	10	3	6

^aApproximate concentration at which minimum critical mass occurs.

A very simple technique,³ which gives fairly good results for ^{233}U , ^{235}U , and ^{239}Pu , was used to calculate the critical masses. For ^{242m}Am and ^{245}Cm , supporting calculations were made with the HAMMER system,⁴ and approximately the same results were obtained. The calculated critical masses are given in Table II, together with masses that are judged to have sufficient assurance of being subcritical.

Calculations were also made for mixtures of ^{245}Cm and ^{244}Cm . A mass of ^{245}Cm that is assuredly subcritical may be calculated from the formula

$$M_{245} = 25 + 1.0 \times R \text{ grams,}$$

where R is the atomic ratio of ^{244}Cm -to- ^{245}Cm .

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Criticality of Transuranium Actinides—Undermoderated Systems, *S. R. Bierman, E. D. Clayton (PNL)*

Interest in the criticality of the 14 elements making up the actinide group has, in the past, been limited essentially to ^{239}Pu and the isotopes of uranium having mass numbers of 233 and 235. However, some of the lesser known actinides are beginning to find applications for which they are peculiarly adapted and, consequently, are, or will be, produced in significant quantities with respect to criticality. To permit the production, recovery, and

Fig. 1. Infinite multiplication constant-metal atoms in water.

use of these actinides, even on a developmental scale, reliable criticality data, both experimental and calculational, must be obtained. Otherwise, an excessive and unnecessary financial burden will be incurred by over-restrictive controls in the processing of these materials.

Except for ^{239}Pu , experimental data exist for only two of the transuranium actinides. Data from replacement-type experiments by Stubbs et al.¹ indicate that the bare spherical critical mass of 19.6 g/cm^3 ^{238}Pu metal should be slightly less than that for ^{239}Pu metal. Employing similar experimental techniques, Barton² obtained $27.7 \pm 2.5 \text{ kg}$ for the bare spherical critical mass of 13.5 g/cm^3 ^{244}Cm metal. Compounding this lack of experimental data is the lack of complete, evaluated sets of nuclear data with which reliable calculations can be made. However, in an effort to establish some basis for the safe handling of these actinide elements and to act as a guide in planning possible future criticality experiments, survey calculations have been made at Battelle-Northwest to determine criticality parameters of those transuranium actinides that appear to have the greatest potential for application. To date, the calculations have concentrated on ^{237}Np (used as a neutron target for producing ^{238}Pu), ^{239}Pu (used as an α -decay heat source), ^{241}Am (used as a neutron target for producing other actinides and for its 60-keV gamma ray), ^{242}Am (highest known cross section for thermal-neutron fission), and ^{244}Cm (used as a heat source and as a starting material for higher actinides such as ^{252}Cf).

The nuclear data, used in the calculations, are believed to be the best currently available and are from various sources, including results of the Los Alamos nuclear explosion cross-section measurements³—some of which are preliminary. To make the calculations, the nuclear data were averaged over 68 quarter-lethargy units between 10 MeV and 0.414 eV, and 0.01-eV energy intervals between 1 and 0.05 eV, and 0.001-eV energy

duced in the computer code GAMTEC-II⁴ to produce 1 thermal group and 32 broad epithermal groups appropriately structured to define the respective cross sections as a function of energy. These multigroup constants were then used in the neutron transport theory code DTF-IV⁵ for making the criticality calculations. Some results are shown in Table I where the critical mass for both bare and reflected metal spheres, at theoretical densities, are presented along with k_{∞} values for each of the indicated actinides.

An important factor in the criticality of the actinides is that those containing even number of neutrons characteristically have a fission threshold and experience little or no subthreshold fission. Consequently, the inelastic scattering process, which degrades the neutron spectrum, is particularly important in determining the criticality of these actinides. Also, the presence of small amounts of moderator has the same effect on the neutron spectrum and can easily reduce k_{∞} of these nuclides to less than unity. This is illustrated in Fig. 1, where calculated k_{∞} values are shown as a function of metal atom concentration in water. For the even nuclides, ^{238}Pu , ^{240}Pu , ^{244}Cm , and ^{241}Am , k_{∞} rapidly becomes less than one upon the addition of water to the theoretical metal system and criticality is no longer possible for these nuclides.

The even-neutron nuclides discussed herein represent a class of elements in which criticality is reduced, rather than enhanced, through moderation. In the undermoderated state, however, the critical mass for some of these nuclides is, in fact, less than that of the common fissile nuclides ^{233}U , ^{235}U , and ^{239}Pu . This may be somewhat unexpected, but is not surprising. Thus, the nuclear criticality of these nuclides must be considered in future operations that might involve significant quantities of the material. Also, it is evident that experimental criticality measurements, and improved cross-section data, are desirable to confirm the results.

TABLE I
Calculated Critical Radii and Masses of Actinide Metals in Spherical Geometry

Actinide	g/cm ³	k _∞	Critical Radius, cm			Critical Mass, kg		
			Bare	Water Reflected ^a	Steel Reflected ^a	Bare	Water Reflected ^a	Steel Reflected ^a
²³⁷ Np	20.45	1.6389	8.47	---	---	52.1	---	---
²³⁸ Pu	19.6	2.8055	4.44	4.08	3.80	7.2	5.6	4.5
²³⁹ Pu	19.6	2.8612	4.90	4.0	3.87	9.7	5.2	4.8
²⁴⁰ Pu	19.6	1.636	12.45	12.18	10.55	158.4	148.3	96.4
²⁴¹ Am	11.7	1.9436	13.23	12.90	11.35	113.5	105.2	71.6
²⁴² Am	11.7	3.0117	5.55	4.26	4.28	8.4	3.8	3.8
²⁴⁴ Cm	13.5	2.4552	7.43	7.30	6.31	23.2	22.0	14.2

^a20-cm reflector.

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Criticality Safety Analysis

The Use of Differential Current Albedos in Monte Carlo Criticality Calculations, G. E. Whitesides (UCC-CTC), J. T. Thomas (UCC-ND)

In the evaluation of the safe storage and transport of arrays of fissile materials, an important consideration is the criticality condition of the array when it is reflected by various materials. The materials of most interest are those that have low absorption properties and high scattering properties. Previous work¹ has shown that the Monte Carlo method is an accurate and relatively inexpensive method of determining the criticality condition of arrays of fissile materials, except in the case of reflection by a thick low absorbing medium. The "adjoint biasing" technique² reduces by as much as a factor of 4 the expended computer time required to obtain an answer to a given accuracy for such systems. The time required, however, is still too great to allow for the study of a large number of configurations of an array. To provide the capability of performing routine criticality search studies with the Monte Carlo method, an albedo technique has been developed and used successfully to reproduce experimental results.

To accurately treat the neutron at the core-reflector interface, it is necessary that one know the probability that the neutron will return from the reflector, the emergent energy distribution relative to the incident energy, the emergent angular distribution relative to the incident angle, and the spatial distribution relative to the incident coordinates.

By restricting the application of this technique to rectangular reflectors with "large" faces, the consideration of the spatial distribution relative to the incident coordinates can be eliminated. The assumption is then made that the neutron is returned at exactly the same point at which it strikes the reflector. For criticality safety purposes, the result will always be conservative for "small" faces, and will tend toward an exact solution

as the face becomes large relative to the area near corners which affect the return probabilities.

Shown in Fig. 1 are the results of a study made to determine the effects of face size for a water-reflected $2 \times 2 \times 2$ array of highly enriched uranium metal units.

The other needed differential albedo information may be obtained from S_N calculations by performing a "fixed source" calculation in which the source is directed into a plane, consisting of the reflector material, with a given angle and energy. Calculations of this type are made for all angles and all energies. The fractions returned relative to the incident neutron provide information whereby a neutron that strikes the reflector can be returned with the correct angular and energy distribution.

To apply the differential albedo, it is necessary only to note the energy and angle of the neutron striking the reflector, look up the probability of return for a neutron of that given angle and energy, and if the neutron is to be returned, select the angle and energy of the returned neutron from the given angular and energy distribution.

This differential albedo consists of the characteristics of the reflector and is independent of the fissile material enclosed by the reflector. Consequently, once the differential albedo information is generated for a given reflector material it may be used to compute any array with no need to perform additional reflector calculations.

The computer time advantage of using the differential albedo information over that required to do the actual tracking in the reflector will vary depending upon the importance of the reflector to the system. For uranium metal systems, reflected by water, the time spent tracking neutrons in the reflector often is 10 to 20 times that spent tracking in the fuel. Calculations have been made which indicate it is possible to compute such systems with the albedo method in 1/15 the time required to rigorously treat the reflector.

For most survey-type studies in which one desires to determine the criticality condition of reflected arrays of fissile material, the differential albedo offers a powerful and accurate tool that can significantly reduce the cost of computing such systems. For arrays with "large" reflector faces, the differential albedo technique should eliminate completely the need to track in the reflector, since the results can be shown to be in agreement with actual tracking results.

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Pulsed-Neutron Determination of k_{eff} and τ in Low-Enrichment Uranyl Nitrate, R. H. Chow (Cal St Coll-Long Beach), S. R. Bierman (PNL)

Values of k_{eff} and the neutron age to thermal energy have been determined by pulsing bare, homogeneous assemblies of anhydrous uranyl nitrate salt of 11.2 H:U atomic ratio, and 2.26 wt% ^{235}U enrichment. One assembly was a 21-in. cube; the other a parallelepiped, 21 x

Fig. 1. Difference between rigorous solution and albedo technique vs edge size.

TABLE I

Values of the Effective Multiplication Factor, k_{eff} , and Neutron Age to Thermal Energy, τ , for Uranyl Nitrate Assemblies

B_0^2 (cm^{-2})	α_0 (sec^{-1})	k_{eff}			τ (cm^2)		
		Exp	Two-Group	Age-Model	Fundamental	1st Harmonic	GAMTEC-II
0.00967 ^a	5787 $\pm$ 15 ^c	0.640 $\pm$ 0.01	0.6319	0.5677	53.4	45.4	54.5
0.0137 ^b	7433 $\pm$ 23 ^c	0.547 $\pm$ 0.02	0.5389	0.4435	53.6	69.5	54.5
				Averages	53.5	57.5	54.5

^aIncludes 3.4% correction for presence of a thin polyethylene reflector.

^bIncludes 7% correction for presence of a thin polyethylene reflector.

^cStandard deviation produced by computer least-squares program.

21 $\times$ 14 in. Each assembly was faced externally with sheet cadmium. The neutron pulses were produced by the Kaman Nuclear A-800 generator in 3- to 5- μ sec bursts at 2/sec, or 4/sec. The neutrons were detected by a 96%, ¹⁰B enriched BF₃ detector, $\frac{5}{8}$ -in. diam, and 9-in. sensitive length. The data, recorded by a TMC 256-channel analyzer with a modified pulsed-neutron logic module, were corrected for counting losses and fitted by a computer program allowing a three-parameter, single exponential, or five-parameter, double exponential least-squares options.

Certain harmonics were suppressed by positioning the neutron source, some were eliminated by aligning the detector axis with a modal line, and others selectively extracted by the double-exponential least-squares analysis. Independence of the decay constant of detector position, and of the initial channel on which the fitting commenced, were also used as criteria for establishing a prompt fundamental decay mode.

The prompt fundamental mode decay constant α is related to k_{eff} by the formula well known in pulsed-neutron literature, or readily deduced,¹

$$k_{eff} (1 - \beta) = 1 - \alpha / [v \Sigma_a (1 + L^2 B_0^2)]^{-1}$$

Values of the absorption rate, $v \Sigma_a = 14798 \text{ sec}^{-1}$, and diffusion area, $L^2 = 6.214 \text{ cm}^2$, were taken from a GAMTEC-II calculation made for the same material in another study in which $k_{\infty} = 1.0294$ was also measured.² If the delayed-neutron fraction is assumed to be $\beta = 0.0075$, then the values of k_{eff} obtained are as presented in Table I, where theoretical values deduced from the two-group diffusion kernel and age models are included for comparison.

If a slowing-down model is specified, the neutron age to thermal energy may also be deduced from the measured α . The two-group diffusion kernel model gives

$$\tau = B_{l,m,n}^{-2} \left[\frac{k_{\infty} (1 - \beta)}{(1 + L^2 B_0^2) - \alpha_{l,m,n} / \Sigma_a v} - 1 \right],$$

where l, m, n designates a particular mode. Thus, the age can be deduced from higher mode decay constants as well as from the fundamental, but with less reliability, as the values in Table I presented for both the fundamental mode and the 2,1,1 mode would suggest.

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Studies of Surface Density Spacing Criteria Using KENO Calculations, R. L. Stevenson, R. H. Odegarden (USAEC)

H. C. Paxton has noted an empirical relationship between the critical volume per unit base area, or surface density, of an air-spaced plane array of discrete units of fissile material and the critical thickness of a uniform slab of the same material.¹ It has been proposed that this relationship be used to establish safe spacing criteria by limiting stored units to some maximum "fraction critical" and the surface density to some fraction of the critical, uniform slab thickness.² The fraction critical is defined as the ratio of the mass of a single unit to the bare critical mass of the same fissile material in a similar shape. In a more recent study,³ Gutman noted the complex behavior of arrays of metal units but suggested less conservative criteria than Paxton, for application to solution systems.

Calculations were made using the KENO Monte Carlo Code⁴ and Hansen-Roach cross sections⁵ to test the validity of the proposed safe array criteria and determine significant variables for infinitely large, water-reflected plane arrays of cylinders in square lattices. The calculations investigated the effects of ²³⁵U density, fraction critical, cylinder radius, cylinder height, reflector proximity, and reduction from an infinite plane to a large finite array. The KENO results indicate that the critical surface densities are more sensitive to the parameters defining the stored units than hitherto recognized and, even after some allowance for finite array size, suggest that the proposed limiting criteria of 0.3 fraction critical and surface density at one-half the reflected slab thickness may not suffice at ²³⁵U concentrations above 500 g/liter.

Figure 1 illustrates the marked effect of the density of the fissile material in the stored units on the ratio of the surface density (t_a) of the array to the critical thickness (t_c) of the reflected slab for units at 0.3 fraction critical. Other calculations, not illustrated here, for tall cylinders of enriched uranyl fluoride solution with fraction critical values ranging from 0.035 to 0.8, show that the surface density does not approach the critical, reflected slab thickness until the fraction critical falls below 0.1.

Fig. 1. Ratio of surface density to critical slab thickness as function of ^{235}U concentration in fissile units for water-reflected planar arrays of cylinders.

The calculations indicate that, regardless of concentration, units at no greater than 0.3 fraction critical should be safe when uniformly spaced to maintain the loading below $200 \text{ g } ^{235}\text{U}/\text{ft}^2$.

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Effects of Interspersed Water Moderation on Subcritical Arrays of 3.5 wt% Pellet Containers, R. E. Kropp (UNC-New Haven), P. Lemberg, J. R. Tomonto (UCC-Elmsford)

Computer calculations were performed for subcritical arrays of containers containing 3.5 wt% UO_2 pellets as a portion of a fissile-material shipping container evaluation.

The containers consist of an 18-gauge (0.075-in.) steel cylindrical outer container surrounding a welded 14-gauge (0.098-in.) steel cylindrical inner container. The inner and outer containers are separated by wood laminate supports and thermal insulation. The UO_2 pellets are held within the inner container in a rolled, grooved, expanded polyurethane blanket which is inserted into a thin fiberpak. Three fiberpaks are placed in each container.

Structural tests have demonstrated that the inner container retains its integrity during standard drop tests so that the UO_2 pellets are contained in the original 8.5-in.-diam inner cylinder. This is a subcritical diameter¹ for enrichments up to and including 3.5 wt% UO_2 at optimum water moderation and complete water reflection.

Reactivity levels for arrays of infinitely long cylinders were calculated using the DTF² code as a function of moderation between the inner and outer containers and as a function of the amount of water external to the containers. Reflector boundary conditions were applied at the boundary between containers in all cases. The 16-group Hansen-Roach³ cross-section set was used with the ^{238}U cross sections modified to agree with Hellstrands resonance integral correlation.⁴ Thermal self-shielding effects were conservatively neglected. Linear anisotropic hydrogen scattering was accounted for in the DS₄ approximation to the transport equation.

Figure 1 indicates that k_{∞} decreases with the addition of interspersed moderation for internal H/U ratios ≥ 1.4 . For internal H/U ratios < 1.4 , the presence of interspersed water moderation increases k_{∞} but the value never becomes ≥ 1.0 .

These effects occur because an internal H/U ratio of 1.4 represents the optimum moderation for arrays of the fuel-water mixture within the containers with no container walls or interspersed moderation. The addition of interspersed moderation to the system when internal H/U ≥ 1.4 results in increased water and steel absorption and k_{∞} decreases. The addition of interspersed moderation with H/U < 1.4 results in an initial increase in k_{∞} until optimum moderation (H/U = 1.4) is reached and increased moderation beyond this value results in a decrease in k_{∞} as before. The k_{∞} with interspersed moderation cannot be greater than k_{∞} at the optimum H/U ratio because of differences in region importance.

Figure 1 also shows the effect of placing the interspersed moderator adjacent to the inner container as well as between containers. This increases k_{∞} slightly; however, k_{∞} of the array does not exceed that at the optimum internal H/U ratio.

Fig. 1. k_{∞} of containers as a function of H/U atom ratio within the fuel region, water moderator thickness and location.

The maximum k_{∞} for these pellets at optimum moderation was between 1.2 and 1.45 depending on the pellet diameter.

These results extend the recent calculational results of Thomas³ on arrays of 93.2 wt% enrichment uranium cylinders. The conclusion reached by Thomas, that as little as 0.125 in. of steel surrounding the individual units of a reflected array is sufficient to maintain subcriticality independent of the amount and distribution of added interspersed hydrogenous moderator, appears valid for subcritical arrays of 3.5 wt% UO_2 pellets as well.

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Evaluation of the Criticality Problem in Irradiated Fuel Shipment, J. D. Rollins,* J. L. Ridihalgh (BCL)

Two important considerations in the design of shipping containers for irradiated fuel are the control of nuclear criticality and the effective dissipation of decay heat (particularly, under hypothetical accident conditions¹). Upon close examination it is seen that these two facets of cask design are not altogether unrelated. Methods of accomplishing heat dissipation in recent container designs include the use of a fuel basket material, viz., copper, which has a high thermal conductivity. In addition, the copper, because of its strategic location within the shipping cask, plays a secondary role in reducing k_{eff} of the fuel element-cask arrangement. However, the latter effect of this material has not been previously evaluated in any detail.

A series of calculations was performed with the KENO program² on lead-reflected, water-moderated rectangular arrays of second-generation PWR fuel assemblies.³ The neutron cross sections used in the calculations were the Hansen-Roach 16-energy-group set.⁴ An assessment of the potential "in-cask" criticality problem was made by first assuming that cold, clean fuel assemblies were separated by varying dimensions of water only (i.e., no fuel basket) within the lead-shielded cask. The results of these calculations (Table I) indicate that PWR fuel element arrays, consisting of four or more elements (assemblies) nominally spaced, warrant consideration of the fuel basket material with respect to the control of nuclear criticality.

Additional calculations were performed for the 3×3 fuel array wherein the interassembly spaces were assumed to contain copper, cadmium, and copper-cadmium alloy, respectively. The separate cases were calculated to determine the nuclear poisoning effect of copper relative to that of cadmium, which is commonly added to control criticality. In addition, it was desired to determine the effect on reactivity of increasing the

TABLE I

Calculated k_{eff} Values for PWR Fuel Arrays in Water*

Fuel Array ^a	Center-to-Center Spacing (in.)	Interassembly Spacing (in.)	k_{eff}
1 × 1	-	-	0.86 ± 0.02
2 × 1	9.5	0.5	0.92 ± 0.02
2 × 2	9.5	0.5	1.02 ± 0.02
3 × 2	9.5	0.5	1.12 ± 0.02
3 × 3	10.0	1.0	1.20 ± 0.015
3 × 3	10.5	1.5	1.15 ± 0.02

*Lead-reflected.

^aDescription of fuel assemblies making up the arrays: 3.2% enriched UO_2 rods, 0.422-in.-diam, clad (Zr) thickness of 0.0243-in., fuel rod pitch of 0.563, and fuel length of 144 in.

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TABLE II

Calculated k_{eff} Values for PWR Fuel Arrays in Baskets of Various Compositions*

Fuel Array	Basket Web Material	Web Thickness (in.)	k_{eff}
3 × 3	Cu	1.5	0.91 ± 0.016
3 × 3	0.1% Cd ^a	1.5	0.97 ± 0.02
3 × 3	0.2% Cd ^a	1.5	0.92 ± 0.016
3 × 3	0.5% Cd-Cu	1.5	0.91 ± 0.013
3 × 3	1% Cd-Cu	1.5	0.89 ± 0.016
3 × 3	2% Cd-Cu	1.5	0.86 ± 0.013
3 × 3	3% Cd-Cu	1.5	0.86 ± 0.011
3 × 3	Cu	1.0	0.958 ± 0.008
3 × 3	0.5% Cd-Cu	1.0	0.913 ± 0.011
1 × 1	Cu	-	0.70 ± 0.02

*Lead-reflected, water-moderated.

^aWeight percent based on 1.5-in. Cu web.

percentage of cadmium in the cadmium-copper alloy because of concomitant increases in material costs as well as corresponding decreases in heat dissipation capability. Data from these calculations indicate that copper (basket) webs in thicknesses ranging up to 1½ in. can significantly reduce the k_{eff} (~20%) of a 3 × 3 fuel array-cask assembly (Table II). In fact, the 1½-in. thickness of copper sufficiently poisons the fuel array that the addition of 0.5 wt% of cadmium to the copper has essentially no effect on k_{eff} . Further addition of cadmium results in a maximum reactivity decrease of 5.5% $\Delta k/k$ with a poisoning saturation level being reached at about 2.0 wt% of cadmium.

The case of the 1-in. copper webs (Table II) shows less effect than the previous case but does, however, further illustrate that the initial addition of poison, whether it be copper or cadmium, is much more effective than larger quantities added subsequently.

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A New Plutonium Button Storage Arrangement, K. R. Ridgway (At Richfield Hanford)

The largest known storage array of plutonium metal buttons has been assembled at Hanford. The Monte Carlo code GEM III,¹ was used to determine the safe array. However, potentially high neutron levels from multiplication and spontaneous fission actually limited the array size. Neutron counting during the array assembly verified the subcriticality of the array and defined the actual radiation level.

Prior to this study, the GEM III code has been well correlated with plutonium metal ingot array experiments.^{2,3} The bases for the array calculations were for the worst foreseeable case as follows: (1) 2.5-kg buttons;

Fig. 1.

(2) 3 wt% ^{240}Pu ; (3) maximum moderation (0.001 g/cm³ water to simulate 16 gpm fire fog and 4 water columns simulating men); and (4) an albedo of 0.78, compared to 0.75 proposed by the British⁴ to simulate the 9-in.-thick concrete enclosures. The results of the calculations for lengthwise button spacings of 12 and 9 in. are shown in the figure and least-squares fits of these points indicate reflected critical arrays of 3600 and 2200 buttons, respectively. The 9-in. spacing gave the maximum storage and still met the Atlantic Richfield Hanford Company criticality prevention criteria⁵ of one-half of a critical array.

The 24- × 11- × 9-ft vault arrangement was based upon button accessibility, maximum button storage with acceptable radiation levels, and minimum cost. The 1092 button array consists of three double rows running lengthwise with the following spacing: 24 in. between the double rows, 12 in. between buttons in the double rows, and lengthwise and vertical button spacing of 9 and 12 in., respectively. The accumulated buttons were of two types, 0.755 and 0.374 height-to-diameter ratio, with an average weight of 2.1 kg. Six percent of the buttons were under 2.0 kg and 18% were slightly over 2.2 kg. The other average parameters of the completed array were: 19.45 g/cm³ metal density, 11 wt% ^{240}Pu , 8.7 kg/ft² areal plutonium density, and 0.34 g/cm³ overall plutonium density of the vault (a 100% increase over existing Hanford vaults).

Neutron counts and radiation dose rates were taken during vault loading. Extrapolation of the inverse count-

rate plot in the figure indicates the critical array to be >2600 2.1-kg buttons. This is the same order of magnitude as the extrapolated calculations showing 2200 2.5-kg buttons critical. GEM III calculated a k_e of 0.848 ± 0.012 for the actual array using the average button weight and no moderation. This indicates low neutron multiplication and agrees with the button per count rate (N/CR) plot in the figure. The final vault dose rate was 550 mrem/h including 150 mR/h gamma which was considered acceptable because of infrequent vault access.

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Test Reactor Operations Experiment

Operation of an All-Gas-Core Critical Experiment, J. F. Kunze, L. S. Masson, G. D. Pincock (GE-Idaho), R. E. Hyland (NASA-Lewis)

The first static (i.e., non-flowing) all-gas-core critical experiment initially went critical on May 17, 1967, at the National Reactor Testing Station. The experiment was in support of the cavity reactor program conducted for NASA for the purpose of determining the nuclear feasibility of the gas-core rocket reactor¹ concept. The reactor consisted of a large D₂O reflector tank, outer dimensions 12 ft diam × 10 ft long, with an inner cavity 6 ft diam × 4 ft long. Placed inside the cavity was the core tank, 4 ft diam × 43 in. long.

Prior to this gas-core experiment, numerous cavity reactor configurations² had been made critical using a simulated gas core constructed of thin (0.001-in.) foils of uranium dispersed uniformly throughout the core volume. Significant fuel worth discrepancies (calculated vs experimental) and deviations from calculated critical mass had cast doubt on the validity of this simulation of a gas core, and lead to the performing of an actual gas-core experiment to evaluate the adequacy of the simulated cores. Operating conditions required 20 to 40 kg of UF₆ in the gas phase contained within the $1.26 \times 10^6 \text{ cm}^3$ core. This required a minimum temperature of 185°F in the core and corresponding pressures of 35 psia to ensure that all the fuel material was in the gaseous phase. The thin-walled Al core tank was heated with hot air, as shown in the figure.

Approach to criticality was made in the usual step-wise continuous fashion of incremental fuel additions. Start-up with a cold core was always quite smooth and free of reactivity fluctuations. However, the shutdown cooling process created some unusual reactivity spikes evidently caused by selective condensation of the UF₆ on the cooled walls of the tank. Noise analyses at operating conditions revealed a few small "resonances" in the Bode Plot in the 0.05- to 0.10-Hz range, probably caused by gas circulation inside the tank.

The gas-core data showed that the simulated cores constructed of foils had a critical mass bias of about 15%. These foil cores were then redesigned to a second mock-up that exhibited only about a 4% bias on critical mass. This experiment demonstrated not only how foil experiments could be appropriately planned and interpreted, but also how gas experiments can be more easily performed. A series of all-gas experiments in spherical geometry is now being conducted with greater operating efficiency than is possible with a foil core mockup.

The coaxial flow cavity reactor concept has been studied under cold flow, non-nuclear conditions.³ The flow pattern was seen to contain gas waves of substantial amplitude, and these have been mocked up in the critical experiment using foil fuel.⁴ Waves of large amplitude created reactivity perturbations of as much as 3%Δk. However, control systems of the drum or sleeve type were measured and found to provide as much as 10 and 20%Δk, respectively, in reactivity swing worth. This, plus the long (3 msec) prompt-neutron lifetime makes the cavity reactor an easily controllable system despite the potentially large worth of waves.

Fig. 1. Cavity reactor showing fuel waves on core boundary.

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Fast Reactor Experience and Fast Flux Technology

ZPPR Start-Up and Operating Experience, R. G. Matlock, P. I. Amundson, R. E. Kaiser, J. C. Young, R. O. Vosburgh, J. M. Gasidlo, R. J. Forrester (ANL-Idaho)

The Zero Power Plutonium Reactor (ZPPR) is a split-table-type experimental reactor operated by Argonne National Laboratory at the National Reactor Testing Station in Idaho.

ZPPR is designed to be used for physics studies of power breeder reactor systems in support of the AEC's LMFBR program, and can accommodate mockups characteristic of the proposed 300- to 500-MW(e) demonstration plants and 1000-MW(e) commercial plants. Composition simulation is accomplished by assembling fuel, coolant, and structural materials, in the form of small plates, into drawers which are, in turn, loaded into the reactor matrix.

Uncertainties in currently available basic cross-section data and in calculational techniques require that experiments be performed on mockups of this size to provide checkpoints for theoretical predictions of fast reactor parameters. ZPPR will be used to investigate, among

other things, Doppler, temperature, and sodium void reactivity coefficients for such systems and will provide experimental data for use in evaluation of calculational techniques. In particular, experimental data will permit an evaluation of the usefulness of zoned reactor measurements by comparing results obtained in zones with those of full size mockups.

ZPPR is the largest fast critical assembly in the United States and will permit the investigation of cores containing up to 3000 kg of plutonium. Facility construction began on August 8, 1966, the initial plutonium loading commenced March 27, 1969, and criticality was achieved on April 18, 1969 with a loading of 357 kg ^{239}Pu + ^{241}Pu . The approach-to-criticality was performed by completely loading the configuration as calculated, with the plutonium fuel replaced by depleted uranium. Starting from the core center, plutonium fuel was then incrementally substituted for the depleted uranium, the amount being determined from assessment of the subcritical multiplication data.

As of July 1, 1969, ZPPR had been started up 120 times with about 65 different loading configurations. The largest critical mass was 530 kg ^{239}Pu + ^{241}Pu . The initial assembly being studied is in support of the Fast Test Reactor Program and consists of a cylindrical plutonium-

Fig. 1. Reactivity vs time for a poison rod calibration in ZPPR Assembly 1.

fueled core surrounded axially and radially by sodium-nickel reflectors. The fuel utilized in this experiment is a ternary alloy clad in 0.015-in. stainless steel and comprised of 69.3 wt% depleted uranium, 28.2 wt% plutonium containing 11.5% ^{240}Pu , and 2.5 wt% molybdenum.

A large fraction of the physics data obtained from a critical facility requires that precision reactivity measurements be made. In addition, operational safety checks require that prompt shutdown reactivity parameters be measured frequently. Consequently, considerable effort has been directed toward developing techniques for rapid and accurate reactivity measurements and calibrations.

In a plutonium-fueled fast critical assembly, all types of reactivity measurements are affected by the presence of a ^{240}Pu spontaneous fission source. The necessity for low-power operation in critical facilities means that the source will almost always be of significant reactivity worth.

Calibration of the source term is accomplished by inverse kinetics analysis of count rate as a function of time when the reactor is on a positive period of about 100 sec.

Control- and safety-rod calibrations are performed by removing reactivity slowly, starting from a steady power level. The reactivity history is calculated using inverse kinetics, and is of the form shown in Fig. 1. The total rod worth is then determined from the average values immediately preceding and immediately following rod motion.

The prompt shutdown parameters of primary interest are the reactivity removal rates and total reactivity worths of safety-related devices.

Time-dependent neutron population data for these reactivity measurements are obtained from high sensitivity ion chambers located outside the reactor core. This information is multiscaled into an on-line computer at a rate of up to 100 cps, and is used to determine reactivity as a function of time during the course of a power transient. This technique allows a complete safety-rod scram-time and worth measurement to be made within one minute as compared to several hours by other methods.

Mathematical Methods of Transport Theory

Monte Carlo Eigenvalue Calculations of Fast Critical Assemblies Using VIM-1, *Leo B. Levitt, R. C. Lewis (AI)*

Eigenvalue calculations of fast critical assemblies such as ZPR-3-48 and ZPR-3-11 have been made using the newly developed VIM-1 code. VIM-1 has been designed expressly for plate-drawer-type criticals such as those mentioned above, directly using detailed point cross-section libraries such as ENDF/B, completely contained in the high-speed computer core with each isotope having its own energy grid and interpolation scheme.

Benchmark calculations of GODIVA and JEZEBEL have been made in order to compare results with more conventional multigroup Monte Carlo and S_N calculations reported in the literature. As can be seen in Table I, these calculations are in agreement with the consensus of results obtained using other codes with ostensibly the same ENDF/B cross-section sets.

VIM-1 uses detailed angular distributions, prepared in the form of cumulative distribution functions of CM scattering angle at energy grids tailored to the needs of each isotope, and uses a discrete level treatment of

inelastic scattering wherever data are available. Distinctive features of VIM-1 include a simplified procedure for generating the spatial distribution of source neutrons, a new procedure for Monte Carlo treatment of unresolved resonances, and an optimized data packing scheme which permits complete point cross-section data sets for a large number of isotopes to be stored simultaneously in the high-speed computer core memory. These sets contain energy grids running from 10 eV to 10 MeV.

Calculations reported in this abstract used analog Monte Carlo with an absorption-type estimator, although alternatively VIM permits the use of a variable amount of absorption weighting at each collision.

Table I contains a brief summary of eigenvalue calculations made with VIM-1. They should be regarded as preliminary since they were not made with the latest ENDF/B data. They do, however, indicate that a non-multigroup, fast critical assembly Monte Carlo code can now be used as a tool for analyzing fast critical experiments and that benchmark calculations can now be performed at moderate cost using standardized sets of heterogeneous input data.

TABLE I

Assembly	No. Histories	Maxwellian Fission Temperature	k_{eff}	Leakage	Computer Running Time (CDC 6600)
GODIVA	50,000	1.28 MeV	1.012 ± 0.004	0.5768	240 sec
JEZEBEL	50,000	1.41 MeV	1.015 ± 0.004	0.6661	184 sec
ZPR-3-48 (Heterogeneous Mockup)	10,000	1.374 MeV	0.998 ± 0.009	0.0340	1110 sec
ZPR-3-11 (Spherical Mockup)	19,600	1.28 MeV	1.010 ± 0.005	0.0417	775 sec

Thermal Reactor Physics

MTR-Phoenix Fuel Critical Experiments and Analysis, J. W. Kutcher, E. C. Davis, C. M. Heeb (PNL)

The MTR-Phoenix Fuel Experiment^{1,2} is a unique pure plutonium-fueled depletion experiment supported by a comprehensive set of critical experiments. Phoenix fuel³ is a mixture of plutonium isotopes with a relatively high content of ²⁴⁰Pu. During irradiation, this fertile ²⁴⁰Pu is converted to fissionable ²⁴¹Pu and partially compensates for the loss of fissionable ²³⁹Pu. The MTR experiment is designed to demonstrate the longer reactivity lifetime characteristics of this concept.

The MTR-Phoenix Core is a 3 × 9 rectangular array of 19 fixed fuel elements (16 fuel plates each) and 8 fuel-follower (13 plates each) shim elements. Each plate is an Al-21 wt% Pu alloy which is 0.040 in. thick and is clad in 0.020-in. aluminum. The total plutonium content of the core is 9.80 kg, and the isotopic weight percentages³ are 0.5/65.3/23.2/7.6/3.4 for ²³⁸Pu/²³⁹Pu/²⁴⁰Pu/²⁴¹Pu/²⁴²Pu, respectively.

In preparation for the experiment, extensive calculations⁴ and selected critical facility measurements⁵ were made. The basic analytical approach is the multi-group method in the S_n and diffusion theory approximations to the transport equation. The fast and epithermal spectra used in computing the multigroup cross sections are calculated using HRG.⁶ The thermal spectra are calculated using Battelle Revised THERMOS.^{7,8} The Nelkn kernel is used to describe the thermal-neutron scattering in water.

The nuclear data is the 1965 Westcott evaluation⁹ for ²³⁹Pu and ²⁴¹Pu and the 1959 Leonard¹⁰ measurements of ²⁴⁰Pu cross sections. An empirical adjustment to the 1-eV resonance absorption based on measurements⁵ of k_{∞} in plutonium slab cells was used. This "Wings-240" treatment amounts to an addition of 660 b to the 1-eV ²⁴⁰Pu resonance for the 40-mil absorber plate thickness used in the MTR-Phoenix core. Using this technique, the critical approach¹¹ experiments with 8% ²⁴⁰Pu and the engineering mockup critical experiment^{4,5} with 19% ²⁴⁰Pu yield calculated values of k_{eff} within 1.0%.

A two-dimensional diffusion theory code in four energy groups¹² was used to predict the critical loadings. These critical predictions were documented previous to the experiment. When the critical loading was achieved it was with the predicted loading of 7 fixed fuel elements, 4 fuel-follower shims, and 4 shims that had beryllium followers. The shim-free k_{eff} as deduced from measured rod bank heights and reactivity calibrations are 1.001 ± 0.001 and 1.103 ± 0.008 for the initial critical and fully loaded cores, respectively. The corresponding calculated k_{eff} are 1.016 and 1.107. Taking into account the effects of the decay of ²⁴¹Pu into ²⁴¹Am during the time between the isotopic analysis and the experiment, reduces the calculated k_{eff} for the initial critical from 1.016 to 1.010.

The reactivity worth of each of the fuel-follower shims was determined from period measurements. The total rod worth was 274 mk, with interior shims worth 60% more than exterior shims. The value of $\beta/1$ -effective neutron delayed fraction divided by prompt-neutron lifetime—was determined from reactor noise data to be $59.03 \pm 0.69 \text{ sec}^{-1}$.

The power distribution in the Phoenix Core was determined from fission product activity traverses of ²³⁵U pins, Pu wands, and removable fuel plates. The peak power, used to determine maximum operating power, was found to occur in the fuel-follower section of an interior shim element, about an inch below the bottom of the fixed fuel core. The ratio of this peak to the core average was 4.3 ± 0.5 .

The MTR-Phoenix Fuel Experiment is the first burnup experiment for a reactor core fueled entirely with plutonium. Thus, its reactivity time behavior with irradiation and its isotopic distributions will provide yet a more stringent test of methods used to predict plutonium reactors. The complete set of experimental results including especially the depletion portion should be extremely important to those concerned with calculating effects of even partial plutonium cores.

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Fast Neutron Spectra—I

Testing of Heterogeneity Methods Used in Fast Reactor Critical Analyses by Comparison with Theoretical (GEDANKEN) Experiments, R. G. Palmer, J. P. Plummer, R. B. Nicholson (ANL-Idaho)

The effect of heterogeneity in the ANL plate-type fast criticals (ZPPR, ZPR-3, etc.) is a complex problem for which only approximate calculational methods are currently available. Comparison of measured and calculated quantities do not offer conclusive tests of such methods due to the additional uncertainties in cross-section data. One approach to methods testing is to calculate the results of theoretical experiments (GEDANKEN models) that are complex enough to exhibit significant heterogeneity effects but which are simple enough to be calculated exactly by available techniques.

Since there appears to be no clear-cut answer on theoretical grounds as to which is the best weighting procedure when spatially homogenizing cross sections across a heterogeneous cell, the object of this study is to compare various weighting methods on calculations of k_{eff} , sodium voiding, and central reactivity worths using GEDANKEN "exact" results as the basis.

The first GEDANKEN consisted of a ^{235}U -reflected infinite slab core 11 cells across, each cell consisting of one 0.5-cm plate of Pu-U (U:Pu = 3.67:1), two 1-cm plates of sodium, and two 0.5-cm plates of graphite. Two 12-group cross-section sets were prepared using MC², one having isotropic scattering and the other having P_1 anisotropy. The exact k_{eff} of the system was calculated using the Double- S_N Code TESS¹ by extrapolating to S_∞ .

The results of bivariate,² real-flux, and volume-weighting procedures for normal, half sodium-voided and fully sodium-voided cores are shown in Table I where they are compared to the exact values. The fluxes for the weighting procedures were obtained from TESS calculations on the unit cells with added pseudo-absorption D_0B_g cross sections. Also shown in Table I are the calculated worths of thin central slabs of ^{239}Pu , ^{235}U , and ^{12}C .

TABLE I
Results of Various Weighting Procedures
on Experiments in GEDANKEN 1

	Exact	Real Weighting	Bivariate Weighting	Volume Weighting
Heterogeneity Factor, $\Delta k/k$				
Normal code	0.00634	0.00614	0.00554	---
Half sodium-voided core	0.00518	0.00501	0.00443	---
Fully sodium-voided core	0.00391	0.00370	0.00334	---
Sodium-Void Effects, Δk				
Half sodium-voided	-0.00159	-0.00156	-0.00154	-0.00042
Fully sodium-voided	-0.00236	-0.00238	-0.00213	+0.00008
Central Sample Worth, Δk				
0.01 cm ^{239}Pu	0.00567	0.00570	0.00556	0.00563
0.10 cm ^{235}U	-0.00665	-0.00679	-0.00657	-0.00654
0.25 cm ^{12}C	-0.00183	-0.00182	-0.00159	-0.00161

Relative to the exact values, the errors in the real-flux-weighted results are 5% or less. The corresponding bivariate results are somewhat worse and the volume-weighted results for the sodium void effect are poor. The central sample worths appear to be fairly insensitive to the treatment of the environment. For this GEDANKEN model there appears to be no advantage in using bivariate weighting. The exact heterogeneity factor using the anisotropic cross-section set was 0.00664, hence anisotropy does not appear to be a significant factor but further checks on this are being made using a 22-group set.

Other GEDANKEN cores are now being studied; in particular, a two-zoned-core simulation of a power-flattened reactor in infinite slab geometry. A cylindrical 2D core with radial plane plate structure identical to GEDANKEN 1 is being calculated to study the effect of leakage parallel to the plates.

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Effect of Hydrogen Content of Graphite in Fast Critical Experiments, Philip F. Palmedo, Hans Ludewig, Arnold L. Aronson (BNL)

Many of the clean fast reactor critical experiments that are currently being used to test cross-section sets and analytical techniques include graphite in the core. Only in rare cases is there an indication given that the hydrogen content of the graphite has been considered in the analysis of these experiments. For most US experiments, the hydrogen content of the graphite is not even reported. Although usual amounts of hydrogen do not have a significant effect on the criticality of such systems, other measurable parameters may be appreciably affected.

For example, Fig. 1 shows the distribution of the ^{235}U fission rate measured in a cell composed of ^{235}U , stainless steel, and graphite.¹ The theoretical distributions were calculated using the DSN Code ANISN, and cross sections based on the ENDF/B evaluation.² In one case no hydrogen was considered, and in the other, the measured 300 ppm by weight was introduced into the graphite. The net change in the ^{235}U fission rate at the center of the fuel to the center of the graphite was 4%.

To determine the effect of hydrogen on other parameters, ZPR-3 Assembly 48³ was analyzed with and without H. The amount of hydrogen was arbitrarily set equal to that found for the BNL graphite; i.e., 300 ppm. Separate cross-section sets were determined for each case using MC² and the cross-section data of ENDF/B. Two-region spherical calculations were performed using ANISN in the S_4 approximation to calculate the eigenvalues, fluxes, and adjoints. The central reactivity worths were calculated using these ϕ and ϕ^* in a first-order perturbation approach.

Fig. 1. ^{235}U fission distribution in cell.

TABLE I

Effect of Hydrogen on Parameters of ZPR-3 Assembly 48

Parameter	Calculated		Experiment ^a
	Without H	With H ^a	
k_{eff}	0.988	0.991	1.000 $\pm$ 0.004
Lifetime, $\times 10^{-6}$ sec	26.6	27.4	
Central Reaction Rate Ratios ^b			
^{235}U fission	0.0322	0.0318	0.0307 $\pm$ 0.0004
^{239}Pu fission	0.946	0.941	0.976 $\pm$ 0.010
^{235}U capture	0.270	0.274	
^{238}U capture	0.140	0.139	0.138 $\pm$ 0.004
^{10}B capture	1.26	1.28	
Central Reactivity Worths, relative ^c			
^{235}U	100	100	100
^{239}Pu	-7.94	-8.01	-7.44
^{238}U	1.36	1.36	1.34
Na	-0.110	-0.0932	
O	-0.137	-0.116	
^{10}B	-96.0	-99.8	-112.3
Ta	-31.4	-33.7	
C	-0.0620	-0.0321	

^a300 ppm by weight in graphite^bRelative to fission in ^{235}U ^cNormalized to ^{235}U worth, set = 100 in each case; ^{235}U worths differed by 0.4% between the H and no-H cases.

As shown in Table I, although some parameters are not appreciably altered by the presence of hydrogen, others, notably the worths of O, Ta, and C, are significantly affected.

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Reactor Physics for Plutonium-Fueled Fast Reactor Design—I

A Postanalytical Study of Eight ZPR-3 Benchmark Criticals Using ENDF/B Data, A. L. Hess, R. G. Palmer, J. M. Stevenson* (ANL-Idaho)

In the period between June 1966 and December 1969, the ZPR-3 critical program included a series of eight relatively simple plutonium-fueled assemblies, the principal objective of which was to provide a variety of clean fast integral data with which to test current and any future neutron cross-section compilations. Starting with the well-known Assembly 48, the series included 48B, 49, 50, 53, 54, 58, and 59. The core compositions of 53 and 54 were almost identical, the only difference between the assemblies being that 53 was depleted-uranium-reflected and 54 steel-reflected. Such was the case with 58 and 59 except that 59 was lead-reflected rather than steel.

This paper reports the results of a series of uniform path calculations on the assemblies and some of the important experiments using the ENDF/B-MC² route in the ANL ARC System.

The plate structure of ZPR-3 is such that even the simplest drawer loading is significantly more heterogeneous than the corresponding power reactor subassembly. Consequently, heterogeneity calculations and spatial homogenization of the cross sections over the cell for the subsequent whole reactor calculations played a major role in the study.

The cell-averaged cross sections were generated using an MC²-CALHET combination,¹ the MC² for the resonance region treatment and the collision probability code CALHET for the final overall flux advantage/disadvantage treatment. Since MC² employs the two-region equivalence principle, most of the core cells had to be divided into several two-region subcells to treat the different self-

shielding conditions for the different plate forms of the same resonance isotope. First-order corrections were made to the resonance absorber plate thicknesses to account for the finite size of the other two plate dimensions. Simple flux weighting was used in the final homogenization over the cell, the spatial flux distributions being derived by CALHET.

The resulting homogenized cross sections were then used in conventional one- and two-dimensional diffusion theory calculations on the as-built core dimensions to obtain the appropriate k_{eff} 's and flux distributions. One-dimensional and reduced-group two-dimensional transport theory calculations were performed to obtain extrapolated S_{∞} correction factors.

*Attached from UKAEA, Winfrith.

Calculations of sample reactivity worths were based on first-order perturbation theory with corrections for self-shielding, self-multiplication, and multiple-scattering effects in the sample. Subsidiary calculations were made in some of the cases to determine the effect of other weighting procedures, namely bilinear and normal volume weighting, used in the description of the core environment around the sample.

Since the necessary modifications to the ENDF/B-to-MC² processing codes to accommodate the recent changes in ENDF/B format were not initially available, the calculations were all based on the first (1968) ENDF/B compilation. To ascertain the trend of some of the ENDF/B revaluations on the results, certain calculations were repeated using some of the new data introduced in the coarse-group stage of the calculations.

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Reactor Physics for Plutonium-Fueled Fast Reactor Design—II

Statistically Significant Biases in the FTR Neutronics Model, R. A. Bennett, S. L. Engstrom, J. V. Nelson, W. R. Young (PNL)

Calculational models used in reactor design are biased with respect to the various neutronics characteristics; e.g., critical masses may be consistently overestimated. The biases are due to systematic errors in the model or the nuclear data which are difficult to identify. In fact, one is usually unable, during the course of the design of any single reactor, to make significant strides toward eliminating biases. One is, rather, faced with the problem of identifying what biases and uncertainties exist. The design requirements may then be appropriately adjusted. Results are presented here of a current assessment of biases and uncertainties in the FTR neutronics model.

These evaluations are based upon analyses of measurements made to date on criticals built by Argonne National Laboratory, as a part of the BNW FFTF Critical Experiments Program. All assemblies simulated PuO_2 - UO_2 -fueled, Na-cooled, Ni-Na-reflected or partially Ni-Na-reflected systems, and include ZPR-III Assemblies 48A,¹ 51,² 52,³ and 56B,⁴ and ZPPR-1.⁵ The analyses used multigroup diffusion theory in one and two dimensions. However, the diffusion theory values of the multiplication constants were corrected by +0.5 to +1.3% $\Delta k/k$ to values obtainable from S_{12} transport calculations and by +1% $\Delta k/k$ for platelet heterogeneity. The cross-section set used throughout is a modified version⁶ of the Bondarenko⁷ 26-group set, which was energy-averaged to obtain 5-, 8-, and 13-group sets.

Since the FTR will be fixed in size, the criticality design parameter is fissile enrichment. FTR design models overestimate the required enrichment. The projected bias, in terms of mass of fissile material is +5%, but, based upon analyses of four FTR criticals, this number is uncertain by $\pm 1\%$. Hence, the calculated enrichment is reduced 0.5% due to the bias and increased 1% to ensure the required excess reactivity.

* Biases in calculated values of power density distributions have been evaluated using measured spatial distributions of the fission rates in ^{239}Pu . Calculated distributions are in good agreement with measurements made through B_4C control rods. However, for FTR in its operating state, the majority of the control rods are withdrawn. In simulations of the latter condition, that is for measurements between or without control rods, the calculated radial peak-to-average power density ratio is on the average $9 \pm 1\%$ greater than measured. For power peaking along the central axis, the same ratio is $3 \pm 0.5\%$ greater than measured. These values are multiplicative; hence, the total power peaking is overestimated by $12 \pm 1\%$. Unless allowance is made for this bias, the FTR would have a lower central flux at a set power level or a higher power for a set central flux level.

Both in-core and peripheral poison rods are to be used in FTR for the primary and secondary control and safety systems. For B_4C control rods located on the central axis of the core, the calculated values deviate from the experimental values by $\pm 9\%$. For peripheral B_4C rods, the calculated values deviate from the measured worths by ± 2 to $\pm 4\%$. Since no statistically significant biases

have been found in the calculated worths of control rods, the design values need only reflect the calculational uncertainties.

Reactivity worths of a range of sample sizes and compositions were measured in this series of criticals. For ^{239}Pu , the statistically significant bias in the calculated value is +30%, independent of sample size in the range of a few grams to as much as 32 kg. For ^{235}U , the ratio of calculated-to-experimental value is ~ 1.5 for small samples. Analyses of a series of sodium void experiments performed in ZPR-III, Assembly 51, yield ratios of calculated-to-experimental values for the reactivity effect of voiding over a $\frac{1}{2}$, $\frac{1}{4}$, and full core that are ~ 0.95 .

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Joint Experiments on MASURCA, ERMINE, and SNEAK in Support of the Fast Breeder Projects PHENIX and SNR, C. Clouet d'Orval (CEN-Cadarache), P. Engelmam, F. Helm, G. Jourdan (KFZ-Karlsruhe), A. Meyer-Heine (CEN-Cadarache), S. Pilate (Belgonucléaire), A. P. Schmitt, O. Tretiakoff (CEN-Cadarache), R. Vidal (CEN-Fontenay-aux-Roses), G. Wittek (KFZ-Karlsruhe)

It was decided between the centers of Cadarache and Karlsruhe to perform the critical experiments in support of the PHENIX and SNR projects as a joint program.

This cooperation, designed to make optimal use of the existing facilities, comprises

1. an exchange of reactor materials, both fissionable and nonfissionable, thereby removing some of the restrictions otherwise imposed on the experimental programs by the material inventory available. In particular, 90 kg of plutonium from MASURCA have been in use at SNEAK over the past year.

2. a careful intercomparison and adaptation of experimental techniques. This includes the exchange of devices and methods and the actual performing of measurements by members of the SNEAK group at French facilities and vice versa.

3. the evaluation of the measured results at both Cadarache and Karlsruhe and a joint investigation of any discrepancies found between different calculations as well as between calculations and measurements.

A perfect simulation of the PHENIX or SNR core being difficult to analyze, it was decided to carry out the present experiments on clean media similar to the PHENIX core zones 1 and 2 and on uranium media with corresponding enrichments. For easy reference, these media shall be called Pu 1, U1 (low enrichment, internal core zone) Pu 2, U 2 (high enrichment, external core zone).

These core compositions have been used systematically in two large classes of experiments:

1. Measurement of important integral parameters such as material buckling (by flux mapping and substitutions), reactivity worths and spectral indexes, made essentially in MASURCA.
2. Experiments having a more technological aspect such as power-traverses, Na-void and control rod studies made essentially in SNEAK.

Up to now, the following configurations have been studied in MASURCA: Pu 2 in the central zone of a Pu 2/U 2 zone loading (60-cm core height), U 2 in single-zone cores of first 60-cm, then 90-cm height, and U 1 in both a single-zone core (90-cm height) and a zoned loading.

These experiments include some basic measurements as well as central control rods and Na-void using zoned cores.

In SNEAK-2 (core height 60 cm), two concentric enrichment zones were first loaded entirely with U 1 and U 2; later a 90° sector containing Pu 1 and Pu 2 was introduced. The size of this sector was increased by surrounding it with a MASURCA-Pu-fueled buffer, to establish a better equilibrium spectrum in the experimental region. Since this core had a radial structure similar to the prototypes, the main interest was in the radial dependence of the parameters measured.

Axial measurements are being performed in SNEAK-6 which is 90 cm high and has a cylindrical central zone of the composition Pu 1 fueled first with SNEAK platelets, then with MASURCA rodlets.

In the fast-thermal facility ERMINE, complementary measurements are performed using both SNEAK and MASURCA materials.

The exchange of calculations showed that criticality is predicted by both Cadarache and Karlsruhe with an accuracy of 1 to 2%. In the case of Na-void calculations, discrepancies by as much as a factor of 2 were found between the results of the two laboratories. The reasons for these are now being investigated by a thorough comparison of the calculations and the underlying data.

Detailed calculations on the control rod experiments are being performed at Karlsruhe using flux synthesis calculations in XY/Z geometry. The results show that the reactivity of the absorbers used (Ta and B₄C) is generally overestimated by 5 to 15%.

After conclusion of the series of experiments in support of SNR and PHENIX it is planned to continue the cooperation in future work to be performed on our critical facilities.

Data Acquisition and Analysis for Fast Reactor Physics in JAERI, J. Hirota, S. Katsuragi, M. Hirata (JAERI)

The JAERI's integrated program of data acquisition and analysis for fast reactor physics consists of several sub-programs which are evaluation of nuclear data, processing of group constants, development and improvement of computing codes, experiment and analysis, and adjustment of group constants. The Japanese nuclear data committee is responsible for evaluation of nuclear data.

Processing of group constants is divided into two major parts. The first part is the study and production of group constants for heavy nuclides in the resonance region. Many useful computer programs have been developed. Group constants consistent with low resolution measurements, taking interference effects into account, were obtained. The second part is the production of group constants for heavy nuclides in the non-resonance region, and for light and medium nuclides. These constants were produced, under better interpretation of nuclear data, using a processing program based on the ENDF/A format. Compiling these group constants, a 70-group-constants set (JAERI-FAST set) was prepared. The 25-group ABBN-type set was also produced from the set. The set was used for calculations on a large fast reactor, in an international comparison suggested by R. D. Smith. Extensive evaluation and improvement of the JAERI-FAST set are now in progress.

Basic computer codes such as one- and two-dimensional diffusion, S_N and P_1 codes are being revised to use the JAERI-FAST set. Codes for calculating a fine spectrum and an ultra-fine spectrum in homogeneous media have been developed. Preparation of spectrum calculation codes for heterogeneous media is in progress. Development of a versatile computer code system is being carried out to meet various requirements for reactor analysis.

The JAERI Fast Critical Facility (FCA) of horizontal split-table type went critical for the first time in April 1967. Since then, series of critical experiments and analyses were undertaken on 20% enriched uranium metal (FCA-I), intermediate energy (FCA-II), diluted (FCA-III), and zoned (FCA-IV) assemblies. The zoned assemblies had a unit k-infinity test region in which sodium-void and Doppler experiments were carried out. The replacement experiment was also performed to obtain information on the capture cross sections of iron, nickel, stainless steel, and molybdenum.

Modifications to FCA necessary for the use of plutonium were completed at the beginning of this year. To justify the group-constants set and calculational method used in the Japan Experimental Fast Reactor (JEFR) design, a series of mockup experiments are scheduled. The first mockup experiment is being made on Assembly V-1 which has a core spectrum similar to the JEFR.

Following the V-1 physics mockup, Assembly V-2 will be constructed as an engineering mockup. Using reactor-grade plutonium, effects of plutonium higher isotopes will be examined by the end of this year. In addition, simulated fission products will be used to study the burnup effect. In 1972, a partial mockup experiment for the Japan Prototype Fast Reactor (JPFR) will be started

using zone and sector loading, especially for obtaining information on the safety aspect of the reactor.

To improve the procedure for group-constant adjustment by integral data, reliability of the adjusted group constant has been examined. A rule to check a systematic error in reference cross sections was introduced, taking notice of the standard deviation of the adjusted group constant. In addition, two types of correlation are introduced. One is based on the energy dependence of the cross section and the other is based on the relative measurement of the cross sections. A numerical run is now being made using some integral data of the benchmark and FCA experiments.

The Demonstration Reactor Benchmark Critical Assemblies Program, C. E. Till, W. G. Davey (ANL)

A program of critical assemblies to provide the experimental base for design of LMFBR demonstration reactors has been developed and is currently underway at Argonne National Laboratory. Initially, individual meetings were held between the ANL staff involved in the planning of the critical assemblies programs and representatives of each of the three industrial concerns involved in the demonstration plant Project Definition Phase. At these meetings, each firm outlined its current demonstration reactor design, the information it felt is required from the critical experiments, and its current views of priorities and time-scales. This information was then used to define a program of benchmark critical experiments aimed at satisfying the stated requirements of each firm as completely as possible.

The resulting Demonstration Reactor Benchmark program will occupy the two facilities, ZPPR and ZPR-6, for approximately the next two years. The assemblies and experimental programs on each are tied together to give a complete package of information in the appropriate time span. In outline, the ZPPR assemblies will be two-zone demonstration reactor-like systems, while the ZPR-6 assemblies provide specialized support measurements on identical or closely related compositions. The first ZPPR assembly is a clean two-zone core, and the second will be a related control-poisoned system. The main ZPR-6 program will concentrate on temperature and heterogeneity-related studies on the compositions of the ZPPR assemblies, using the Variable Temperature Zone (VTRZ). Prior to the VTRZ work, ZPR-6 will be occupied first with a single-zone uniform core of the central zone composition of ZPPR. This provides (a) a system in which the material properties of the ZPPR core can be

established without the complexities and possible ambiguities associated with a two-zone core; (b) the normalization for the VTRZ studies to follow; and (c) a well-established data-testing benchmark that is in the range of large LMFBR compositions. This assembly will be followed by small-zone configurations leading into the VTRZ program.

ZPPR Assembly 2 is the first assembly in the overall program. It is a clean version of a generalized demonstration reactor configuration. It is a two-zone (plutonium, uranium), O_2 -fueled core, with a 50/50 volume split between the two zones, and a ratio of plutonium contents of 1.5 between the outer and inner zones. The axial and radial blankets have appropriate volume fractions of depleted UO_2 and sodium, and are surrounded, in turn, by a steel reflector.

The inner and outer core zones have one- and two-drawer unit cells, respectively, for reasonable simplicity in analysis. Further, and very importantly, the core compositions can be precisely matched by the rodde-fuel, sodium-filled calandria combinations currently being procured for pin-plate geometry comparison in the ANL criticals. A substantial portion of the program involves the use of the pins and calandria, both as direct substitution in the normal plate core as well as in auxiliary studies on ZPR-6. Finally, plate-fuel with higher plutonium isotopic contents more similar to proposed demonstration plant fuels, but identical in fissile content to the normal critical assembly fuel, is on hand, and will be used similarly to the pin-calandria combinations.

The dimensions of Assembly 2 are given in Table I, as calculated using MC²-generated ENDF/B cross sections and two-dimensional diffusion theory. Initial criticality has just been achieved, and the critical mass, uncorrected for various minor effects, is 1066 ± 5 kg fissile ($^{239}\text{Pu} + ^{241}\text{Pu} + ^{235}\text{U}$).

TABLE I
Calculated Values for ZPPR
Assembly 2

Height, in.	36
Diameter, in.	74.4
Volume, liters	2570
Mass plutonium total, kg	1250
Mass fissile, kg	1116.4

Nuclear Data for Reactor Design—II

Effects of Fission Spectra on the Calculated Nuclear Parameters in the FCX Reactor, S. Salah (W-Astro)

Effects of fission spectra on the calculations of basic nuclear parameters were investigated for the FCX (Flexible Critical Experiment) reactor.¹ This investigation was initiated because of recent reporting by Grundl and Fabry^{2,3} of neutron fission spectra, which is in disagreement with the prevailing differential and integral data based on n-p scattering and time-of-flight measurements⁴; therefore, previously calculated basic nuclear parameters with Cranberg's fission spectra were recalculated with the fission spectra reported by Grundl and Fabry.

The FCX is a cylindrical two-zoned reactor. The central zone is essentially a graphite enriched uranium mixture and the outer zone is a graphite reflector. The core region of this reactor has an intermediate energy spectrum typical of NERVA reactors with a median fission energy of ~ 300 eV.

Types of nuclear calculations for this analysis consisted of the energy-dependent reaction rates with activation detectors, space-dependent material reactivity coefficients, an effective multiplication constant of the assembly, and the space- and energy-dependent neutron spectra. Absolute reaction rates of Dy, In, Au, W, Mn, Cu, ^{235}U , ^{238}U , and S were calculated with Grundl and Fabry fission spectra using one-dimensional discrete ordinate transport code ANISN,⁵ and compared with the previously measured and calculated results.⁶ Material reactivity worths of C, H, and enriched U were also calculated in this analysis and were compared with the previously measured and calculated results⁷ using one-dimensional linear transport perturbation code PERT-1D.⁸ An effective multiplication constant and the space- and energy-dependent neutron fluxes were recalculated with the ANISN code and were compared with the results obtained previously by using Cranberg's neutron fission spectra.

The results of reaction rate calculations with Grundl and Fabry fission spectra indicate that all of the detectors, except for S and ^{238}U , are within 2 or 3% of the values calculated with Cranberg's fission spectra. The results of the high energy threshold detectors S and ^{238}U change considerably because of different fission spectra and higher order scattering as shown in Fig. 1. The agreement for those threshold detectors is improved with the exception for the ^{238}U data in the reflector. The experimental ^{238}U data in the reflector does not follow the trend shown in the S pellet data.

Radial material reactivity worths of C are ~ 5 to 8% higher in the present computations compared with the previous calculations and were found to be in better agreement with the experimental measurements. Present results of radial reactivity coefficients of enriched uranium are also higher by 2 to 4% over previous calculations but are in less agreement with the measurements. The hydrogen reactivity coefficients showed small change in reactivity ($\sim 1\%$) due to use of Grundl and Fabry's fission spectra.

Effective multiplication of the assembly decreased by about 1% because of Grundl and Fabry fission spectra resulting in a subcritical prediction. Space- and energy-dependent spectra changed appreciably only above 1 MeV.

Fig. 1. Comparison of measured and calculated absolute reaction rate of 0.275-in. sulfur and 0.005-in. depleted ^{238}U foils (190 ppm ^{235}U).

This change was about 20 to 30% higher in magnitude using Grundl and Fabry fission spectra compared to the results obtained by using Cranberg data.

The results of this analysis indicate that use of Grundl and Fabry fission spectra gives better results except for the enriched uranium reactivity coefficients. Because the Grundl and Fabry fission spectrum of ^{235}U has been verified by McElroy,⁹ the use of this fission spectra for the nuclear calculations may be more appropriate.

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Analysis of Critical Experiments Using Version II of the Evaluated Nuclear Data File (ENDF/B), W. W. Little, R. B. Kidman, R. E. Schenter, L. D. O'Dell, R. W. Hardie, K. B. Stewart (PNL)

The second version (first revision) of the Evaluated Nuclear Data File (ENDF/B)¹ has recently been released from the National Cross Section Center for testing and evaluation. Version II differs from Version I in two respects: it incorporates more recent microscopic data, and it offers a more flexible and realistic data format. Indeed, the extensive format changes, which in turn have necessitated extensive modifications in the processing codes, have delayed Version II processing and evaluation. This paper describes what is thought to be the first analysis of critical experiments using Version II.

The ENDF/B Version II data were initially processed by the ETOX² code, which computes infinite-dilution cross sections and shielding factors. Effective multi-group (26 groups) cross sections for each zone in each critical assembly were then computed using the one-dimensional 1DX diffusion code.³ Two-dimensional eigenvalues and reactivity coefficients were then calculated using the 2DB⁴ and PERT-V⁵ codes. All of the above codes are compatible, and ETOX, 1DX, and 2DB are also part of the AEC's program for LMFBR code coordination.⁶

The critical assemblies studied include ZPR-III Assemblies 48, 49, 50, 53, 54, and 56. These are all plutonium-fueled criticals ranging in volume from about 200 liters (Assembly 54) to about 600 liters (Assembly 56). Assemblies 54 and 56 have an iron-nickel reflector, whereas the others have a uranium blanket. For each assembly, k_{eff} , β_{eff} , $\delta k/Ih$, isotopic reactivity coefficients, Doppler coefficients, and reaction rate ratios were computed using Version II. For comparison, identical computations using ENDF/B Version I and a modified version of the Bondarenko⁷ data were also performed.

Table I gives the computed eigenvalue for each assembly and each of the three data sets. The given values contain all correction factors which were thought to be significant, including edge effects, spiking, transport, and heterogeneity. For heterogeneity, however, only the crude approximation of +1% δk for each assembly was employed. It can be seen from Table I that Version II is about 3% underreactive for both the blanketed and reflected systems. Although much more study is needed, these results strongly indicate that the Version II cross sections are substantially in error, and thus require another revision before ENDF/B can be reliably employed in LMFBR analyses.

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TABLE I
 k_{eff} Values for Selected Critical Assemblies

ZPR-III Assembly	Cross-Section Set		
	Modified Bondarenko	ENDF/B (Version I)	ENDF/B (Version II)
48	1.003	0.979	0.968
49	1.005	0.977	0.965
50	0.996	0.981	0.962
53	1.005	0.995	0.973
54	0.993	1.024	0.963
56	1.000	0.991	0.983

Measurement and Calculation of Reactor Parameters

Analysis of Heavy- and Light-Water-Moderated Thermal Reactor Lattices Using ENDF/B Data, W. Rothenstein* (BNL)

A number of water-moderated thermal reactor lattices which have been described in the literature¹⁻³ have been reanalyzed with the HAMMER code⁴ using ENDF/B data.⁵ The calculations have revealed several areas in which the present analysis differs from those made previously.

The HAMMER code incorporates many of the features employed in the previous studies together with some improvements. A zero buckling transport theory heterogeneous cell calculation employing escape probabilities provides the flux times volumes weighting for obtaining the 54-group cross sections and precedes the B-1 MUFT homogeneous treatment of the lattice.

The most detailed comparisons between the present and previous calculations have been made for some of the standard heavy-water lattices for which full computational details were available at BNL. These lattices covered a wide range of rod sizes and volume ratios. The critical bucklings obtained by Honeck³ were used as input to the HAMMER calculations and the broad-group output was for the same 4 groups with boundaries at 0.821 MeV, 5.53 keV, and 0.625 eV.

The thermal calculations made with the aid of THERMOS in the HAMMER program yielded identical values of f as before, notwithstanding the use of the Haywood kernel,⁶ as opposed to the Nelkin kernel which had been employed in the past. The value of η was lower, however, by amounts ranging from 1.1% for the tight and 0.9% for the loose lattices. The difference was traced down to α_{th} of ²³⁵U and σ_c (2200) of ²³⁸U. In the earlier data, α decreased by 7% between 0.025 and 0.060 eV before reaching the low energy value again at 0.11 eV, and then rising in the 0.29-eV resonance. According to the ENDF/B data, α remains constant up to 0.075 eV and then starts rising in the vicinity of the resonance. The dip in α , consistent with early measurements of Palevsky et al.,⁷ occurs at energies not far from the peak of the thermal spectrum. In addition, σ_c (2200) of ²³⁸U is 2.75 b by interpolating in the ENDF/B data and 2.71 b in the original studies. Thermal leakage in the new analysis somewhat exceeds the leakage obtained previously for tight lattices, whereas the opposite is the case at large pitch owing to the difference in the manner in which D_{th} is calculated.

In the epithermal groups, the ENDF/B data lead to rather larger values of δ_{28} and the fast effect. For the standard lattices, Honeck had not taken heterogeneity effects into account in group 2, from 0.821 MeV to 5.53 keV, but had introduced appropriate corrections to the eigenvalues of the experimental lattices. At present, the HAMMER calculations allow for heterogeneity in all groups. Consequently the present eigenvalues of the standard D₂O lattices are low especially for thick rods and large pitch. Resonance integral calculations made by the ZUT and TUZ sections of HAMMER were corrected for resonance interference using the REPCDC⁸ code. Uranium-238 resonance capture is not very different in the present calculations than it was before, but the shielding of the ²³⁵U resonances is now taken into account. The deuterium scattering cross section was increased by 5% above the ENDF/B recommended value to bring the age in D₂O in line with experiments.

The eigenvalues of the experimental lattices reported in the 1965 review article³ are now, in general, between 0.98 and 0.99 using ENDF/B data and the B-1 approximation of HAMMER, whereas they were only 0.5% less than unity in the previous analysis. While much of this discrepancy may be attributed to the thermal cross sections of ²³⁵U and ²³⁸U, thermal leakage and the epithermal effects mentioned influence k_{eff} by small amounts in opposite senses.

The situation for the BNL light-water lattices¹ is similar. Two group comparisons have shown that k_{th} is rather lower in the present calculations, while k_f is generally a little larger. The value of $k_{eff} = k_f + k_{th}$ is less than unity by just over 1%, except for the tighter lattices where the deviation is about 2%.

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Inverse Multiplication Monitoring of Subcritical Reactivity Changes in FTR, S. L. Engstrom, R. A. Bennett, V. O. Uotinen (PNL)

Subcritical states of FTR will be monitored using inverse multiplication techniques, in lieu of techniques under development, such as noise analysis. Although it is known that the accuracy of this technique is sensitive to different loading patterns, data for evaluation of errors in fast plutonium criticals have not been reported. To develop data for FTR design and operational planning, an experimental investigation was made by Argonne National Laboratory in ZPR-III Assembly 56B.¹

Two sets of four B₄C-Na-steel control rods, worth ~1% $\Delta k/k$ and 2% $\Delta k/k$, were inserted into the reflector of a slightly subcritical version of Assembly 56B. The larger rods are approximate mockups of FTR peripheral control rods. Corresponding steady-state flux levels were monitored with sensors located at four positions, including the core center and 100 cm from the core centerline on the reactor midplane, as illustrated in Fig. 1. Reactivity worths inferred from the latter two sensors differed by 3% and 12% for the small and large rods, respectively.

In an attempt to identify which counter was least perturbed, a series of subcritical calculations was performed using two-dimensional diffusion theory. Calculated reactivity worths were compared with calculated values inferred from corresponding changes in ^{235}U and ^{10}B reaction rates throughout the core. These results are presented in Table I for the locations of the experimental sensors and are shown graphically in Fig. 1 for other locations on the reactor midplane. The tabulated values show that sensors located at the core center agree with calculations most closely. The calculated lines of "equal overestimation," shown in Fig. 1, illustrate that any other position is worse.

These results illustrate that special instrumentation must be installed in ZPR-III for such measurements and that these analytical techniques must be applied to the interpretation of inverse multiplication data obtained during startup and refueling of FTR.

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Fig. 1. Assembly 56B quarter core areal profile. Lines show the % overestimation of the worth of four large B_4C rods.

TABLE I

Calculated Reactivity Worths Deduced from Calculated Reaction Rates of Various Detectors

Sensor Location	Sensor	Four Small B_4C Rods ^a			Four Large B_4C Rods		
		Exp Worth (Ih)	Cal ^b Worth (Ih)	R ^c	Exp Worth (Ih)	Cal ^b Worth (Ih)	R ^c
Center of Reactor	Reaction Rate in ^{10}B	968 ± 10	1005	1.000	1804 ± 21	1923	1.003
	Fission Rate in ^{235}U		1003	0.998		1915	0.999
	ϕ Total		1002	0.997		1909	0.996
"Standard" ^d ZPR-3 Sensor Locations	Reaction Rate in ^{10}B	1000 ± 10	1056	1.051	2021 ± 21	2255	1.176
	Fission Rate in ^{235}U		1052	1.047		2215	1.155
	ϕ Total		1045	1.040		2153	1.123

^aSmall rods were one-fourth the size of the large rod and occupied the quadrant of the large rod closest to the core center.

^bCalculated worths from Δk calculations.

^cR is the ratio of the worth deduced from calculated reaction rates to the worth as obtained from the calculated eigenvalues.

^dThe "Standard" location is the sensor position generally used for subcritical measurements in the ZPR-3.

Nuclear Criticality Safety

Criticality of U(3.85) Rods and Cylindrical Annuli in Water, E. B. Johnson (UCC-ND)

In recent years the Oak Ridge Critical Experiments Facility has reported data for large uranium-metal cylindrical annuli of 1.95% enrichment in ^{235}U with water moderator and reflector.^{1,2} These studies have been extended to metal enriched to 3.85% in ^{235}U , both as solid rods and as cylindrical annuli.

The outside diameters of the annuli were 7.2, 6.2, and 5.2 in., with inside diameters of 2.6 in. Solid rods slightly less than 2.6 in. in diameter could be inserted into these annuli to produce effectively solid rods of each of the three outside diameters. All rods and annuli were 30 in. long.

These rods and annuli were latticed in water in both square and triangular patterns, each at several spacings to determine the effect of moderation on criticality. The units in each lattice were arranged in the manner most favorable for criticality so that the minimum number required was determined. The units in subcritical arrays were similarly arranged. Curves for each of the two patterns were obtained for each of the three diameter annuli and for each of the four diameter rods. In each case, the number of rods and the mass required for criti-

cality was greater for solid rods than for annuli of the same outside diameter. Figure 1 shows a typical curve for the 7.2-in.-o.d. rods and annuli. Criticality was achieved by the addition of water, in the absence of control rods, and the reactivity associated with a single rod; hence, the increments of reactivity by which a lattice could be adjusted were large. As a result, only a few lattices were critical when completely reflected by water and it was necessary to interpolate, on the basis of the height of water at criticality, between two lattices, one supercritical and the other subcritical when completely water-reflected, that differed by one rod in order to arrive at an estimate of a critical lattice at that spacing.

It was reported¹ that the number of rods required for criticality in a square pattern of the U(1.95) cylindrical annuli, 7.2-in. o.d. and 2.6-in. i.d., was a factor of 2 greater (at optimum moderation) than that for these annuli arranged in triangular patterns. In the present case, the same effect was observed but the magnitude was reduced to a factor of about 1.3 for the larger diameter annuli. However, for the same outside diameter (7.2 in.), the minimum critical number of solid rods was a factor of 1.7 greater when arranged in a square pattern than when in a triangular pattern.

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Fig. 1. Critical lattices of U(3.85) annular cylinders and solid rods moderated and reflected by water. The outside diameter of the rods and annuli was 7.2 in.; the inside diameter of the annuli was 2.6 in.

Critical Experiments with Homogeneous PuO_2 , S. R. Bierman, E. D. Clayton (PNL)

A series of critical experiments with unmoderated PuO_2 was recently completed at the Plutonium Critical Mass Laboratory of Battelle-Northwest. These experiments were part of a continuing research effort concerned with determining the criticality parameters of under-moderated plutonium as a function of ^{240}Pu isotopic concentration. This effort is oriented toward providing experimental data that can be used in establishing criticality safety guide lines and in checking calculational models over the intermediate neutron energy spectrum. Measurements on 15 and 5 H/Pu fuel covering ^{240}Pu isotopic concentrations of 2.2, 8, and 11.5 wt% have been previously reported¹⁻³ as has the description of the facility.³ This paper is concerned with the results of the latest series of experiments in this program in which the

TABLE I
Critical Dimensions in Spherical, Cylindrical, Cubic, and Slab Geometries—Unmoderated PuO_2 at 5.77 g Pu/cm^3 and 18.35 wt% ^{240}Pu Isotopic Content

Geometry	Bare Assemblies		Plexiglas Reflected	
	λ (cm)	X^2 (cm)	λ (cm)	$X^2(a)$ (cm)
Infinite slab	3.62 ± 0.2	12.17 ± 0.28	8.20 ± 0.17	3.34 ± 0.10
Sphere	3.60 ± 0.2	15.81 ± 0.20	8.17 ± 0.38	11.24 ± 0.38
Infinite cylinder	3.58 ± 0.2	11.28 ± 0.20	8.17 ± 0.38	6.69 ± 0.38
Cube	3.57 ± 0.2	26.48 ± 0.28	8.13 ± 0.38	17.36 ± 0.54

^a Critical thickness of slab or cube and critical radius of sphere and cylinder.

^{240}Pu isotopic content has been increased to 18.35 wt% and the H/Pu atomic ratio decreased to near zero (0.038).

Experimental data were obtained from both bare and Plexiglas reflected rectangular parallelepipeds constructed from $2 \times 2 \times 1\frac{1}{2}$ -in. blocks of PuO_2 compressed to a density of 5.770 ± 0.039 g Pu/cm^3 . To achieve criticality with a bare assembly, some PuO_2 -polystyrene fuel at an H/Pu atomic ratio of 5 had to be used as a driver with the unmoderated PuO_2 . By making calculational corrections for the PuO_2 -polystyrene driver region, a critical buckling of $262 \pm 2 \text{ m}^{-2}$ was determined for the bare unmoderated PuO_2 fuel. The reflected critical assemblies ranged from a near-cubic geometry to thin slabs for an infinite slab thickness of $3.34 \text{ cm} \pm 0.10$ cm and a corresponding extrapolation length of 8.20 ± 0.22 cm. The measured critical heights of each reflected assembly are shown in Fig. 1 as an inverse function of the assembly's cross-sectional area.

Fig. 1. Measured critical height.

Using a constant buckling conversion, critical sizes have been derived from these experimental data for spherical, cylindrical, cubic, and slab geometries. These critical dimensions are given in Table I.

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Criticality Measurements on Uranium Metal Solution Combinations, D. C. Hunt, R. E. Rothe (Dow-Colo)

Critical parameters are not well known¹⁻⁷ for systems where fissile metal shapes are immersed in fissile solutions; and yet, many operations in a nuclear materials processing plant are of such a nature. If the criticality engineer is to provide safe but economical operational limits, he requires additional experimental data on these systems.

A measurement program was initiated at Rocky Flats to provide some of these needed data. The first data in this series are reported here for systems composed of enriched uranium metal spheres symmetrically positioned within enriched uranium solution cylinders. In both the metal and the solution the uranium is enriched to 93.18 wt% ^{235}U (5.38% ^{238}U , 1.01% ^{234}U , and 0.43% ^{236}U). The average density of the metal assemblies ranges from 17.7 to 18.0 g/cm³. The uranyl nitrate solution concentrations used are 11.0, 21.3, and 104.9 g U/liter. Critical masses and radii of the metal spheres are given in Table I for solution cylinders having 51.1- and 38.4-cm diameters and various thicknesses.

These measurements are performed in an environment similar to that found in a typical processing plant. The system centers range from 1.6 to 2.0 m above the floor, and they are about 0.5 m from a thick neutron-reflecting wall. Type-304 stainless-steel tanks, with 0.5-cm wall thickness, contain the fissile material during experiments. The metal sphere is suspended by a 0.6-cm-diam stainless-steel rod through the assembly. For the 11.0 g/liter measurements, the metal spheres were supported just below the equator by a 0.3-cm-thick stainless-steel plate which poisoned the system sufficiently such that the 9.8, 4.5, and 1.6 kg critical masses reported in Table I became 12.4, 7.9, and 5.6 kg, respectively, when this plate was used. The 11.0 g/liter data given in Table I have been corrected for this effect; no other adjustments to the data have been made for other environmental effects.

TABLE I

Critical Parameters of Uranium Metal Spheres Symmetrically Positioned Within Uranium Solution Cylinders

Uranium Solution Cylinder Description			Uranium Metal Sphere Critical		
Conc (g/liter)	Diam (cm)	Thickness (cm)	Mass (kg)	Radius (cm)	
11.0	51.1	30	19.0 ± 0.2	6.31 ± 0.02	
		50	16.7 ± 0.2	6.04 ± 0.02	
		70	16.8 ± 0.2	6.06 ± 0.02	
21.3		30	9.8 ± 0.2	5.06 ± 0.03	
		50	4.5 ± 0.2	3.92 ± 0.06	
		70	1.6 ± 0.2	2.76 ± 0.11	
104.9		18	4.1 ± 0.2	3.79 ± 0.06	
21.3		38.4	16	~25 ± 2	7.2 ± 0.3
			20	16.9 ± 0.2	6.05 ± 0.01
	32		10.3 ± 0.2	5.10 ± 0.02	
	48		6.9 ± 0.2	4.51 ± 0.04	
104.9	16		12.2 ± 0.2	5.45 ± 0.05	

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Criticality Calculations on Uranium Metal-Solution Combinations, D. C. Hunt, R. E. Rothe (Dow-Colo)

Certain operations in fissile material processing plants typically involve dissolution of uranium metal in nitric acid.¹⁻⁴ To allow nuclear safety evaluations of such operations, a series of criticality calculations was performed on systems of enriched uranium metal spheres in uranyl nitrate solution. Related calculations have been reported on plutonium metal in plutonium nitrate solution^{5,6} and on uranium metal slabs in uranium water mixtures.⁷

Two types of calculations are presented. First, a survey is performed using both diffusion theory calculations (the MDC⁸ computer code) and transport theory calculations (the DTF⁹ computer code). Second, the results of DTF calculations which approximate experimentally measured systems¹⁰ are compared with the experimental results. Both sets of calculations use the 16 neutron-energy-group cross-section sets of Hansen and Roach.¹¹

In the first set, diffusion theory calculations are performed if no thin regions, such as tank walls, are included. When such regions are considered, DTF calculations in the S_4 approximation are done. In these calculations, the metal density is 18 g/cm³, the uranium isotopic composition is 93.19 wt% ²³⁵U and 6.81 wt% ²³⁸U, and all regions are concentric spheres. The principal variables are the metal mass, the solution volume, and the concentration of uranium in solution.

Some representative calculational results appear in Fig. 1 where the metal mass is plotted vs solution concentration for critical systems with the total volume (solution plus metal) held fixed. A poisoning effect is predicted for small metal masses such that adding a

Fig. 1.

TABLE I
Comparison of DTF Calculations with Experimental Results

Sphere Density (g/cm ³)	Sphere Radius (cm)	Solution Concentration (g U/liter)	Observed ^a Solution Thickness (cm)	Computed Solution Thickness (cm)
18.03	6.31	-----	16.44	10.69
18.02	6.04	11.01	21.93	16.71
18.02	6.06	11.01	24.34	22.90
17.96	5.06	21.25	17.69	16.73
17.89	3.92	21.25	24.05	23.85
17.73	2.76	21.25	27.64	27.56
17.88	3.79	10.49	11.77	11.58

^aObtained by a buckling conversion

small sphere of metal to a critical solution-only system makes it subcritical.

In the second calculation set, the S_{16} approximation is used and the spatial mesh is determined according to the criteria specified by Hunt and Coonfield.¹² The results of these calculations are compared with experimental results in Table I.¹⁰

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Minimum Critical Mass of ^{235}U Reflected by Natural Uranium in Water, A. W. Gray, L. L. Jones, Jr. (W-BAPL)

An analytical study has been completed to define minimum critical mass parameters for a spherical region containing a ^{235}U - H_2O mixture reflected by a natural uranium- H_2O mixture. This study was prompted by the minimum critical mass value of 250 g ^{235}U reported by Jarvis and Mills¹ in 1967. The study indicates that for an optimum natural uranium- H_2O reflector, the minimum critical mass for the central sphere lies in the range of 75 to 200 g ^{235}U .

The analytic study was divided into three phases. In the first phase, homogeneous two-region systems were examined using the diffusion theory WANDA Code.² The configuration studied consisted of a spherical core surrounded by a homogeneous mixture of natural uranium and H_2O . The H-to- ^{235}U ratio was varied in the central core and reflector region. The results indicate a minimum critical mass of 451 g ^{235}U occurs for a H/ ^{235}U equal to 450 in both regions.

In the second phase, a Monte Carlo program was utilized to compute K_{eff} for uranium metal rod- H_2O moderated lattices for which experimental data are available. The lattices included natural, slightly enriched, and

TABLE I

Enrichment wt% ^{235}U	Rod Dimensions o.d./Length (cm)	Lattice Pitch (cm) Square or Hexagonal	Experimental Result Critical No. or K_{eff}	Calculated K_{eff} with 95% Confidence-Interval
Natural	3.0/00	4.5 - hex	0.970 ± 0.015^a	0.963 ± 0.016
1.136	3.048/76.20	4.7 - sq	195 ± 9^a	0.967 ± 0.012
3.063	2.35/60.96	4.57 - hex	47^{b-c}	0.999 ± 0.007
3.063	1.524/40.64	3.607 - hex	100^{b-c}	0.993 ± 0.017
93.6	0.3175/30.48	1.587 - sq	149 ^d	1.001 ± 0.013

TABLE II

^{235}U Mass (grams) in central core at H/ ^{235}U = 450	Reflector Configuration (outermost reflector) is H_2O in each case	Calculated K_{eff} with 95% Confidence-Interval (~10,000 neutron histories)
539	17,000 kg natural uranium-optimum lattice	1.115 ± 0.016
235	17,000 kg natural uranium-optimum lattice	1.029 ± 0.016
539	225 kg natural uranium-optimum lattice	0.989 ± 0.018
539	225 kg natural uranium-homogeneous mixture-H/ ^{235}U = 450	0.983 ± 0.024
539	Infinite H_2O	0.915 ± 0.017
235	225 kg natural uranium-optimum lattice	0.873 ± 0.015

highly enriched uranium metal rods. Results obtained for the Monte Carlo calculations are tabulated in Table I.

In the third phase, the Monte Carlo program was used to calculate $^{235}\text{U} + \text{H}_2\text{O}$ spheres closely reflected by an optimum lattice of natural uranium rods. The optimum lattice parameters were assumed to be those determined in Ref. 3. The results obtained are shown in Table II.

The results indicate that K_{eff} varies linearly with mass of ^{235}U in the central region over the range 0 to 600 g. This indicates that the minimum critical mass in an H_2O -moderated system reflected by a natural uranium metal lattice is 135 ± 65 g ^{235}U . When the amount of natural uranium metal is limited to 225 kg, the minimum critical mass increases to 565 ± 50 g ^{235}U . A comparison of results obtained in phase 1 and phase 3 indicates that when the reflector is limited to ~200 kg uranium, latticing does not decrease the minimum critical mass.

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Comparisons of Experiment with Calculations for Strongly Absorbing Rod Lattices, J. K. Fox (Id Nucl Corp)

During chemical reprocessing of reactor fuels, one may encounter types that are not designed for water moderation. Strongly absorbing fuel pins such as EBOR or EBR-2 fuels when placed in water lattices have deep flux depressions in the thermal range. Few-group calculations do not give good agreement with experiment for these fuels. The same is somewhat true of lattices of metal rods enriched from 3 to 5% for the larger rod sizes. A better spatial and energy representation of the depressions is obtained if one uses several groups in the thermal range.

Square pitched lattices of EBOR pins¹ in water have been studied at ORNL with the spacing being varied. These data were used in two sets of comparisons. In one set, the usual four-group structure was employed. In the second set, there were three groups in the fast range, ten groups in the resonance range, and ten groups below 2.38 eV. Group constants for groups above thermal were obtained using the B_1 approximation in PHROG.² Resonance integrals were evaluated with the Nordheim formulation. The thermal range constants were obtained using INCITE,³ an integral B_1 approximation code, with the McMurry-Russell scattering kernel. Disadvantage factors were obtained in all groups with SCAMP,⁴ a multigroup S_n transport code, in a unit cell calculation. These factors were then applied to a second PHROG-INCITE set for the final homogenized constants in the fuel region. Eigenvalues were obtained using MONA,⁵ a multigroup diffusion code in one dimension with transverse leakages represented by input bucklings. The comparisons are shown in Table I. The length of the EBOR pins of 76 in. means that errors in estimating the Z input buckling are small. The three separations chosen are from an enlarged plot of the data and represent a wide variation in fuel-to-water ratios. The experimental lattices were rounded to approximate a cylinder with infinite water reflection. The agreement with experiment is much better for the 23-group cases than for the 4-group results. For the multigroup cases, there is a slight trend for wider separations.

Experiments were also done at ORNL⁶ for square lattices of 4.89% enriched metal rods in water. These experiments used 2.0675-cm-diam rods that were 60 cm in length. The same codes were used as before in these calculations. A 12-group structure was chosen with 7 groups in the thermal range. Here the Z direction leakage is significant. Many experiments on rod lattices indicate that the extrapolation in water is usually between 7.0 and 8.5 cm. The values reported in Table I corresponded to an extrapolation of 7.5 cm. Extrapolations required for exact agreement with the three experiments are: 8.95, 8.3, and 9.25 cm, respectively. Thus, the agreement for each metal-to-water ratio is good.

TABLE I

Comparisons of Experiment with Calculations

A. Lattices of EBOR pins in water: 62.4 T enriched $UO_2 + BeO$, Av pin o.d. = 0.364 in., 76 in. length, square lattices in rounded array outline.			
Pitch ^a (cm)	Experimental Critical # Pins	Calculated keff	
		23 Group	4 Group
1.2395	220 $\pm$ 1	1.001	1.018
1.8447	94.0 $\pm$ 0.5	0.994	1.032
2.8956	79.0 $\pm$ 0.5	0.991	1.039
^a Separations from plot of experiments.			
B. Lattices of 4.89% enriched ^{235}U metal rods in water: Rod length = 60 cm, rod diam = 2.0675 cm Square lattices in rounded outline.			
Pitch (cm)	Experimental # Rods	Experimental keff	Calculated keff
3.20	51	1.0023	0.9995
4.14	39	1.0043	1.003
5.20	54	1.0005	0.996

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Certification of Nuclear Criticality Safety Specialists, Norman Ketzlach (AI), Panel Moderator

The American Nuclear Society Technical Group for Nuclear Criticality Safety established a committee in June 1968 to investigate the need for a certification program to qualify specialists in nuclear criticality safety. The following are some of the questions that must be answered before a decision can be reached:

1. What are the objectives of certification?
2. What are we qualifying?
3. Whom are we qualifying?
4. Will industrial nuclear safety be enhanced?

In determining whether a certification program is desirable, one must consider its possible impact in the future as well as today. The need for a certification program for nuclear criticality safety specialists was discussed by interested parties during the November 1968 and June 1969 meetings of the ANS. Because of the great interest shown in the subject, both for and against a certification program, the Technical Group decided to sponsor a panel discussion on the subject. The panelists selected to cover this topic have experience in government and industry as well as with the certification program in an allied field.

There are a number of reasons given for certification. In general, they include:

1. to elevate the standards and advance the cause of the profession by encouraging its study and improving its practice
2. to encourage and insist on the highest standards of ethics and integrity in the practice
3. to establish a level of competence of specialists
4. to provide personal benefit to those having certification.

It is felt that a certification program should be directed to those involved in the application of nuclear criticality safety principles to plant processes. In this connection, it is necessary to define the principal objectives and qualifications for certification in nuclear safety. Otherwise, it is felt by some that a regulatory agency will eventually establish its own criteria.

At the present time, several states require that the director of their health physics program be a certified health physicist. The requirement for a certified health physicist is specified in a number of government contracts.

Recommended practices, established by technical society groups (e.g., Society for Nondestructive Testing), are used by both government and industry in establishing a general framework for qualification and certification programs.

The reasons given against a certification program for nuclear criticality safety specialists include the following:

1. Nuclear criticality safety involves a team approach representing many technical specialties and levels in the organization. No individual can qualify in all these areas.
2. Certification would be misleading. A person may qualify in some areas but not necessarily those of greatest importance to the organization the certified specialist works for.
3. Nuclear criticality safety should not be set apart but should be considered an integral part of an overall safety program.
4. The reasons given for certification in general could be accomplished through the ANS Technical Group by other routes, such as sponsoring technical sessions for the presentation of criticality data and training and educational programs.

There are presently two certification programs in allied fields. One is conducted by the American Board of Health Physics for health physicists; the other, by the Institute of Nuclear Materials Management to qualify nuclear materials managers. The former program is about ten years old; the latter, about seven. There are also certification programs sponsored by the American Industrial Hygiene Association and the American Board of Radiology. The history of these other programs as well as an evaluation of the degree to which they meet their original intent should be of assistance in answering the basic question of whether Nuclear Criticality Safety Specialists should be certified?

Nuclear Criticality Safety Computations—Tutorial

Basic Theory and Concepts, H. C. Honeck (duPont-SRL)

The fundamental problem in nuclear criticality is the computation of neutron reaction rates in a multiplying assembly. The balance between reaction rates determines the behavior of the assembly. Simple formulas can be derived to illustrate this balance and the concepts of effective multiplication, migration area, and buckling.

Reaction rates can be estimated using several computational techniques, all of which are derived from the basic equation describing neutron transport. This "Transport Equation" is very complex and contains constants (cross sections) which are experimentally measured. The transport equation must be approximated to obtain numerical solutions both because of the complex nature of the equation and because of the nonanalytic form of the cross sections. Each of these approximate methods is well suited for some applications, but none is economically applicable to all situations. Thus, the criticality analyst must select the best method for the problem at hand and he must understand the approximations used and be able to estimate their validity. Most often this selection of the best method depends upon extensive comparisons of calculated and experimental results, the topic of another paper in this session. The choice also depends on economic considerations, this is, the tradeoff between accuracy and computing dollars.

The approximate methods used frequently for criticality calculations are classified into several broad categories: Monte Carlo, discrete ordinates, integral transport, and diffusion theory. Within each category there exist several computer programs, each with a different treatment of spatial detail and energy group structure. Several of these methods are topics for later papers in this session. In the remainder of this paper we will describe some general aspects of each method and indicate the types of situations where each is used.

The Monte Carlo method is potentially the most exact and flexible method for estimating reaction rates. In this method, the computer simulates individual neutron events so that nuclear processes and complex geometry can be modeled in great detail. The accuracy and computation time increase with the number of events simulated. Many refinements of the method have been developed; therefore, coupled with today's fast computers, Monte Carlo is a widely used method.

The discrete ordinates method is obtained from the transport equation by approximating the neutron angular distribution in terms of a set of discrete angular ordinates and an appropriate interpolation formula between ordinates. The resulting set of coupled linear equations can be solved rapidly and accurately. Combined with the multigroup approximation, the discrete ordinates method is applicable to a wide class of problems which can be described by simple one- or two-dimensional geometry.

The diffusion theory method further restricts the neutron angular distribution to a linear variation. The resulting set of linear equations can be solved very rapidly. Multigroup diffusion theory is applicable to problems which can be described by simple one- or two-dimensional geometry, which are large in terms of

distances between neutron collisions, and which are nearly homogeneous.

The integral transport method is obtained from the integral form of the transport equation with the approximation that the scattering process is nearly isotropic. The resulting multigroup matrix equation can be solved rapidly. Integral transport theory is best suited to the analysis of complex heterogeneous lattices. The results are generally used to produce averaged cross sections for later calculations in discrete ordinates or diffusion theory.

The S_N Method, K. D. Lathrop (LASL)

The major difference between the Boltzmann transport equation and many other important equations of mathematical physics is that the domain of integration includes both space and angle (velocity). Many of the deterministic methods for numerical solution of the Boltzmann equation simplify the equation by eliminating the angle dependence. In the spherical harmonics equations the flux is expanded in terms of known angular functions, while integral equation formulations are obtained by integrating over the angular variable. Both of these approaches pay a price for simplification. The spherical harmonics approximation leads to sets of coupled differential equations in the space variable for which boundary conditions are hard to formulate and for which numerical solution methods are hard to generalize for arbitrary order of approximation. Integral equation formulations lead to kernels which extend over the whole space domain, producing dense coefficient matrices in the algebraic equations which are used in the approximate solution of the integral equation.

The present S_N technique deals directly with both the space and angle variables in the differential-integro form of the Boltzmann equation. Historically, the method also tried to eliminate the angular dependence by assuming a set of n connected straight line segments in the angular cosine; hence, the name $S(\text{egment})_n$. Although accurate, this method proved hard to generalize, and now the S_N label refers to a discrete ordinates formulation. For example, in a spherical geometry in which the flux depends only on the distance, r , from the origin, the flux depends only on the cosine, μ , of the angle between the radius vector and direction of particle motion. Then, the Boltzmann equation for the flux $\psi(r, \mu)$ in each group of a multigroup approximation can be written

$$\frac{\mu}{r^2} \frac{\partial(r^2 \psi)}{\partial r} + \frac{1}{r} \frac{\partial[(1-\mu^2)\psi]}{\partial \mu} + \sigma \psi = \frac{\sigma_s}{2} \int_{-1}^1 \psi(r, \mu') d\mu' + S, \quad (1)$$

where we have assumed isotropic scattering for simplicity. In the discrete ordinates approximation, $\psi(r, \mu_m)$ is the ordinate obtained from the solution of Eq. (1) evaluated at the discrete value $\mu = \mu_m$. In such an approximation, Eq. (1) might become

$$\frac{\mu_m}{r^2} \frac{\partial(r^2 \psi_m)}{\partial r} + \frac{1}{r} \frac{[(1-\mu_{m+1/2}^2)\psi_{m+1/2} - (1-\mu_{m-1/2}^2)\psi_{m-1/2}]}{\Delta \mu_m} + \sigma \psi_m$$

$$= \frac{\sigma_s}{2} \sum \Delta \mu_m \psi_m + S_m, \quad (2)$$

where we have written

$$\psi_m = \psi(r, \mu_m)$$

and

$$\Delta \mu_m = \mu_{m+1/2} - \mu_{m-1/2}$$

In this equation, μ_m and $\Delta \mu_m$ define a quadrature set, and some sets are more accurate than others. Equations like (2) are easy to formulate in any geometry, and, when spatially differenced, these are readily solved to obtain the flux as a function of space and angle.

There are many programs which solve the multigroup discrete ordinates equations, and they fall into three categories: (a) one-space dimension programs for planes, spheres, and cylinders; (b) two-space dimension programs for rectangles, cylinders (infinite and finite), and spheres; and (c) time-dependent programs. In all these programs it is easy to refine the angular resolution, the space resolution, and the energy resolution. In all these programs it is easy to specify a variety of useful boundary conditions along the system surfaces. For static calculations, most S_N programs perform inhomogeneous source, criticality, or other eigenvalue computations. All such programs can generate the angular flux at every point in the whole system as a part of the computation. The one-dimensional programs are computationally cheap, while two-dimensional programs can compute k_{eff} for a very detailed description of an entire reactor in two hours (CDC-6600).

S_N computations are most appropriate when transport effects are large, i.e., when absorption or leakage is large, sources are isolated, or when streaming effects dominate; otherwise, diffusion theory is adequate. For this reason S_N methods are frequently used in unit cell calculations or in shielding problems. In applications, S_N programs provide a detailed flux that is reproducible; in many circumstances it is possible to perform perturbation calculations by subtracting results of two S_N computations. More systematic perturbation results can be obtained by using both direct and adjoint fluxes from S_N computations.

S_N programs suffer from a lack of geometric generality; not only is there no three-dimensional program generally available, but also the specification of general curved surfaces in 2-D codes is painful and wasteful of mesh boundaries and, hence, of computational time. S_N programs in more than one space dimension also suffer from the ray effect (a defect of the present discrete ordinates formulation) in which streaming along the discrete directions leads to flux distortions; e.g., whenever sources are isolated in absorbing media. Presently, the only remedy for these distortions is to increase the number of directions, at the expense of computing time. These defects plus the general problem of convergence acceleration are the subject of much current research.

The Monte Carlo Method as Applied to Nuclear Criticality Safety Calculations, G. E. Whitesides (UCC-ND)

Until a few years ago the Monte Carlo method was treated as a curiosity by most criticality safety personnel, theoretically capable of solving criticality safety problems, but too expensive to be of any practical use in routine calculations. Two different economic considerations have occurred which have caused renewed

interest in the method. First, as computers have grown in capacity and speed, the cost per computation has gone down significantly; second, the economics of manufacturing, storing, and shipping of fissile materials have reached the point that the use of inexact or less rigorous methods often results in intolerable economic inefficiencies.

With this background, the capabilities of the Monte Carlo method will be examined. The major advantage currently being exploited is the geometric capability of the Monte Carlo method. While finite difference techniques, in general, are restricted to regular geometries such as spheres, cylinders, and slabs, the Monte Carlo method is essentially unrestricted by geometric complexity. In particular, the rigorous computation of finite, complex, arrays of fissile material is not possible with any other method.

The other major advantage of the method is the ability to use very detailed cross-section information instead of group-averaged cross sections. While much work is done using detailed cross sections, the lack of extensive validation of particular sets of data has thus far prohibited the use of such data in routine criticality safety analysis.

The analog or direct simulation approach to the solution of neutron transport problems makes the understanding of the application of the Monte Carlo method quite simple. Upon closer examination it is evident there is often need to supplement the analog method with techniques to reduce the computer time required to solve a particular problem. A summary of the commonly used techniques will be presented along with a brief justification of their use.

Extensive use of the Monte Carlo method at Oak Ridge has provided data which indicate that, for many criticality safety problems requiring more than one geometrical dimension for description, the Monte Carlo method is the least expensive method of obtaining a solution. The major gain, however, is often not the saving of computer time, but the gain in confidence which results from being able to calculate exactly the fissile system under consideration, with no geometrical approximations.

Nuclear Cross Sections—Multigroup Representation of Neutron Cross Sections for Criticality Calculations, Gordon E. Hansen (LASL)

In the 1940's one- and two-group neutron diffusion or transport calculations proved very successful in predicting criticality of the thermal and small-fast chain-reacting systems then in vogue. Group parameters, however, were based principally on integral data rather than microscopic cross-section data. Thus, for the thermal systems one relied on observed values of fission-neutron age to thermal and resonance escape probability, while for the fast systems one relied on various fission-neutron or mock-fission-neutron cross sections. One may note that, for the fast systems, group cross sections for a mixture of elements were obtained by simply adding up contributions of element group cross sections; whereas for thermal systems, the mixture-dependent structure of the neutron flux in the resonance region prevented such additivity. Common to both types of systems were neutron energy spectra which were sensibly space independent, and that was the feature that permitted successful one- and two-group diffusion calculations.

Interest in multigroup diffusion calculations arose with interest in reactive systems whose neutron energy spectra were space dependent. An example was KAPL's intermediate-spectrum reactor for submarines consisting of an undermoderated core and moderating reflector. Actually, the effect of spatial dependence of neutron spectra on criticality is subtle but its effect on power density or other reaction rate density distributions is gross. Successful prediction of such distributions requires multigroup, or equivalent, diffusion calculations even for so-called thermal reactors. The present talk, however, is concerned with nuclear cross sections only as they sensibly influence criticality.

Several years ago, in a *Nuclear Science and Engineering* article entitled "Polarization effects on Neutron Transport," Bell and Goad estimated the effect on neutron diffusion of angular correlation between successive neutron scatterings and concluded such correlations could contribute several percent to the reactivity of fairly heavy element intermediate-spectrum systems. This conclusion completely undermines our old faith that the scalar Boltzmann transport equation was the precise equation for neutron transport and that single neutron-nucleus interaction cross sections furnished the required input data. Several percent in reactivity sounds like quite a bit, and one might ask whether any of the intermediate-spectrum critical assemblies reveal such a polarization effect on criticality. I believe the answer is "Not quite yet: several percent in reactivity is just too small an effect. There is still enough leeway in selection or evaluation of nuclear cross-section data to permit us to reproduce observed criticality with our usual method of calculation." We can sort of illustrate the status of nuclear cross-section data for intermediate-energy neutrons by pretending to search for a "polarization-effect" on criticality.

We look at bare systems comprised solely of ^{235}U and ^{238}U . These satisfy the fairly heavy element criterion; the various neutron cross sections for each isotope have been measured and remeasured; the entire concentration range is covered by good criticality data; and the near space-independence of the neutron spectra simplifies the criticality calculations (two or three groups are ample, but of course one must calculate the so-called infinite-medium spectrum in great detail). The questions that have to be dealt with (but not in this summary) are:

1. What are the uncertainties in the available ^{235}U and ^{238}U cross-section data, or equivalently, what is the accuracy of the "evaluated cross-section data sets"?

Conclusion: insufficient for our present game.

2. To make applicable for criticality calculation, how far into the unresolved resonance region must one superimpose fine structure on the observed mean reaction cross sections.

3. To what extent can one improve the accuracy of the ^{235}U and ^{238}U cross-section data sets by weighting-in diagnostic data from the uranium criticals such as ^{235}U capture to ^{235}U fission ratios, or weighting in other criticality data for which polarization effects are negligible?

A Method for Testing Neutron Cross Sections and Neutronics Computer Programs by Comparison with Integral Neutron Experiments, Viktor E. Hampel (UCLRL-Livermore)

The utilization of nuclear energy has always relied to a large extent upon comparison of calculations with

experiments. This has contributed to safe and reliable operations, and has permitted predictions for more complex applications. This effect is supported by the facilities and publications of the Criticality Data Center at ORNL, of NSIC, and RSIC, and of our national laboratories under the auspices of the U.S. Atomic Energy Commission. In addition, the world literature abounds with documents which show, for specific cases, whether some aggregate of fissile materials is safe, and whether it can be put to use as anticipated.

In the past, widely differing computational tools and disparate nuclear constants were often employed to simulate on computers nuclear processes that interact in an integral experiment. In the absence of a prototype experiment, the comparison was usually carried out for one or a few critical or bulk assemblies, selected from the literature because of their similar geometry, composition, and neutron energy spectrum. Under these conditions of normalization, the accuracy assigned to calculations assumed an acceptable but limited range of validity.

In recent years this situation has changed. A gradual transition is apparent from personal and institutional neutronics codes and cross sections to a fewer number of highly developed computer programs, and to more universal and comprehensive libraries of computerized microscopic cross sections and evaluated data.¹⁻⁴ The systematic validation of these computer programs, their associated processing codes and the vast stores of nuclear constants, is now of primary concern.

Today, there does not yet exist an accepted and sufficiently diversified library of integral neutron experiments which could serve as the proving ground for routine testing and improvement of codes and cross sections. However, incipient efforts have been made in the United States and abroad. This paper examines the different approaches being taken, and outlines principal procedures of one such method which has been developed by the Physical Data Group, Lawrence Radiation Laboratory, Livermore.⁵

Specifically, collections of indexed information on integral neutron experiments have been prepared in France, the United Kingdom, and the United States, and these serve as ready references to nuclear safety and reactor design engineers. In addition, Pendlebury began a program in 1964 at Aldermaston for the mechanized fitting of neutron cross sections to a select few of the VERA and ZEBRA experiments.^{6,7} In the United States, Harry Alter of Atomics International is currently directing the testing of the ENDF/B library. These tests compare calculated integral parameters for 15 to 20 different experiments; in particular, in support of the LMFB program, those of the ZPR assemblies.⁸ The interpretation of calculations for some of these originally heterogeneous systems containing numerous materials is often admittedly difficult.

At Livermore, we have started with the simple spherical assemblies of ^{235}U , ^{238}U , and ^{239}Pu , reflected by one element or by one compound, in varying thicknesses. This will ultimately be extended to simple cylinders, slabs, and solutions. However, to provide a bridge to other efforts now in progress, some of the ZPR, VERA, and ZEBRA configurations are included.⁹

To make selection of pertinent documents from the world literature easier, a computer program has been prepared for storing descriptive text and numerical data.¹⁰ The resulting library of integral experiments¹¹ currently contains about 1500 references, abstracts, and

comments from which 115 unclassified spherical assemblies² have been identified to serve as direct input to SORSE Monte Carlo and ANISN neutron-transport calculations. Whereas SORSE serves to establish primary integral parameters and neutron energy spectra which are less limited by implications imposed by fluxweighting, selfshielding, and geometry, ASISN calculations in the S_{16} - P_3 -66_{grp} format are used for sensitivity studies and for examining such measurements as central reactivity worths and reaction ratios.

To permit intercomparison of nuclear constants from different sources, translation routines have been written to transfer data to and from the ENDF/B and AWRE magnetic tape libraries. All these bibliographic and numerical data and computer programs are stored in the IBM-1360 Photo-Digital Storage System² which is linked to CDC-7600 and CDC-6600 computers, thus permitting quick and routine testing, comparison, and refinement of codes and nuclear constants. The results and the implications of this group effort will be presented.

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²The IBM-1360 has a total storage capacity of 10^{12} bits. Readout times average 1.5 million bits per second.

Data Applicable to Nuclear Criticality Safety Guidance

Criticality Calculations and Measurements on Enriched Uranium Metal-Solution Combinations, D. C. Hunt, R. E. Rothe (Dow)

Criticality measurements and calculations on enriched (93.2 wt% ^{235}U) uranium metal spheres symmetrically immersed in enriched (93.2 wt% ^{235}U) uranyl nitrate solution have been presented.^{1,2} The measured critical masses of the metal spheres and the total uranium masses at criticality were reported for uranium solution cylinders of three nominal concentrations, three diameters, and several heights. Fourteen transport theory calculations (using the DTF code³) were reported on systems which approximated the experimental assemblies for one diameter, four heights, and three concentrations. In addition, results from diffusion theory calculations (using the MDC code⁴) relating critical sphere radii to critical solution concentrations were given.

In this paper, 25 additional DTF calculations approximating the symmetric experimental assemblies are presented. These calculations are done in the S_{16} angular approximation with saturated spatial meshes⁵ using Hansen-Roach⁶ 16-energy-group neutron cross sections. Also, each of the systems are calculated using the multi-group Monte Carlo code KENO⁷ with the Hansen-Roach cross sections. The experimental geometry is more

accurately reproduced in the KENO calculations because of the geometric flexibility typical of Monte Carlo methods. The cylindrical shape of the experimental vessel, the glove-box enclosure around it, and the nearby concrete assembly room walls are included in the geometric description of the problems. The calculations use from 20 to 30 batches of 300 neutrons each with source effects assumed to die away after 5 batches are processed. A typical standard deviation for the neutron multiplication factor is 0.008. Table I gives the DTF and KENO calculated results along with the experimental parameters. No systematic bias appears to exist in either calculational method between the computed and observed critical sphere radii. The critical radii are, however, overpredicted by both codes more frequently than they are underpredicted. KENO yielded poorer results at higher solution concentrations. For one such case, the computed sphere radius is 4.39 ± 0.24 cm compared to the observed radius of 3.81 ± 0.02 cm. The reason for this discrepancy is not known but may be related to the hydrogen scattering model used in KENO. In KENO, the Hansen-Roach energy group transfer cross sections are used to determine the energy loss in a collision and a mean scattering angle then found from the first two Legendre polynomial coefficients.⁷

Additional experimental criticality data are presented for systems in which the uranium metal spheres are

TABLE I
Comparison of DTF and KENO Calculational Results*

Solution Cylinder Properties			Sphere Density (g/cm ³)	Sphere Radius (cm)		
Diameter (cm)	Height (cm)	Concentration (g/liter)		Experimental	Computed ^a	Computed ^b
51.1	30	13.12 $\pm$ 0.05	18.012 $\pm$ 0.002	5.948 $\pm$ 0.003	5.90	5.935 $\pm$ 0.035
	50		18.006 $\pm$ 0.000	5.700 $\pm$ 0.003	5.74	5.69 $\pm$ 0.32
	70		18.004 $\pm$ 0.000	5.679 $\pm$ 0.007	5.685	5.74 $\pm$ 0.07
	30	21.25 $\pm$ 0.22	17.964 $\pm$ 0.005	5.085 $\pm$ 0.027	4.93	5.04 $\pm$ 0.07
	50		17.890 $\pm$ 0.006	3.922 $\pm$ 0.003	3.85	3.80 $\pm$ 0.10
	70		17.725 $\pm$ 0.005	2.792 $\pm$ 0.033	2.69	2.832 $\pm$ 0.034
	16	104.8 $\pm$ 0.6	17.889 $\pm$ 0.017	3.81 $\pm$ 0.02	4.055	4.39 $\pm$ 0.24
	19.2 $\pm$ 0.1		---	0.0	15.88 ^c	17.92 $\pm$ 0.61 ^d
38.4	20	11.47 $\pm$ 0.12	18.036 $\pm$ 0.001	6.555 $\pm$ 0.005	6.438	6.535 $\pm$ 0.14
	32		18.024 $\pm$ 0.002	6.122 $\pm$ 0.010	6.160	6.07 $\pm$ 0.20
	48		18.012 $\pm$ 0.003	6.024 $\pm$ 0.017	6.060	6.025 $\pm$ 0.042
	16	24.20 $\pm$ 0.59	18.076 $\pm$ 0.067	7.17 $\pm$ 0.34	6.36	6.97 $\pm$ 0.13
	20		18.014 $\pm$ 0.000	6.053 $\pm$ 0.005	5.91	6.18 $\pm$ 0.165
	32		17.974 $\pm$ 0.003	5.099 $\pm$ 0.022	5.18	5.34 $\pm$ 0.15
	48	104.8 $\pm$ 0.6	17.901 $\pm$ 0.001	4.520 $\pm$ 0.038	4.67	4.255 $\pm$ 0.30
	16		17.982 $\pm$ 0.002	5.443 $\pm$ 0.032	5.21	5.62 $\pm$ 0.18
	20.8 $\pm$ 0.1		---	0.0	18.84 ^c	20.975 $\pm$ 1.30 ^d
26.5	18	12.55 $\pm$ 0.35	18.045 $\pm$ 0.003	6.756 $\pm$ 0.01	6.94	7.05 $\pm$ 0.10
	20		18.039 $\pm$ 0.001	6.683 $\pm$ 0.005	6.825	6.90 $\pm$ 0.16
	32		18.034 $\pm$ 0.000	6.38 $\pm$ 0.002	6.475	6.045 $\pm$ 0.10
	18	24.72 $\pm$ 0.07	18.039 $\pm$ 0.000	6.725 $\pm$ 0.007	6.765	6.85 $\pm$ 0.11
	20		18.037 $\pm$ 0.000	6.585 $\pm$ 0.15	6.588	6.70 $\pm$ 0.19
	32		18.019 $\pm$ 0.004	6.071 $\pm$ 0.011	6.050	6.075 $\pm$ 0.16
	18	103.0 $\pm$ 0.7	18.015 $\pm$ 0.001	6.059 $\pm$ 0.001	6.190	6.215 $\pm$ 0.23
	20		18.006 $\pm$ 0.001	5.704 $\pm$ 0.003	5.845	5.925 $\pm$ 0.12
	32		17.891 $\pm$ 0.002	3.892 $\pm$ 0.012	4.20	3.92 $\pm$ 0.24

*All spheres are symmetrically positioned in solution cylinders

^aDTF Results

^bKENO Results

^cComputed solution cylinder height using the buckling factor option³

^dComputed solution cylinder height

asymmetrically positioned within the solution cylinder. These data refer to spheres which are radially symmetric in the solution cylinder but axially asymmetric. These data may be used to simulate the criticality of a sphere sinking through a tank of fissile solution and are especially valuable in determining if a symmetrically positioned sphere yields the most reactive system.

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Critical Approach and Exponential Experiments with Prototype FFTF Driver Fuel Pins in Water, C. L. Brown (PNL), C. A. Rogers (WADCO)

Material bucklings and extrapolation distances have been measured for 25.2 wt% $\text{PuO}_2\text{-U(Nat)O}_2$ fuel pins in water. The pins used in the experiments are a prototype design of driver fuel pins that will be used in the Fast Flux Test Facility. This work is part of a criticality measurements and calculational program to provide safe limits for the fabrication, storage, shipment, and reprocessing of FFTF fuels. Results are presented in Table I.

The composition and dimensions of the fuel pins used in the experiments are as follows:

Pin Dimensions		
	o.d. (in.)	Length (in.)
Fuel Column	0.195	27.25
Cladding (316 SS)	0.230	28.70

Fuel Enrichment	Fuel per Pin
25.2 wt% $\text{PuO}_2\text{-U(Nat)O}_2$	$\text{PuO}_2\text{-UO}_2$: 138.4 ± 1.3 g Pu: 30.75 ± 0.03 g U: 91.16 ± 1.03 g O: 16.49 ± 0.17 g
Plutonium Composition	Fuel Density
^{238}Pu : 0.04 ± 0.01 at. % ^{239}Pu : 86.19 ± 0.06 at. % ^{240}Pu : 11.88 ± 0.06 at. % ^{241}Pu : 1.73 ± 0.01 at. % ^{242}Pu : 0.16 ± 0.01 at. %	10.35 ± 0.09 g/cm ³ (93.34 $\pm$ 0.79% theoretical)

TABLE I

Results of Critical Approach and Exponential Experiments
with 25.2 wt% $\text{PuO}_2\text{-U(Nat)O}_2$ Pins in Water
(Fuel Height: 27.25 in.)

Lattice Pitch (in.)	Water-to-Fuel Volume Ratio	Critical Approach Experiments		Exponential Experiments		Material Buckling (m ⁻²)	Extrapolation Distance ^b (cm)
		Number of Pins for Criticality ^a	Critical Mass (kg Pu)	Number of Pins in Assembly	Relaxation Length (cm)		
0.450	4.26	4.17 ± 5	12.8	222 270	-14.59 -19.28	153.8 ± 2.7	8.01
0.600	8.60	219.2 ± 0.6	6.74	120	-13.71	172.7 ± 2.2	7.25
0.779	15.42	156.2 ± 0.8	4.80	78 90	-12.72 -15.05	160.9 ± 1.0	6.89
0.900	21.01	149.8 ± 0.6	4.61	66 78	-12.10 -14.27	138.5 ± 1.0	6.90
1.039	28.45	165.0 ± 0.2	5.07	84	-15.37	111.6 ± 0.3	6.61
1.200	38.37	232.6 ± 0.1	7.15	120	-19.67	77.0 ± 1.0	6.11
1.350	48.92	410 ± 10	12.6	---	---	~ 46.8	---

^aObtained from least-squares fit of "back-off" inverse multiplication measurements.

^bExtrapolation distances are with ± 0.25 cm.

TABLE II

Minimum Critical Parameters for 25.2 wt%
PuO₂-UO₂ Fuel Pins in Water

Plutonium Composition: 86.4 at.% ²³⁹Pu;
11.9 at.% ²⁴⁰Pu; 1.7 at.% ²⁴¹Pu
Pin Diameter: 0.20-in. pellets; 0.015-in. SS Clad

Minimum Critical Parameter—Cylindrical Geometry ^a					
Fuel Height (in.)	Mass (kg Pu)	No. of Pins	Vol (liter)	Cylinder Diam (in.)	Minimum Critical ∞ Slab Thickness (in.)
27.25 ^b	4.6	149	29.8	9.22	---
36.0 ^b	5.7	140	37.3	8.98	---
Infinite	---	127	---	8.60	3.64

^aMinimum critical cylinder diameter, volume, and slab thickness occur at a water-to-fuel volume ratio of 10; minimum critical mass and number of pins, at a water-to-fuel volume ratio of 21.5.

^bPin length used in experimental measurements.

^cPin length of FFTF fuel pins.

Experiments were performed in a 48-in.-diam × 72-in.-high water tank located in the PNL Critical Approach Facility. Using lattice plates with 0.45- and 0.60-in. triangular pitches, it was possible to measure the critical number of pins and the material bucklings for seven lattice spacings. For all but the 0.45- and 1.35-in. lattices, neutron multiplication measurements were carried out to within 96% of the critical number of pins. At this point, the neutron source was removed and a "back-off" experiment was performed, utilizing neutrons from the spontaneous fission of ²³⁸Pu and ²⁴⁰Pu as the neutron source. This technique produced a linear extrapolation to the critical number of pins to within ± 1 pin. Material bucklings were determined by the flux shape method in which the flux distribution is measured perpendicular to the source plane.

Minimum critical parameters for the FFTF driver-fuel-type pins were derived from the experimental data. Results are given in Table II. The values of primary interest are those for 36-in.-long pins, since this is the pin length currently planned for the FTR core.

Criticality of Plutonium Nitrate Solutions Containing Raschig Rings, R. C. Lloyd, E. D. Clayton (PNL)

A new Standard for the use of Raschig rings as neutron absorbers in solutions of fissile materials has been under preparation by the ANS for several years.¹ Few data, however, exist on the use of Raschig rings in systems of U and Pu.^{2,3}

Experiments were recently performed to obtain further data for use in establishing criticality safety guidelines and for checking calculational models for poisoned systems. These experiments provide the first known data on stainless-steel Raschig rings for use in criticality control and also data on borosilicate-glass Raschig rings in plutonium solutions. The stainless-steel rings contained 1 wt% boron. Two types of glass rings containing ½ wt% and 4 wt% boron were used. Plutonium nitrate solutions

containing 8.3 wt% ²³⁹Pu throughout a concentration range of 63 to 412 g Pu/liter were used in the cylindrical vessels containing the Raschig rings.

The systems examined were reflected with water. The experimental vessels used were 42 in. in height with diameters of 12, 18, and 24 in. Critical approach measurements were used to determine the critical height of the assembly containing the ½ wt% B rings, while exponential measurements were used on the other assemblies.

The variation of critical height with plutonium concentration for the 24-in.-diam assemblies containing the ½ wt% B rings is shown in Fig. 1. The critical height is very sensitive to concentration change in the 60-g Pu/liter region, but is very insensitive in the 200- to 400-g Pu/liter range, approaching a minimum between 300- to 400-g Pu/liter.

Buckling values determined for the assemblies are given in Table I. A water-reflected extrapolation length of 7 cm was assumed for determination of buckling values. The 4 wt% B glass rings poisoned the system sufficiently to produce negative buckling for lower concentrations, but the buckling was positive for the 391-g Pu/liter concentration. The stainless-steel rings resulted in a negative buckling for both of the concentrations (275, 412 g/liter) measured.

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Fig. 1. Raschig ring criticality.

TABLE I
Effect of Raschig Rings on Criticality

Material	Raschig Rings		Solution Conc (g Pu/liter)	Buckling (m ⁻²)
	Boron Content (%)	Volume Displacement (%)		
Glass	1 1/2	19.3	391	0.011135
	1 1/2	19.3	271	0.011300
	1 1/2	19.3	197	0.010912
	1 1/2	19.3	110	0.008462
	1 1/2	19.3	74	0.006024
	1 1/2	19.3	63	0.004996
	4	18.78	391	0.002
	4	18.78	271	-0.0006
	4	18.78	121	-0.006
Stainless Steel	1	27	275	-0.011
Steel	1	27	412	-0.006
None	0	0	275	0.0176

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Effect of Water Sprinklers on Array Criticality Safety Analyses, G. R. Handley (UCC-ND)

In facilities for the production and recovery of fissile material, containers commonly called "birdcages" are

often used to maintain a safe spacing between units of the fissile material. When water sprinkler heads are installed above arrays of loaded birdcages, the criticality safety analysis becomes considerably more difficult than for unmoderated arrays. Most sprinkler heads will emit water at a density of about 0.05 g/cm³; but, with such variables as overlapping and accumulation, the exact density of the water moderation varies. Therefore it should be considered optimum for criticality safety analysis. The density of optimum, interspersed, low-density water moderation is a function of many parameters. Two parameters, array size and unit shape, have been investigated.

The three curves of Fig. 1 were plotted from the results of 24 KENO¹ code calculations using Hansen Roach sixteen-energy-group cross sections.² Three mockups were devised in which the density of interspersed water moderation was varied. Curve A of Fig. 1 represents a 6 × 6 × 3 array mockup of C² cylinders.^{3,4} The C² units were 20.96-kg U(93.2) solid metal cylinders with an H/D = 0.94, as used in experiments at the Oak Ridge Critical Experiments Facility.^{3,4} The units were centered in 50.8-cm (20 in.) cubes with low-density water moderation. The array was closely reflected on the bottom by concrete (60 cm thick and extending 366 cm past the edge of the array) and on the other five sides by water (366 cm thick) at the same density as the interspersed water moderation. As shown in Fig. 1, optimum moderation of this array occurs at 7% of the full water density with a k_{eff} of about 0.94.

The mockup for Curve B was similar to that just described for Curve A except that the array size was changed to 2 × 2 × 2. Thickness of the reflectors remained the same as for the 6 × 6 × 3 array, and they touched the array. It can be seen that optimum moderation will occur at a considerably greater water density in this smaller array than was optimum for the 6 × 6 × 3 array.

Fig. 1. Neutron multiplication factor for arrays of U(93.2) metal units as a function of the moderating and reflecting water density.

The mockup for Curve C was identical to that of Curve A except for the unit shape. Instead of the solid cylinder, a hollow cylinder ($H/D = 1$) of equal mass with a 1469-cm³ central cylindrical void (this void never contains water) was used in the $6 \times 6 \times 3$ array. As can be seen, there are several differences between Curves A and C, the most important of which is that the array of solid cylinders is more reactive when dry, but the array of hollow cylinders is more reactive when both are optimally moderated. This reversing will cause considerable difficulty in choosing an appropriate array unit to be used in calculations needed to demonstrate the criticality safety of an actual plant array consisting of numerous forms of fissile material.

Using a solid cylinder in the calculation of plant problems will not suffice unless its mass is increased by an amount which will ensure subcriticality of various geometric configurations in the presence of an optimum interspersed moderation. The amount of mass increase needed when analyzing storage in 20-in. birdcages is estimated to be 10 to 20%.

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Calculated Critical Arrays of Fissile Materials, K. R. Ridgway (Atlantic Richfield)

To safely ship and store large numbers of fissile units, the critical reflected array size must be determined for each mass, spacing, and internal moderation of the subcritical unit under consideration.

This paper presents Monte Carlo calculations by both the GEM 4¹ and KENO² codes for critical arrays of 2, 3, and 4 kg plutonium metal pieces and critical arrays of 10L and 3L Dow type³ shipping containers of fissile solutions. These calculations indicate that the assumption of a water reflector, as specified for Class II and III shipments in Federal and IAEA Regulations^{4,5} would result in a critical array of metal units three times larger than one reflected by concrete. Care, then, must be exercised when shipments that meet regulations are to be stored within concrete enclosures.

GEM 4 and KENO calculations have been correlated to small critical experimental arrays of plutonium metal with good agreement, and the reactivities calculated by both codes for larger less dense arrays also agree.^{6,7} The codes have also been correlated using data from plutonium and uranium critical solution experiments.⁸ The data are presented in graphical form where the number of subcritical units in a critical cubic array is compared to the average plutonium density or unit spacing in the array. The units are plutonium metal spheres of 19.7 g/cm³ density and 5 wt% ²⁴⁰Pu. The array reflectors are 12-in.-thick Portland concrete, 2.3 g/cm³ and 6-in.-thick water at the outside cell boundaries. Calculations were made by GEM 4 and checked with KENO using 16-

Fig. 1.

group Hanson-Roach cross sections⁹ as modified for KENO. Calculations were carried to a statistical accuracy of $\pm 0.02\%$ k. The plotted curves for a k of 0.98 ± 0.02 have an 83% statistical confidence that the k for the arrays would be < 1.0 . A 2% statistical deviation in k at critical leads to a $\pm 20\%$ variance in critical number.

The Dow-type 3- and 10-liter shipping containers are 4.74-in. o.d., 0.1-in.-thick polyethylene bottles containing fissile solution within a flanged 5-in. schedule 40 or 80 pipe and centered inside one or two 55-gal drums by framework and vermiculite insulation.

The calculated minimum critical number of undamaged 10 liter containers is 150 at any plutonium concentration up to 450 g/liter and fully reflected by 12 in. of concrete. This most reactive configuration is a two-tier array. For a single tier array, calculations indicate an infinite array would be subcritical.

The minimum critical number of 3-liter containers of solutions with fissile material concentrations up to 450 g/liter was calculated to be 2250 containers for plutonium nitrate and 600 containers for ²³³U nitrate. The most reactive subcritical configuration is an array $15 \times 15 \times 10$ for plutonium and $10 \times 10 \times 6$ for ²³³U.

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Surface Density as a Useful Parameter in the Storage and Handling of Fissile Metal,
S. J. Altschuler, C. L. Schuske (Dow)

Surface density, the allowable weight of fissionable material per unit, floor, or shelf area of the storage

space, is a simple, useful method in determining safe storage arrays in fissile process plants.

KENO¹ calculations have been made for plutonium (96% ²³⁹Pu, 4% ²⁴⁰Pu) and fully enriched uranium (93.4% ²³⁵U) metal spheres and cubes on a square mesh. The cross sections used were the Hansen-Roach B Set² with the exception³ of H, Ca, and Si. These calculations are compared with previously used experimental data derived by *in situ* measurements.

The arrays were either bare or semi-reflected by being placed upon an infinitely thick (6 in. sufficed) concrete

Fig. 1. Relationship between critical surface density and surface-to-volume ratio of infinite square mesh arrays on an infinite concrete slab.

Fig. 2. Relationship between critical surface density and surface-to-volume ratio of unreflected, infinite square mesh arrays.

slab. Arrays $30 \times 30 \times 1$ units in size were large enough to avoid effects due to leakage from the edges of the arrays for the bare and the larger of the semi-reflected cases. For the semi-reflected arrays, the units are smaller and the leakage area is essentially halved so that leakage from the array edges becomes significant. So that valid comparisons can be made, it is necessary that infinite arrays be used. This requires $200 \times 200 \times 1$ arrays in these cases.

Figures 1 and 2 illustrate the results of an attempt to correlate the surface density at which the array is critical with the surface-to-volume ratio of the identical units comprising it. The results for semi-reflected uranium spheres agree very well with values¹ upon which this concept was based in 1957. These plots are useful since conditions represented by points underneath the curve are safe (non-critical). One has only to measure the dimensions of a storage unit, calculate its surface-to-volume ratio, weigh it or know its density, and employ these curves to obtain the allowable spacing. Extension of this method would seem to be indicated for other fissile material; for example, oxides, alloys, or aqueous solutions. The application of this method is discussed.

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Safe Fissile Material Spacing in Water, R. D. Carter (Atlantic Richfield)

At the Hanford plant, a variety of irradiated fuel elements require safe storage prior to reprocessing. Arrays of such subcritical units may be safely stored in water if a water gap between units, sufficient to inhibit excess neutron interaction, is provided. The size of the gap is limited by the composition of the unit, the k -effective of the unit and the maximum k -effective permitted for the array. In the past, a computer calculation has often been necessary to determine array safety but a simple relationship for determining safe unit spacing for less than complete isolation has been developed which can be used in many studies and which can reduce computer calculation.

If the k -effective of an array, k_a , is shown graphically as a function of the edge-to-edge spacing, a curve for one type of unit can be widely different from that for another (especially for different unit k -effective values). However, if the ratio $(k_a - k_u)/(k_\infty - k_u)$ versus the spacing is plotted, where k_u is the k -effective of an isolated unit and k_∞ the k -infinity of the material in the unit (or of the array at zero spacing), then for a number of fissile systems the values are found to be closely grouped. A limiting curve can then be drawn which can be used to determine safe water gaps for many types of units.

This approach is limited to the extent that for the smallest data spread the fissile systems of the units should be well moderated (small water gaps for under-moderated systems can result in k_a values greater than k_∞). However, fuel elements or containers of fuel elements which consist of slightly undermoderated lattices can be included by making the limiting curve somewhat more restrictive. A number of different infinite arrays

Fig. 1.

of infinitely long, cylindrical or square plutonium solution units and square enriched uranium rod lattices of various sizes and compositions were investigated with several different computer codes. One-dimensional plutonium cylinder calculations were performed by the DTF-4 code.¹ Two-dimensional calculations for plutonium solution units, various unclad enriched uranium rod lattices and BWR- and PWR-type fuel designs at 3.0 and 3.5% enrichment were performed with the GEM 4² and KENO³ Monte Carlo codes.

A sample of the results is shown in the graph. The limiting curve shown can be used to determine a safe, near-optimum array spacing of many types of subcritical fissile units, provided that k_a based on safe operation is set and that k_u and k_∞ are known. The curve implies that various types of fissile units may be mixed together, the limiting spacing being determined by the unit with the highest value of $(k_a - k_u)/(k_\infty - k_u)$.

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A New Small Mass Critical Configuration, R. S. Olson, M. A. Robkin (U of Wash)

The attempt to determine the lowest possible critical mass has been of interest since the concept of fission was introduced. With the increased interest in nuclear safeguards, knowledge of the lower limits of nuclear criticality has taken an added importance. Experimental work by Jarvis and Mills¹ has established a critical mass between 250 and 350 g of ^{235}U .

Based on the observation that an infinite slab of material with $\eta > 1$ immersed in an infinite nonabsorbing moderator has a zero critical thickness, a slab geometry, two-group diffusion analysis of a thin, centrally located core, treated as a group-coupling boundary condition, was undertaken to determine whether critical masses substantially below these values could be attained. Reflection and moderation was by thick regions of D_2O surrounding the core. The temperature of the core and adjacent moderator was lowered to 4°K, the normal boiling point of ^4He , to rethermalize neutrons striking the core and to take advantage of the absorption characteristics of the fuel. Loss of neutrons by transverse leakage was reduced by the use of D_2O edge reflectors, neutron savings being determined by albedo calculations. Group constants were determined by GAMTEC II.² The region of calculation was the prism defined by the fuel sheet, with neutron conservation including transverse leakage, albedo of the side reflectors and a source based on reflected neutrons and the collision probability for a reflected neutron transiting the prism.

The results of this analysis are shown in Fig. 1 for a ^{235}U - D_2O system with and without edge reflectors. The three temperatures shown are assumed values for rethermalized neutrons at the core surface. Similar results were obtained using ^{239}Pu as fuel. The minimum critical masses obtained were 35 g for ^{235}U and 22 g for ^{239}Pu .

As a check, calculations were made for a 1-D equivalent of a ^{235}U -fueled, D_2O -reflected cavity reactor reported by Stratton.³ The transition to a 1-D system was done by albedo weighting the reflector to obtain an equivalent slab reactor with one-half the cavity reactor's core surface area. The experimental core thickness was given as 2 mil, while our calculated value for the equivalent slab was 2.5 mil. This result coupled with the earlier calculations would indicate that there is a possibility of

Fig. 1. Critical mass of ^{235}U as a function of core height for assumed core neutron temperatures of 10, 20, and 30°K.

obtaining critical configurations of smaller mass than previously found.

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Implications of ANSI Criticality Safety Standard N16.1

A Criticality Safety Specialist's Views, W. S. Geiger (W-NFD)

In consideration of this standard from the point of view of the criticality safety specialist, it is necessary to consider the general scope and implications of the standard as well as the details of the individual sections.

In general, the standard is valuable in that it clarifies that it is not possible to establish safe processes by this or any other standard. The establishment of responsibilities and administrative practices which place nuclear criticality safety on par with other management functions is particularly valuable to the safety specialist.

The fact that neutron interaction between subcritical units is not within scope of the standard leaves a very significant portion of the administration and technical aspects of operations with fissile materials uncovered. Almost all significant operations would involve more than one subcritical limit quantity of SNM. Some single and multi-parameter subcritical limits for single units have been included, but standard safety factors or the actual safety factors that were used are not specified although the text requires incorporation of margins to protect against process variables. In addition, many of these subcritical limits exceed the maximum allowable size for use of the generally used criteria for spacing of subcritical units. Many processes could be completely justified in accordance with simple conservative standards which could be established from the literature. Obviously, it is not practical to rely entirely on such usually overconservative methods, and where they would impose process restrictions more sophisticated analytical methods could be used to demonstrate that the general safety criteria are satisfied. The standard, as written, can not be used to completely justify even simple processes without referring to other sources or references.

In summary, the standard provides too much information, by including the subcritical limits, for it to be strictly a standard for the administration of nuclear criticality safety, and far too little information for it to be of technical value to the criticality specialist. It is recommended that the standard be revised in one of these alternate approaches.

1. Change the scope to cover only the administration of nuclear safety and add information concerning interaction of subcritical units. Technical data could be updated by revising TID 7016, Rev. 1, Nuclear Safety Guide, to include recent developments and current practices.
2. Increase the scope of the standard to supersede TID 7016, Rev. 1, Nuclear Safety Guide, and to cover recent developments in interaction and current practices.

The recent revision of the Walsh Healy Act to incorporate ANSI standards by reference as part of the regulations clearly indicates that these standards are readily adaptable as regulations. However, if the standards do not adequately standardize on the margins of safety incorporated to protect against process variables, then specifications will be imposed through new regulations or for practical purposes through the licensing or contractor approval systems.

For these reasons, and to have standardized criteria for the criticality specialist, I would like to see the standard expanded to reflect current data and practices including such topics as:

1. Tabular data for single and multiparameter units to avoid errors in plotting and reading the graphs.
2. Interaction between subcritical units and acceptable methods of evaluation or criteria.
3. Restrictions or values for simple subcritical units to be used in the interaction evaluation.
4. Intersections of subcritical units of the same and different shapes.
5. Interaction effects of poisoned or concentration controlled units.
6. Effect of reflectors which are better than water, such as concrete and graphite, on the subcritical limits.
7. Thickness of common materials required for nuclear isolation.

A Process Engineer's Views, D. A. Orth (duPont-SRL)

Comments on the standard, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," are, of course, dependent on the reviewer's experience and the safety principles and organizational structure with which he is familiar. So, to put my comments in perspective, it will be helpful to recount briefly the separations operations and the personnel organization at the Savannah River Plant.

SRP separations facilities have processed reactor fuel and target materials in several categories:

1. uranium at ^{235}U concentrations from depleted to fully enriched
2. plutonium ranging from 99% ^{239}Pu to 99% ^{242}Pu
3. ^{233}U
4. mixtures of ^{238}Pu and ^{239}Pu , with ^{239}Pu content ranging from 15 to 90%.

SRP's processing personnel are aligned in dual organizations, one for production and the other for technical assistance. Each organization reports individually to Plant management. Operation of facilities and the meeting of forecasts are responsibilities of the Production Department. A separate Works Technical Department places emphasis on process performance and development, and on process safety. Works Technical personnel are assigned in either of two ways: in a direct assistance capacity to a specific facility, or as specialists in fields such as nuclear safety where they provide limits for new operating conditions. Direct assistance personnel request and incorporate nuclear safety controls in operating procedures, and exercise surveillance to ensure compliance. Additionally, various people in both Production and Works Technical are assigned review and internal audit functions.

In the new standard, the sections on administrative and technical practices are generally satisfactory. However, some items might be qualified. SRP's nuclear safety specialists aren't simply "advisors to supervision"; they furnish directions rather than advice. The standard permits relaxed safety criteria in shielded facilities, but we see no benefit in taking advantage of this. It might be said that if controls can be relaxed in the shielded case, they are excessively conservative for the unshielded case. As a personal point of view, an incident in a shielded area is intolerable even though it hurts no one, because it casts doubt on ability to control the operation.

Now, what does the standard mean to a process engineer?

1. By being published, the standard becomes a reference and acquires a mandatory status whether or not it is incorporated in specific regulations. Non-conformance will have to be explained, and this will promote standardization.
2. The standard should save appreciable time through use as a reference to eliminate the discussion sometimes necessary to reach decisions on permissible administrative and technical practices.
3. Although the concept of arbitrary safety factors had been abandoned at SRP years ago, this change required more calculations and justifications than the use of existing guides. Now, the standard subcritical limits will be used, and margins of safety consistent with the "nuclear characteristics of the process materials and equipment" (as also permitted) will be incorporated.
4. Finally, standardization on ways of setting limits should avoid misunderstandings about materials sent to and received from other sites.

Some additions to the standard may be advisable. Guidelines for mixtures of ^{239}Pu and low-enriched uranium, typical of plutonium-spiked fuels for thermal reactors, would be helpful. Also, another section might be added to show the effects of moderation and reflection with materials other than water. The standard does indeed note several times that "limits shall be applied only when the effects of neutron reflectors . . . are no greater than reflection by an unlimited thickness of water," but materials such as beryllium and carbon (and even lead) that can affect limits appreciably should be discussed. It must be realized that the standard will be interpreted by people with less training than those who assembled it.

A Government Regulator's Views, Thomas G. McCreless (USAEC)

To obtain criticality safety, every member in the nuclear safety chain must fulfill certain necessary requirements. This safety chain consists of the individual user, the process engineer, the criticality specialist, the production manager, and a reviewer outside of the production line. The regulator often fulfills the function of the reviewer outside of the production line.

The Atomic Energy Commission has been charged by Congress in the Atomic Energy Act of 1954 with establishing criteria for the issuance of special nuclear material licenses. Congress has specified that these criteria must be consistent with the common defense and security and the health and safety of the public. Accordingly, the AEC has promulgated rules and regulations in Title 10, Code of Federal Regulations, which specify requirements

that must be satisfied before a license is granted to a potential user. Applications for special nuclear material licenses are reviewed to determine that the applicant has the necessary qualifications to conduct the desired operations safely. The review includes a study of the technical qualifications of the applicant and members of his staff including their training and experience; and a study of the equipment, facilities, and procedures to determine if they are adequate to protect health and to minimize danger to life or property. A favorable finding in all these respects must be made before a license is issued. The license application and all subsequent correspondence becomes a legal record and is available to the public for inspection. Subsequent to the issuance of a license, other members of the regulatory organization periodically inspect the facility to ensure compliance with the license and to appraise the technical qualifications of the personnel involved.

The process of review for a special nuclear material license is time consuming, particularly for operations as involved as those encountered in a fabrication facility. A very important source of licensing delays is, however, the lack of communication. It is in this connection that the American National Standard For Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, N16.1-1969, will serve its most useful purpose. For example, in Section IV, entitled "Nuclear Criticality Safety Practices," administrative and technical practices, which are necessary for ensuring safety, are delineated. Every potential user of special nuclear material should study this section in great detail as it is an excellent starting point for the development of administrative and technical controls and procedures. The policies specified within the standard must, of course, be augmented to meet the needs of the individual company, the qualification of the personnel available, and the operations proposed. The use of Section IV of the standard will result in license applications which are more complete and more understandable, and thus use of the standard should reduce licensing delays.

Some concern about the inclusion of values was expressed during the formulation of Sections V and VI of the standard which are entitled "Single-Parameter Limits for Fissile Nuclides" and "Multiparameter Control," respectively. Fear was expressed that the regulators would impose a mass safety factor of 2.3 on these reduced values. Others argued that some users would want to use these subcritical limits without any safety margin. The values included in the standard should be viewed as merely solving the margin of uncertainty of values previously published and it is reassuring that the included values are, in fact, subcritical. It is comforting that the precautionary warning is readily evident; i.e., specifications must incorporate margins to protect against process uncertainties and variations.

It is regrettable that the standard had to be so limited in its scope as to preclude incorporation of a method, even if not precise, of handling mixtures of different fissile materials. A statement relative to the effect of the presence of other moderators such as graphite, and the limiting degree at which such moderation could be disregarded, would have been helpful.

The opening words of the foreword to the standard state that the purpose of the standard is to provide guidance. It should be borne in mind that the standard is a source of common sense suggestions; suggestions that may have to be strengthened or tempered, as necessary, to form a strong nuclear safety chain.

A Manager's Views, Lyle L. Zahn (*Atlantic Richfield*)

The previous speakers on the panel have described the effort that goes into design and licensing of a facility to handle fissile material. The plant manager, from whose viewpoint I will talk, has the responsibility to establish controls to ensure that the operations will be carried out within the criticality control specifications set by the technical group. These specifications would be expected to have similar bases as presented in ANSI Criticality Safety Standard N16.1. Typically, these controls follow the general patterns described below.

To implement the specifications, process control limits are prepared, then procedures written by engineering personnel to cover the operation. The procedures are then reviewed by operations personnel. If the engineering has been well thought out and maximum use made of engineered safeguards such as geometry control, then the controls, as described in the procedures, are more straightforward. This is the ultimate goal, to keep the complex subject of criticality safety in a workable framework.

Once the procedures have been prepared, the operating personnel must be trained. The training program should be designed to fully qualify each employee for his assignment. Testing of personnel should be performed to determine if they are adequately proficient and knowledgeable in the field of criticality control as well as the other operating skills.

During operation of the plant, deviations from procedures and changes in process conditions that affect nuclear criticality safety must be investigated and acted upon promptly. Maximum use should be made of instrumentation for early detection of process changes so that adjustments can be made before the process control limits are approached. Before process or equipment changes are made they must be evaluated as to their effect on the criticality safety of the plant. The depth of this review must be the same as that required for the initial evaluation.

Audits of the plant operation from a criticality safety viewpoint must be made to ascertain that procedures are being followed and that process conditions have not changed so as to affect the nuclear criticality safety. These audits should be conducted by individuals knowledgeable in the process and nuclear criticality safety and who, to the extent possible, are not immediately responsible for the operation. Action necessary to correct nonstandard conditions found during such an audit should be timely to make the audit effective.

In conclusion, the manager with the responsibility of operating the plant must ensure that the operations will be carried out within the criticality control specifications. Engineered safeguards for criticality control should minimize reliance on administrative control. The ANSI N16.1 Standard on nuclear criticality safety provides the basic guidelines necessary to accomplish these jobs.

Fast Critical Assembly Measurements and Analysis

Predicted Characteristics with Actual Observation for the Demonstration Reactor Benchmark Assembly ZPPR-2, R. E. Kaiser, R. J. Norris, A. P. Olson, N. C. Paik* (ANL-Idaho)

ZPPR Assembly 2 is part of the demonstration reactor benchmark critical assembly program.¹ It is a large, zoned, critical assembly designed to simulate the compositions and configurations of the LMFBR demonstration plants.

Assembly 2 is a simulated oxide-fueled core with two enrichment zones of equal volume, surrounded by a UO_2 -sodium blanket and a steel reflector. The core volume fractions are 32% oxide fuel equivalent, 40% sodium, 19% steel, and 9% void. The inner, lower enrichment zone has a single-drawer cell, with one column of Pu-U alloy per drawer, and the outer zone has a two-drawer cell with one and two columns per drawer, respectively. The enrichment ratio, based on ^{239}Pu + ^{241}Pu concentrations, is 1.51.

The core is 36 in. high, with a 16-in. axial blanket at each end, and cylindrical radii of 25.3 and 35.8 in. for the inner and outer cores, respectively, giving an overall L/D of 0.503 for the equal zone configuration. The radial blanket thickness is about 15.1 in., followed by a 4.6-in. steel reflector. Axially, there is a 5-in. steel reflector in one half and an 8-in. Na-steel reflector in the other half, the reason for the difference being materials supplies.

The initial critical configuration had a volume ratio (inner core/outer core) of 52/48, instead of the intended 50/50, which required subsequent alterations in zone size to achieve equal zone volumes. The loaded mass of the initial critical configuration was 1067.3 ± 1.1 kg (^{239}Pu + ^{241}Pu), which with a total mass correction gave a corrected mass of 1065.2 ± 1.1 kg. Similar masses for the equal zone configuration were 1019.7 ± 1.0 and 1017.9 ± 1.0 kg (^{239}Pu + ^{241}Pu), respectively.

ENDF/B (1968) data were used to generate 27-group cross-section sets via the ARC System² MC² modules assuming consistent B_1 , ultra-fine groups, isotropic scattering, and both homogeneous and heterogeneous resonance self-shielding. Cross-section sets were generated for six zones (unvoided and sodium voided): inner core, outer core, radial blanket, axial blanket (plate geometry), and both core zones in cylindrical pin geometry. Spatial heterogeneity effects were calculated using the ANL-modified code CALHET-2³ to flux-volume-weight neutron cross sections over each core cell.

For comparison, the more approximate but much faster ETOX-IDX⁴ method was used, coupled to CALHET-2 for flux-weighting, to study heterogeneity effects. Table I shows the heterogeneity calculated by each method.

The calculated initial-critical mass of 1095 kg of ^{239}Pu + ^{241}Pu agreed very closely with the measured mass of 1065 kg, closer than could be expected based on previous experience with ENDF/B (1968) for numerous other critical assemblies.

1. C. E. TILL and W. G. DAVEY, "The Demonstration Reactor Benchmark Critical Assemblies Program," *Trans. Am. Nucl. Soc.*, 13, 293 (1970).

2. B. J. TOPPEL, Ed., "The Argonne Reactor Computation (ARC) System," ANL-7332, Argonne National Laboratory (November 1967).
3. F. L. FILLMORE and L. D. FELTEN, "The CALHET-2 Heterogeneous Perturbation Theory Code and Application to ZPR-3-48," AI-69-13, Atomics International (March 1969).
4. R. W. HARDIE and W. W. LITTLE, Jr., "1DX, A One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections," BNWL-954, Battelle Northwest Labs. (March 1969).

Critical Experiment Normalization for LMFBR's, Eugene R. Specht (AI)

Significant improvements are possible in the nuclear performance of LMFBR's if the major nuclear parameters are normalized by appropriate critical experiments. Care must be taken to utilize the same basic techniques and cross-section data for analyzing pertinent experiments as are being used in the LMFBR design. Critical experiment data are currently available which can be used to normalize the reactor fissile mass, breeding ratio, control rod worth, fuel burnup decrement, sodium void worth, Doppler effect, sodium temperature coefficient, reactor fluences, and reactivity coefficients for the materials in the reactor.

Although better experimental data will be generated in the demonstration reactor benchmark program¹ and in the demonstration reactor engineering mockups, LMFBR design work currently in progress calls for the use of experimental data that are already available. Table I shows the basic design parameters of interest, the experimental information utilized, and the normalization factors that have been obtained for the design parameters. Since many of the design parameters cannot (or have not been) measured directly, experimental data were employed

TABLE I
Summary of Heterogeneity Effects on the Eigenvalue, k

	Heterogeneity Effects on k in %	
	ETOX-IDX Path	MC ² -2D Diffusion Path
Resonance self-shielding heterogeneity effect (Δk)	0.60 Pu: -0.24 238U: 0.84	0.6
Spatial heterogeneity effect (Δk) from CALHET		
Inner zone	0.98	0.96
Outer zone	0.90	0.93
Core average	0.94	0.95
Total heterogeneity effect (approximate result)		
Spatial heterogeneity effect estimate by CALHET plus resonance self-shielding heterogeneity effect	1.54	1.55
Total heterogeneity effect MC ² -Flux weighting by CALHET-2-dimensional diffusion path	---	1.9
2-D Transport effect correction 1-D Axial: 0.20 1-D Radial: 0.24	0.37	
Predicted eigenvalue	0.9937	0.9950

*On attachment from Westinghouse.

TABLE I
Normalization Experiments

Reactor Parameters	Critical Assembly Information			Overall Reactor Parameter Normalization Factors
	Critical Experiment Data Used	Assemblies Used for Measurements	Calculated Data ÷ Measured Data	
Core fissile enrichment	k_{eff}	ZPPR-2	1.024	1.042
Control rod size	$\Delta k/k$ of large rods	ZEBRA-7C	1.17	0.86
Breeding ratio	$\frac{^{235}\text{U captures}}{^{239}\text{Pu fissions}} \& \frac{^{241}\text{Pu fissions}}{^{239}\text{Pu fissions}}$	ZEBRA-7A	0.95 & 0.99	0.92
	k_{eff} α ^{239}Pu	ZPPR-2 ZPR-3-57	1.024 0.97	
Large sodium void worth	$\Delta k/k$ sodium at center $\Delta k/k$ sodium at edge	ZEBRA-7G	0.9 1.05	1.04
Fuel burnup decrement	$\Delta k/k$ ^{239}Pu & ^{241}Pu $\frac{^{235}\text{U captures}}{^{239}\text{Pu fissions}} \& \frac{^{241}\text{Pu fissions}}{^{239}\text{Pu fissions}}$	ZEBRA-7A & ECEL-16 ZEBRA-7A	1.35 & 1.87 0.95 & 0.99	1.11
	k_{eff} α ^{239}Pu	ZPPR-2 ZPR-3-57	1.024 0.97	
Isothermal Doppler constant [$T(dk/dt)$]	$\Delta k/k$ per g $\text{UO}_2/^\circ\text{C}$	ECEL-17	0.68 ^a	1.47
Tantalum rod heating	$\frac{\text{Ta absorptions}}{^{239}\text{Pu fissions}}$ $\frac{\text{Core } \gamma \text{ heating}}{\text{Total } \gamma \text{ heating}} \left(\begin{array}{l} \text{from thermo-} \\ \text{luminescent} \\ \text{detectors} \end{array} \right)$	ZEBRA-7C ECEL-17		
Flux spectra	$\phi(E)$ from proton recoil spectrometer	ZEBRA-7A	0.9 at 1 MeV (Norm at 150 keV)	1.11 at 1 MeV

^aPreliminary Data.

which measured the most significant portions of the design parameters. For instance, the integral capture-to-fission ratios (α) of ^{240}Pu and ^{241}Pu have not been measured in a fast critical assembly. Although these parameters are needed to determine the breeding ratio of an LMFBR they are not nearly as important as the core's fissile mass, ^{235}U captures/ ^{239}Pu fissions, ^{241}Pu fissions/ ^{239}Pu fissions, and the integral capture-to-fission ratio (α) of ^{239}Pu .

To minimize the errors inherent in the use of large normalization factors, the basic cross-section file employed should be adjusted. The procedure for doing this has been well worked out by Rowlands² and others, but the data now available from the plutonium-fueled ZPR's (when added to the data from ZEBRA, SNEAK, etc.) should make it possible to arrive at a much more reliable set of adjusted cross sections.

After the cross-section data set have been suitably adjusted by the use of clean "criticals," final shimming normalization factors for the integral FBR parameters (such as fissile enrichment) should be obtained by comparing to experiments that are presently in progress (and planned) for ZPPR and the VTRZ configuration of ZPR-6.¹

1. W. G. DAVEY and C. E. TILL, "The Demonstration Reactor Benchmark Program," *Trans. Am. Nucl. Soc.*, 13, 293 (1970).
2. J. L. ROWLANDS and J. D. MACDOUGALL, "The Use of Integral Measurements to Adjust Cross Sections and Predict Reactor Properties," 1969 BNES Conf., Paper 1.16.

Depleted-Uranium, Nickel, and Steel-Reflected EBR-II Critical Assemblies on ZPR-3—Experiments and Calculations, D. Meneghetti (ANL), W. P. Keeney, R. O. Vosburgh, D. G. Stenstrom (ANL-Idaho), K. E. Phillips (ANL), J. M. Gasidlo (ANL-Idaho)

²³⁵U-fueled critical assemblies having EBR-II-type core compositions and radially reflected by depleted-uranium, nickel, and steel reflectors are being studied on the ZPR-3 critical facility. These critical assemblies are directed toward obtaining experimental data relative to the effects of the various reflectors upon an EBR-II-type core and toward testing the reliability of cross sections and calculational techniques for the ~91-subassembly-size EBR-II cores. In addition, experiments carried out in these criticals enable an improved understanding of the irradiation environment in EBR-II-type cores.

ZPR-3 Assembly 60 is a homogeneous approximation of current-type EBR-II core loadings having a radial blanket rich in depleted uranium. The atom densities of the quasi-hexagonal core are: ²³⁵U - 0.00559; ²³⁸U - 0.00457; Na - 0.0103; O - 0.00214; Fe - 0.0134; Cr - 0.00343; and Ni - 0.00149 in units of 10²⁴ at./cm³. This represents a homogenization of typical numbers of uranium-fueled driver subassemblies, experimental fueled-subassemblies, experimental structural-subassemblies, and control-safety subassemblies. The ~17.5-in.-thick radial blanket has atom densities of: ²³⁵U - 0.0265; Na - 0.00456; Fe - 0.00964; Cr - 0.00254; and Ni - 0.00111. The 13-in. core height of the assembly is axially reflected at top and bottom by sodium and stainless steel contained in gaps and reflectors. The experimental critical mass is 200 kg ²³⁵U including an estimated correction for excess reactivity.

The nickel-rich-reflected ZPR-3 Assembly 61 has a similar core composition except the atom density of ²³⁵U is reduced to 0.00508 and the atom density of ²³⁸U is increased to 0.00505 to compensate for the higher worth of the nickel-rich reflector. The composition of the ~8.7-in.-thick nickel-rich reflector is: Ni - 0.0662; Na - 0.00205; Fe - 0.00662; and Cr - 0.00164. This reflector is surrounded by ~8.7-in.-thick blanket of composition as in Assembly 60. The experimental critical mass is 208 kg ²³⁵U including an estimated correction for excess reactivity.

Two-dimensional, RZ geometry, S₄ transport calculations using the DOT code result in k_{eff} values of 1.017 and 1.025, respectively, for Assemblies 60 and 61. These analyses utilized 29-energy-group cross sections based on the ENDF/B data files and energy-collapsed specifically for the regional compositions of these assemblies by the MC² code. These values include calculated corrections for the thin-plate heterogeneities.

TABLE I

²³⁵U and ²³⁸U Fission Traverse Responses for ZPR-3 Assembly 60

Region	Radial Position (cm)	²³⁵ U _f		²³⁸ U _f	
		Exp ^a	Calc	Exp ^a	Calc
Core	0.0	1.000	1.00	1.000	1.00
Core	16.6	0.859 ± 0.24%	0.85	0.835 ± 0.69%	0.83
Interface	30.5	0.571 ± 0.27%	0.54	0.370 ± 0.90%	0.36
Blanket	38.8	0.342 ± 0.32%	0.31	0.087 ± 1.2%	0.05
Blanket	60.9	0.061 ± 0.68%	0.04	0.0037 ± 3.5%	0.001

^aIndicated percent error due to counting statistics.

TABLE II

²³⁵U and ²³⁸U Fission Traverse Responses for ZPR-3 Assembly 61

Region	Radial Position (cm)	²³⁵ U _f		²³⁸ U _f	
		Exp ^a	Calc	Exp ^a	Calc
Core	0.0	1.000	1.00	1.00	1.00
Core	16.6	0.884 ± 0.73%	0.87	0.853 ± 0.73%	0.85
Interface	30.5	0.810 ± 0.75%	0.75	0.433 ± 0.90%	0.45
Ni	38.8	1.153 ± 0.68%	1.09	0.106 ± 1.1%	0.08
Blanket	60.9	0.117 ± 1.6%	0.13	0.0050 ± 3.5%	0.002

^aIndicated percent error due to counting statistics.

Comparisons of calculations with experimental values of ²³⁵U fission and of ²³⁸U fission detector responses at various radial positions are listed in Tables I and II, for Assemblies 60 and 61, respectively.

An analogous steel-rich-reflected critical ZPR-3 Assembly 62, currently under construction, will also be reported, as well as intermediate transition steps between assemblies.

Material Worth Measurements in the Benchmark Critical Assembly on ZPPR, R. E. Kaiser, R. J. Norris, N. C. Paik* (ANL-Idaho)

A series of small-sample reactivity measurements was made on ZPPR Assembly 2, the demonstration plant benchmark critical assembly, and compared with analytical results. This assembly is a two-zone system which reproduces many of the essentials of current demonstration reactor concepts, and has a much larger critical mass (1018.4 kg ²³⁸Pu + ²⁴¹Pu) and critical volume (2376.5 liter) than any other plutonium-fueled critical facility yet built.

The measurements were performed by oscillating small samples in and out of the reactor, and following the reactivity changes with a calibrated autorod. The average autorod position, ρ_k , was determined for each "in" and "out" position over 100-sec intervals, and the difference in reactivity was determined from

$$D_k = (-1)^k [\rho_k - \frac{1}{2}(\rho_{k+1} - \rho_{k-1})]$$

The errors in the measurements were in the neighborhood of one percent, depending in part upon sample size, and were evaluated according to the methods described by Bennet and Long,¹ and Kaiser and Gasidlo.² Appropriate dummies were run to correct for the sample container in each case.

The samples used were annular or cylindrical in shape. The annular samples have an o.d. of 0.390 in. with varying thicknesses, and the cylindrical samples vary in diameter, allowing for studies of sample size effects. The plutonium samples include four types, depending upon the ²⁴⁰Pu content. Samples are available with nominal 1, 11.5, 22, and 45% by weight of ²⁴⁰Pu. All samples are 2.173 in. in length, the width of a ZPPR matrix tube, to provide an average worth over a single-drawer cell.

The analysis of the measurements was accomplished using first-order perturbation theory in two dimensions, using ENDF/B-68 cross-section data and corrected for resonance self-shielding. The corrections were determined from a plot of the self-shielding factor vs σ_p , where σ_p was calculated for each sample in the cell environment. Further corrections were applied to the ¹⁰B

TABLE I
Measured and Calculated Small-Sample Reactivity Worths in ZPPR Assembly 2

Material and Sample No.	Sample Description	σ_p^a Calculated	Worth, Inhours ^b			
			1st-Order Pert. Calc.	1st-Order Pert. Corrected for Self-Shielding	Measured Worth	Calc Exp
¹⁰ B-7 ^c	Ann., 0.0094 in., 0.497 g	—	-1.17	-1.11	-0.956	1.16
¹⁰ B-5 ^c	Ann., 0.04 in., 1.837 g	—	-4.33	-3.72	-3.22	1.16
¹⁸¹ Ta-2	Cyl., 0.2 in., 18.647 g	60	-0.974	-0.725	-0.571	1.27
¹⁸¹ Ta-1	Cyl., 0.389 in., 70.504 g	35	-3.68	-2.59	-1.96	1.32
²³⁸ U-1	Cyl., 0.2 in., 19.63 g	16	-0.18	-0.155	-0.119	1.30
²³⁸ U-2	Cyl., 0.42 in., 85.7 g	13	-0.797	-0.647	-0.497	1.30
Pu-26 ^a	Cyl., 0.1 in., 4.452 g	60	0.584	0.570	0.5476	1.04
Pu-30 ^a	Cyl., 0.3 in., 38.091 g	27	5.024	4.975	4.59	1.08
Pu-16 ^d	Ann., 0.015 in., 8.239 g	146	0.876	0.866	0.780	1.11
Pu-18 ^d	Ann., 0.025 in., 13.775 g	70	1.48	1.44	1.34	1.07

^a²³⁹Pu (97.2%), ²⁴⁰Pu (1%), ²⁴¹Pu (0.05%), Al (1.1%).

^b1002 lh/% Δk/k.

^cComposition 92.85 wt% ¹⁰B enriched.

^d²³⁹Pu (72.319%), ²⁴⁰Pu (21.66%), ²⁴¹Pu (3.76%), ²⁴²Pu (0.62%), ²⁴¹Am (0.264%).

samples, for spatial self-shielding, based on factors tabulated in ANL-5800.³ First-order corrections for the effects of internal multiplication in the Pu samples were also applied.

A representative portion of the results is presented in Table I, indicating measured worth, first-order perturbation calculated worth, and the self-shielding corrections mentioned above. An adjustment of about 5% has been applied to account for the fact that the measurements were made 3 in. back from the interface, rather than precisely at core center.

1. E. F. BENNETT and R. L. LONG, "Precision Limitations in the Measurements of Small Reactivity Changes," *Nucl. Sci. Eng.*, 17, 425 (1963).
2. R. E. KAISER and J. M. GASIDLO, "Precision of Reactivity Measurements by Servo-Control Techniques in a Reactor with a Source," ANL-7610, Argonne National Lab. (1970).
3. "Reactor Physics Constants," ANL-5800, 1st ed., Argonne National Lab. (July 1963).

*On attachment from Westinghouse.

Advanced Aerospace Reactor System Technology

Neutron Spectrum Measurements in a Compact Fast-Spectrum Critical Assembly,
R. K. Paschall (AI), P. G. Klann (NASA-Lewis)

Differential neutron spectra were measured in the center of a fast-spectrum molybdenum-reflected critical assembly^{1,2} using spherical proton-recoil detectors.^{3,4} The critical assembly was built to investigate geometrical variables and reactivity effects of materials for a small, fast-spectrum conceptual reactor (lithium-cooled, ²³⁵U nitride-fueled). Measurements were made at Atomics International where the critical assembly was designed, built, and operated. Calculations were made at the NASA Lewis Research Center.

The small critical assembly had a cylindrical core 37.5 cm high, and 37 cm in diameter, with Mo reflectors 10 cm thick axially, and 7.5 cm thick radially. Fully enriched uranium fuel was contained in 247 Ta tubes 2.2 cm in diameter. The uranium was constrained in the center of each tube so that ⁷Li₃N could be placed in an annulus around it. Also, foil materials of W, Hf, and Ta (representing structural materials in the conceptual reactor) could be wrapped around the uranium.

Spectra were measured in three of the five core compositions assembled. Composition 1 had uranium only; composition 2 had uranium + ⁷Li₃N; and composition 5 had uranium + ⁷Li₃N + W + Hf + Ta. Median fission energies varied from 600 to 900 keV. Spectra were measured from 50 keV to 2 MeV. This included approximately 85% of the total flux. In composition 5, the spectrum was measured down to 20 keV using a new, unique, hard-wired digital divide circuit for two-parameter gamma discrimination. Gamma discrimination was impossible below this energy because of the small neutron-to-gamma ratio caused by the very hard spectrum. The divide circuit, with a low cost as well as high reliability and availability, is used in place of an on-line computer.

Figure 1 shows the spectrum in composition 5. The calculated spectrum⁵ was arbitrarily normalized to the measurement in the vicinity of 600 keV. Because the measurements include such a large fraction of the total flux, they provide a detailed check on the calculations. The comparison between the calculated 99-group GAM II spectra and the measurements has implications on the design approach and provides a direct overall test of

Fig. 1.

the GAM II cross sections used in design calculations. The experimental confirmation of the predicted low-flux level below 50 keV supports the design assumption that the temperature defect caused by Doppler broadening is small ($\sim 0.25\% \Delta k/k$). The good agreement in the vicinity of the 0.26-MeV ^7Li scattering resonance supports the design temperature defect calculations in connection with the ^7Li coolant expansion. Finally, the gross overall agreement shows that the design treatment of the inelastic scattering results in a generally correct spectrum.

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Fast Critical Experiments and Their Analysis—I

Initial Experimental Results from ZPR-6 Assembly 7, The Single-Zone Demonstration Reactor Benchmark Assembly, L. G. LeSage, E. M. Bohn, J. E. Marshall, R. A. Karam, C. E. Till, R. A. Lewis, M. Salvatores* (ANL)

Assembly 7 of ZPR-6 is the first and least complex of the series of three assemblies to be constructed on ZPR-6 and ZPPR as part of the Demonstration Reactor Benchmark Program.¹ It is a large, dilute plutonium-oxide-fueled single-zone core with the one-drawer unit-cell composition that matches both the inner-core composition of ZPPR Assembly 2¹ and the pin-calandria composition of the Variable Temperature Rodded Zone.² It is also the plutonium-oxide-fueled analog of the large, single-zone ²³⁵U-oxide-fueled Assembly 6A of ZPR-6.³ The blanket of Assembly 7 is depleted uranium, and the core's fertile-to-fissile ratio is ~6.5.

One of the initial objectives of the Assembly 7 experimental program was to define the nuclear characteristics

of the assembly to experimentally establish Assembly 7 as a data testing benchmark assembly. These measurements have been completed and a few of the initial results are listed in Table I.

The first data column of Table I specifies the experimental critical configuration of ZPR-6 Assembly 7 and gives the results of some of the central fission ratio and reactivity worth measurements. The fission ratios were measured with thin-walled Kirn-type fission counters placed in small voids near the center of the core. The reactivity worths, which were measured using an axial oscillator and a calibrated autorod, are the values for the plates in the normal plate unit-cell environment.

The clean geometry models of Assembly 7 are defined in Columns 2, 3, and 4 of Table I so that Assembly 7 may be used for ENDF/B data testing. The calculated heterogeneous cylinder is nearly the same as the as-built heterogeneous cylinder except that small adjustments were made to correct for irregularities in the as-built shape. The heterogeneous and homogeneous spherical models were defined to have the same calculated k-effective as the heterogeneous cylinder. All calculations

TABLE I

Summary of Initial Measurements and for ZPR-6 Assembly 7 Analysis

Description of Parameter	Measurement As-Built Critical Heterogeneous Cylinder	Heterogeneous Cylinder	Calculational Models and Calculated Results ^a	
			Heterogeneous Sphere	Homogeneous Sphere
k-effective (calculated)	-	0.982	0.982	0.982
Mass of ²³⁹ Pu + ²⁴¹ Pu, kg	1109	1107	1010	1171
Core radius, cm	80.34	80.28	87.60	92.01
Core height, cm	152.40	152.40	-	-
Radial blanket thickness, cm	33.56	33.56	33.56	33.56
Axial blanket thickness, cm	34.14	34.14	-	-
Core volume, liters	3090	3085	2820	3662
Fission counter fission ratios				
²³⁵ U/ ²³⁹ Pu	1.047 ± 0.023	1.161 ^b		
²³⁵ U/ ²³⁹ Pu	0.0215 ± 0.0002	0.0217 ^b		
²⁴⁰ Pu/ ²³⁹ Pu	0.1850 ± 0.003	0.1839 ^b		
Central reactivity worths, lh/kg ^c				
²³⁹ Pu	171 ± 5	198 ^d		
²³⁵ U	-8.8 ± 0.5	-12.8 ^d		
²³⁵ U	129 ± 3	168 ^d		
Na	-6.7 ± 0.2	-5.0 ^d		
SS (304)	-5.3 ± 0.2	-5.5 ^d		

^aAll calculations listed below were done with ENDF/B Version 1 cross sections.

^bCalculated in ZPPR Assembly 2 using 2-D diffusion theory in R-Z geometry at a location R=0, Z=8.9 cm using ENDF/B energy-self-shielded cross sections. The location of the fission counters in ZPR-6 Assembly 7 lies within the margin of an asymptotic central spectrum and should correspond closely to these calculations.

^c1% Δk/k ≈ 1034 lh.

^dFirst-order perturbation theory calculations using spatially weighted plate cross sections.

were made using either a one- or two-dimensional diffusion code and MC²-generated ENDF/B (Version 1) cross sections. For the heterogeneous calculation, the cross sections were corrected for plate heterogeneity by using equivalence theory in the resonance region and by using transport fluxes for spatial flux weighting in the unit cell. The calculated values of k -effective were 0.982 for ENDF/B Version 1 and 0.961 for ENDF/B Version 2.

In summary, it appears that the ENDF/B cross sections do not predict the critical size of Assembly 7 very well, and that the discrepancy is greater for the ENDF/B Version 2 data. Also, the $^{235}\text{U}/^{239}\text{Pu}$ fission ratio and the ^{239}Pu , ^{238}U , and ^{235}U reactivity worths are all overpredicted by 11 to 45%. It should be emphasized, however, that the Assembly 7 program is still in progress and that Table I represents only a few of the initial experimental results.

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LMFBR Demonstration Reactor Benchmark Critical Experiments Program—Initial Plate-Rod Heterogeneity Measurements, R. A. Lewis, L. G. LeSage, C. E. Till, J. E. Marshall, E. M. Bohn, M. Salvatores,* G. S. Stanford (ANL)

An initial set of measurements has been carried out to evaluate the effect on measured core physics parameters of the configurational heterogeneity differences between the normal plate-form ZPR fast critical assemblies and rodded assemblies typical of a $\text{Pu}_2\text{-PuO}_2$ -fueled LMFBR. These measurements are the first set of a series of heterogeneity measurements planned for the ANL LMFBR Demonstration Plant Critical Experiments Program¹ currently underway on ZPR-6 (Assembly 7) and ZPPR (Assemblies 2 and 3).

Direct experimental evaluation of configurational heterogeneity effects can be made by comparing the results of measurements of significant core physics parameters in the plate-form environment with the same measurements made in a rodded environment matched to the plate environment in (homogenized) composition. This has been done² in ZPR-6 Assembly 7; measurements at the center of a 10.8- × 10.8- × 24-in. rodded zone at the center of Assembly 7 were compared with central measurements in the regular uniform plate structure of Assembly 7. The rodded fuel assemblies used in the measurements are compared with the normal (nominally 2 in. sq) plate fuel drawers in Fig. 1. ZPR-6 Assembly 7 is a 3000-liter plate-form LMFBR critical assembly; 6.5:1 fertile: fissile $\text{UO}_2\text{-PuO}_2$, and 35:45:20 fuel, sodium, stainless steel. The ZPR rodded fuel assemblies³ utilize nominally $\frac{1}{8}$ -in.-diam × 6-in.-long $\text{UO}_2\text{-PuO}_2$ fuel rods; these assemblies are essentially homogeneous neutronically.

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Fig. 1. Normal ZPR plate-form fuel drawer compared with 2- × 2- × 12-in. (calandria can) rodded fuel assembly. Rodded fuel assembly also shown disassembled with associated ZPR $\frac{1}{8}$ -in.-diam × 6-in. fuel rods.

Four types of rod-plate comparison measurements were made in Assembly 7:

1. central in-cell sodium-voiding reactivity worth measurements
2. central reactivity worths of small sample of various materials
3. intra-cell $^{238}\text{U}/^{239}\text{Pu}$ capture-to-fission ratios and $^{238}\text{U}/^{239}\text{Pu}$ and $^{235}\text{U}/^{239}\text{Pu}$ fission ratios
4. direct measurement of the reactivity difference between compositionally matched 10.8-in.-sq × 24-in. central plate and rodded zones.

Of these measurements, the in-cell sodium-voiding measurements are expected to be the most sensitive to the plate-rod configuration differences. The sodium results together with selected small-sample results are shown in Table I.

Definitive conclusions on the effects of the configurational heterogeneities of the ZPR plates must await further measurements planned for later in the program. These measurements will include other plate and rod arrangements and compositions, larger rodded zones, measurements in significant flux gradients, and measurements at elevated temperatures (to 550°C in the VTRZ).³ However, the measurements reported here strongly indicate that, at least for this plate arrangement, the effects of plate heterogeneities are small, generally <5%, and very close to the calculated predictions.

TABLE I

ZPR-6 Assembly 7 Plate-Rod Configuration Heterogeneity Measurements
(all measurements at room temperature)

Sample	Measurements at Center of Plate Core		Measurements at Center of 10.8- × 10.8- × 24-in. Rodded Zone at Center of Plate Core		Measured Ratio Rod/Plate (B/A)	Calculated Ratio Rod/Plate (MC ² -ENDF/B Version 1)
	Sample Type	(A) Worth of Removing Sample (lh/kg)	Sample Type	(B) Worth of Removing Sample (lh/kg)		
Na	In plate cell; 2.2- × 2.2- × 12-in. sample	6.7 ± 0.2	In rod cell; 2.2- × 2.2- × 12-in. sample	6.72 ± 0.07	1.00 ± 0.03	0.976
Na	-	-	In rod cell; 2.2- × 2.2- × 6-in. sample	6.8 ± 0.2	-	-
Na	In plate cell; 6.5- × 6.5- × 12-in. sample	6.9 ± 0.3	In rod cell; 6.5- × 6.5- × 12-in. sample	6.4 ± 0.3	0.93 ± 0.06	-
C	Small-sample	12.2 ± 0.2	Small-sample	12.5 ± 0.3	1.02 ± 0.02	1.005
304 SS	Small-sample	5.0 ± 0.1	Small-sample	4.6 ± 0.1	0.90 ± 0.03	0.959
²³⁵ U	Small-sample	9.5 ± 0.3	Small-sample	9.0 ± 0.2	0.95 ± 0.04	0.931
²³⁹ Pu	Small-sample	-157.9 ± 0.9	Small-sample	-153.8 ± 0.6	0.975 ± 0.007	0.972
²³⁵ U	Small-sample	-124 ± 2	Small-sample	-120 ± 2	0.97 ± 0.02	0.964

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Measurement and Analysis of Space-Dependent Parameters in a Demonstration Plant Benchmark Critical, R. E. Kaiser, P. I. Amundson, R. J. Forrester, B. Newmark (ANL-Idaho)

The Demonstration Plant Benchmark Critical, ZPPR Assembly 2, is a two-zone system with equal zone volumes and an enrichment ratio of 1:5, outer zone/inner zone. Several measurements have been performed, including counter traverses, small-sample reactivity traverses, and the worth as a function of radial position of a ²⁵²Cf neutron source, which provide comparisons for analytical techniques in zoned reactors. In addition, central fission ratios have been measured and used to normalize the fission counter traverses and thus obtain fission ratios as a function of radial position.

Counter traverses were made for ¹⁰B (n, α) and for ²³⁹Pu, ²³⁵U, and ²³⁸U fission from core center through the

radial blanket and reflector regions. Agreement with calculated values, except in the radial reflector and the outer radial blanket, was quite good. Calculations were performed by two-dimensional diffusion theory with heterogeneously weighted cross sections.

Radial reactivity worth traverses were also performed using small samples, nominally 2.173 in. long × 0.4 in. in diameter. Materials analyzed in this way include enriched boron, tantalum, depleted and enriched uranium, and plutonium, plus the above-mentioned ²⁵²Cf source. The analyses of these traverses were based on two-dimensional perturbation theory, and compare quite well with the measured shape of the radial worth curve. The central worth discrepancy is still present,¹ however, and the curves must be properly normalized to compare shapes. Data for boron and plutonium are included in Table I.

The fission counter traverses described above were normalized, with the aid of central fission ratio measurements, to give ²³⁵U/²³⁹Pu and ²³⁵U/²³⁸Pu fission ratios as a function of radial position as shown in Table I. The analytical values compare well, especially for the ²³⁵U/²³⁹Pu ratio. The discrepancy in the ²³⁵U/²³⁹Pu ratio is believed to be the result of errors in the ENDF/B-68 cross sections for ²³⁵U. This conclusion is supported by the fact that six fission ratios, relative to ²³⁵U, all give about the same error and in the same direction.

All of the analyses were based on two-dimensional diffusion theory, and used ENDF/B-68 cross sections. One may conclude, from comparison of experiment and analysis, that the two-zone system can be calculated as accurately, if not as simply, as a reactor with a single core region. The primary difficulties (e.g., central worth discrepancies) are the same as those encountered in the simpler cores.

TABLE I
Space-Dependent Parameters Measured in ZPPR Assembly 2,
as a Function of Core Radius

Region	Radius, cm	^{235}U (n, f) ^{239}Pu (n, f)	^{238}U (n, f) ^{240}Pu (n, f)	Boron ^a Worth Ih/kg	Plutonium ^b Worth, Ih/kg
Inner Core	0.0	1.067 ± 3.0% (1.161 Calc)	0.0214 ± 3.0% (0.0217 Calc)	-1749.5 ± 3.1	119.7 ± 0.1
	11.039	1.071 ± 3.0%	0.0213 ± 3.0%	-1726.8 ± 20.5	118.0 ± 0.6
	22.078	1.056	0.0217	-1669.2 ± 23.0	113.5 ± 0.8
	33.116	1.071	0.0216	-1566.7 ± 17.2	107.1 ± 0.8
	44.155	1.046	0.0221	-1389.3 ± 19.3	97.0 ± 1.0
	55.194	1.026	0.0239	-1222.3 ± 20.4	85.2 ± 0.7
Outer Core	66.233	1.002	0.0290	-988.6 ± 19.5	70.2 ± 1.2
	77.323	1.009 (1.103 Calc)	0.0300 (0.0283 Calc)	-671.3 ± 21.7	51.2 ± 1.1
Blanket	99.350	1.100 ± 3.0%	0.0099 ± 3.0%	-118.1 ± 16.6	12.4 ± 0.8
	121.427	1.098	0.0029 ± 3.04%	-10.2 ± 17.0	2.8 ± 0.6
Reflector	143.505	0.742	0.0006 ± 4.88%	20.2 ± 16.8	0.6 ± 0.8

^aIh/kg of Sample, 92.85 at.% ¹⁰B.

^bIh/kg of Sample (98.926 wt% ²³⁹Pu, 1.023 wt% ²⁴⁰Pu, 0.05 wt% ²⁴¹Pu).

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Experiments and Analyses of Heterogeneous Cores and Neutron Spectra in EBR-II Critical Assemblies on ZPR-3, D. Meneghetti (ANL), W. P. Keeney, R. O. Vosburgh (ANL-Idaho), G. G. Simon, D. G. Stenstrom (ANL), K. E. Phillips, J. M. Gasidlo (ANL-Idaho)

A series of heterogeneously loaded, stainless-steel-reflected critical assemblies in which assembly matrix element compositions simulate EBR-II subassembly types has been studied on the Argonne Zero Power Facility, ZPR-3. The studies are directed toward the proposed use of a stainless-steel radial reflector in EBR-II and are an extension of the EBR-II-type homogeneous cores surrounded by depleted-uranium-rich, nickel-rich, or steel-rich radial reflectors, which have been reported.¹ Currently, the EBR-II is radially reflected by a blanket rich in depleted uranium. The configurations of the critical assembly cores were designed to obtain experimental data on the effects of the heterogeneities due to the different types of simulated EBR-II subassembly compositions both in internal core regions and in peripheral core regions near the stainless-steel reflector. Intercomparison of calculated with experimental reaction rate traverses through the differing regions allows the reliability of cross sections and calculational techniques used in EBR-II core analyses to be tested. These studies enable an improved understanding of the character of the irradiation environments in EBR-II core loadings especially as affected by presence of the differing subassembly types in various spatial relationships to each other.

The heterogeneous distributions of the critical assembly core-drawers simulating ²³⁵U-fueled driver and partial driver subassemblies, ²³⁹Pu-²³⁵U-fueled 37- and 19-pin mixed-oxide-composition experimental subassemblies, ²³⁵U-fueled control and safety subassemblies, and structural (stainless-steel) irradiation-test subassemblies are shown in Fig. 1 for ZPR-3 Assemblies 63A, 63B, and 63C. The plate loadings of the drawers simulating the 19- and the 37-pin mixed-oxide fuels had compositions of two-to-one and four-to-one ratios of ²³⁵U-to-²³⁹Pu, respectively. All three configurations are radially surrounded by an ~22-cm thickness of stainless-steel-rich reflector, which is surrounded in turn by an ~22-cm thickness of blanket rich in depleted uranium. The critical assembly core heights are 33 cm.

Assembly 63A simulates subassembly heterogeneity distributions of the types found in EBR-II core loadings. Assembly 63B simulates the effects of centrally clustering 21 of the 37-pin-type experimental oxide-fueled subassemblies. Assembly 63C, instead, has eight 19-pin

Fig. 1. Core region configurations of ZPR-3 assemblies 63A, 63B, and 63C.

oxide-fueled drawers and eight 37-pin oxide-fueled drawers located at opposite peripheries of the core adjacent to the stainless-steel radial reflector in order to study effects of the differing compositions adjacent to the reflector.

Two-dimensional, X-Y geometry, S_4 transport neutronics analyses using the DOT code to calculate the detailed reaction rate traverses of $^{235}\text{U}_f$, $^{238}\text{U}_f$, and $^{10}\text{B}_c$ through the differing subassembly types compared favorably with experimental traverses through these heterogeneities. The analyses utilized 29-energy-group neutron cross sections based on the ENDF/B data files and energy collapsed for core, reflector, and blanket regions by the MC² code. Analogous two-dimensional, 20-energy-group, S_8P_3 -approximation, gamma transport analyses also compared favorably with variations of gamma deposition obtained using thermoluminescent detectors through regions of heterogeneities; e.g., structural sub-assemblies.

Measurements of detailed neutron-flux spectra in EBR-II-type critical assemblies and calculational comparisons have direct relevance to such EBR-II problems as dosimetry and material damage. In addition to an improved understanding of the characteristics of the irradiation environment, the effectiveness of basic cross-section data and calculational techniques can be ascertained.

Proton-recoil measurements have been made at core center, in core near to the core-reflector interface, and in the reflector of ZPR-3 Assemblies 61 and 62. These are homogenized EBR-II-type cores radially reflected

by nickel and stainless steel, respectively.¹ In addition, a measurement was made at core center in ZPR-3 Assembly 63B, the heterogeneously loaded assembly which simulates the effects of centrally clustering oxide-fueled subassemblies. The experimentally determined flux spectra at three spatial positions in ZPR-3 Assembly 61 are compared with calculated spectra in Fig. 2. The calculations were performed in one-dimensional cylindrical geometry using the ANISN code in the diffusion-theory option. The 165-group cross-section set has a group width of 1/10 lethargy and was derived from the ENDF/B data files by energy-collapse specifically for the regional compositions of the assemblies by the MC² code. The complicated energy dependence, caused by the resonance scattering cross-section variations of many of the component materials, and its modification from core to reflector, is evident in both calculations and experiments. Similar comparisons have been made for the other EBR-II critical assemblies.

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Anisotropic Effects in a 4000-Liter UO_2 Fast Core, ZPR-6 Assembly 6A, R. A. Karam, G. Stanford, R. Robinson (ANL)

Anisotropic effects in relatively small fast reactors where the spectrum was hard were investigated by Bell et al.¹ In this paper, anisotropic effects in a Demonstration Plant Benchmark Critical, ZPR-6 Assembly 6A,² where the spectrum is relatively soft, were calculated for the unit-cell flux structure, the critical size, and activation distribution.

The 4000-liter Assembly 6A (core radius = 91.4 cm; height = 152.4 cm) had a 30-cm-thick depleted uranium (0.2% enrichment) blanket. Two of the experimental objectives were (a) to establish the critical size of a single zone core and blanket system, and (b) to measure the activation distribution indicative of the overall spatial distributions of the fluxes. The measured critical mass was 1798 ± 1 kg of ^{235}U .

Cross-section sets for Assembly 6A were generated in the manner described in Ref. 3 using ENDF/B Version 1; i.e., MC² homogeneous cross sections, corrected for resonance spatial self-shielding in ^{235}U and ^{238}U , and weighted with the unit-cell fluxes. The k_{eff} of the unit-cell calculated with 1-D, S_n ($n = 16$) and the anisotropic option (P_1 only) was 1% greater than that calculated in the same manner with the transport approximation. This difference disappeared, however, when the transverse dimension was made infinite; this indicated that the decrease in leakage associated with the use of Σ total instead of Σ transport was responsible for this difference. Similar difference in k_{eff} between the two methods was found for the as-built system.

The calculated and measured ^{235}U fission-rate distributions as a function of the core height are shown in Fig. 1. The measured points were obtained with ^{235}U foils placed next to the ^{235}U slab in the unit cell.³ In the blanket region the foils were placed on top of the depleted uranium blocks. The accuracy of the relative activities of the foils was 0.5 to 1.0%. The solid line was obtained with the heterogeneous core fluxes using $S_n = 4$ and the anisotropic option; foil cross sections were those of the

Fig. 2. Comparison of calculated spectra with experimental (proton-recoil) spectra at various positions in ZPR-3 Assembly 61.

Fig. 1. Measured and calculated ^{235}U fission rate distribution ZPR-6 Assembly 6A.

^{235}U in the core. The blanket cross sections were obtained with an MC^2 ($B^2 = 0$) calculation. When the cross sections of the ^{235}U in the blanket were used for the foil cross sections in the core and blanket, the same distribution was obtained as that shown in Fig. 1. The reason for this is the near perfect cancellation of the low-energy self-shielding effect by the high energy flux peaking in ^{235}U plates in the core. Essentially the same distribution was obtained with the transport approximation calculation. Similar discrepancies between measured and calculated ^{238}U fission and capture distributions were found; this indicates that the calculated buckling is greater, hence more leakage, than measurements show. The anisotropic calculation did not rectify the discrepancy with the measured activation distributions.

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Fast Critical Experiments and Their Analysis—II

The FTR-3 Critical Assembly and Experimental Program, J. W. Daughtry, R. B. Pond, R. M. Fleischman,* C. S. Swanson, G. K. Rusch, P. H. Kier (ANL)

FTR-3 was the third of a series of relatively simple critical assemblies built by Argonne National Laboratory for the purpose of obtaining data for testing the analytical methods being developed to design the Fast Test Reactor (FTR). FTR-3 was slightly more complex than the first two assemblies of the series, FTR-1¹ and FTR-2.² It had a cylindrical, two-zone PuO₂-UO₂ core surrounded by a ring of 16 simulated boron control rods. Its radial and axial reflectors were composed of nickel, stainless steel, and sodium. Whenever possible, quarter core symmetry was maintained throughout the FTR-3 experiments to simplify the analysis.

Within the constraints imposed by the requirements of symmetry and simplicity, and by the ZPR-9 facility, FTR-3 was intended to approximate the design of the FTR. The core height was 91.6 cm. The radii of the inner core zone, outer core zone, and radial reflector were 37.6, 60.3, and 85.4 cm, respectively. In addition, a 12-cm-thick stainless steel and sodium shield surrounded the radial reflector. The axial reflector thickness was 31.4 cm. The total volume of the inner and outer core zones was 1045 liters. The 16 peripheral boron control zones of the FTR-3 were approximately evenly spaced around the outer edge of the outer core. Each control zone was 11.2 liters in volume, extended the full height of the core, and were similar in size and composition to the peripheral boron control rods in the FTR design.

The measurements carried out in the assembly were:

1. Critical mass
2. Central neutron spectrum
3. ²³⁵U and ²³⁹Pu central Doppler
4. Central small-sample reactivity worths (major core constituents)
5. Radial sample worth traverses (²³⁹Pu, ²³⁸U, ¹⁰B)
6. Radial and axial reaction rate traverses (²³⁹Pu, ²³⁸U, ¹⁰B)
7. Central fission and capture rate ratios (Pu, U, ¹⁰B)
8. Reactivity worth of B₄C and tantalum control configurations
9. Simulated loop studies
10. Shutdown margin.

Experiments 2 and 3 have been reported.^{3,4} Some results obtained in 1 and 4 are included here. The assembly had a fissile mass of 542 kg of ²³⁹Pu + ²⁴¹Pu + ²³⁵U with an excess reactivity of 79 lh ($k = 1.0007$). A representative group of the small sample central worth results is listed in Table I.

Calculations of k and of some material worths were made with the two-dimensional diffusion theory and perturbation modules of the ARC System.⁵ Cross-section sets were generated for each region of the assembly via the ARC System MC² modules using ENDF/B data (Ver-

sion 1, revised). The ²³⁸U and Pu cross sections were corrected for resonance self-shielding using equivalence theory. Unit cell S_n calculations were used to account for spatial cell heterogeneities. The calculated k was 0.981. The calculated sample worths overestimate the experimental results by up to 40%. Unshielded cross sections were used for the sample worths and the effect of the sample oscillator was not considered.

TABLE I
Small Sample Central Reactivity Worths in FTR-3

Sample Material	Sample Weight, g	Specific Sample Worth	
		Measured lh/kg	Calculated, lh/kg
Fe	32.277	-5.44 ± 0.13	-7.46
Fe ₂ O ₃	5.265	-5.7 ± 1.1	-7.36
Al	11.437	-8.09 ± 0.78	-8.63
Al ₂ O ₃	15.061	-8.99 ± 0.59	-7.92
Polyethylene	4.024	441.0 ± 1.0	-
Ta	13.859	-83.73 ± 0.23	-
	5.893	-95.33 ± 0.78	-
	a	-104	-111.9
Enriched B ^b	4.193	-3350 ± 31	-
	1.837	-4027 ± 85	-
	0.956	-4322 ± 162	-
	0.497	-4528 ± 12	-
	a	-4700	-5475
Enriched U ^c	30.838	179.9 ± 5.1	-
	15.778	182.3 ± 0.7	-
	5.231	182.5 ± 3.2	-
	a	183	227.1
Depleted U ^d	38.163	-15.09 ± 0.19	-
	19.074	-16.11 ± 0.52	-
	6.315	-17.6 ± 1.5	-
	a	-17.3	-20.1
Pu (11% 240) ^e	15.121	226.2 ± 0.7	-
	8.853	223.0 ± 1.1	-
	2.706	221.7 ± 3.7	-
	a	219	276
Pu (22% 240) ^e	13.939	201.9 ± 0.4	-
	8.510	197.9 ± 0.8	-
	3.031	195.2 ± 2.4	-
	a	192	247
Pu (43% 240) ^e	13.923	155.9 ± 0.6	-
	8.508	154.6 ± 0.7	-
	3.442	151.8 ± 1.5	-
	a	151	193

^aExperimental results extrapolated to zero mass for comparison with calculated value.

^b92.88 wt% ¹⁰B.

^c0.662 wt% ²³⁴U, 93.176 wt% ²³⁵U, 0.317 wt% ²³⁶U, 5.844 wt% ²³⁸U.

^d0.233 wt% ²³⁵U, 99.767 wt% ²³⁸U.

^eCompositions of Pu samples given in Ref. 6.

*Temporarily assigned to ANL by PNL; now employed by WADCO.

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International Developments In Criticality Safety

An Early History of Criticality Safety, Roy Reider (LASL)

As early as 1940, Peierls¹ examined the physiological action of the radiations resulting from a nuclear chain reaction under steady-state conditions. He theorized that his chain reaction would emit 200 kW in the form of gamma rays and that a dangerous dose would be delivered in 10 min at 100 m and in 1 h at 250 m. While Peierls' analysis was carried out well before any quantity of fissile material was available, he nevertheless suggested a safety technique which was that the reaction vessel could be quickly destroyed by shelling before fatal consequences ensued.

The first fissile system designed to generate prompt power excursions was a chain reactor devised by Frisch² early in 1945. This was an assembly of active material ($\sim \text{UC}_2\text{H}_{10}$) which had a hole in the center; through the hole was dropped a slug of composition like the assembly. A divergent chain reaction supported by prompt neutrons alone was achieved for about 1/100 sec. This was a remote operation with a barricade for the operator. There was a sophisticated set of redundant interlocks, electrical and mechanical, for the slug and the bottom part of the assembly. "Fail safe" was achieved by relying on gravity; special latches took account of possible power failure.

At Los Alamos in the earliest days of hand assemblies of weapons components there was a cardinal safety rule³ as follows:

"Any changes in an assembly involving fissile material must be made by one person at a time, slowly and in such a way that the direction can be immediately reversed."

The two well-documented⁴ early accidents led to the development of remote-operated facilities. Subsequent misadventures involved no injury to personnel.

The United Kingdom always recognized that misadventures from ionizing radiations would result in adverse reaction, from the public and press, much disproportionate to the reaction from conventional industrial accidents.

While they placed responsibility clearly on the occupier for the safety of all work carried out on his premises they specified the occupier's duty to consult specialists.⁵ A clearance procedure by Criticality Clearance Committees was the first management technique for controlling criticality risk. As the scale of activities increased, an Authority Criticality Inspector evolved. The next step after this was to turn the responsibility for criticality clearance to the establishment requiring the clearance as the establishments developed their own criticality safety capability.

As one reviews the early history of criticality safety there is good evidence that standard industrial accident prevention techniques served well. Where there were failures, the evidence was not obscure that simple, straightforward multiple techniques of procedural control would produce the level of accident control required even by a changing social climate.

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3. R. F. BACHER, "Safety Rules for Handling Active Material at Omega," Los Alamos Scientific Laboratory Internal Document (September 19, 1945).

4. WILLIAM R. STRATTON, "A Review of Criticality Accidents," LA-3611 (September 26, 1967).

5. H. HOWELLS and J. T. DANIELS, "Organisation for Operational Control of Criticality," AHSB(S) R16, United Kingdom Atomic Energy Authority (1961).

Guidance In Nuclear Criticality Safety, J. T. Thomas (ORNL)

The need for specified margins in nuclear criticality safety practices, rather than acceptance of unevaluated subcriticality, is becoming more demanding as the nuclear industry continues to expand. A method more orderly than mere statements of subcriticality obscured by conservatism is the adoption of a basic model with provision to accommodate practical situations commonly encountered. One cannot hope to provide solutions to complex individual problems, so the need for specialists in the art of specifications will remain. There is the possibility, however, of achieving agreement on some standard configurations and examining present-day practices in an effort to gain and record formalized experience. The American Nuclear Society Standards Subcommittee 8 on "Fissionable Materials Outside Reactors" has undertaken the task of compiling a "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials."

The significant thrust of the Guide is the tabulation of water-reflected subcritical arrays of 40 different forms of fissile materials. The tabulated arrays are based upon validated calculations¹ and, as presented, are predicted as a maximum multiplication factor of 0.93 ± 0.01 . The materials are uranium and plutonium as metals and oxides containing varying amounts of water. The density of each oxide mixture was determined by assuming the volumes of oxide and water in the mixture were preserved. Only the main isotopic constituents were considered and others, present in trace quantities, were assumed not to influence the results significantly. Shown in Table I is an example of three materials from the tabulation. The arrays are cubic and are described in the table by the number of spherical units along an edge. The units are centered in cubic cells having the dimensions heading the columns that report the radii of the units required to produce the multiplication factor.

The response of the array multiplication factor to changes in the radius of the units was explored and furnishes an internally consistent relationship allowing prescribed minimum reactivity adjustments to be made for environmental conditions. In this way margins of safety are expressible in terms of the array reactivity. For example, the effects of concrete as a reflector of an array and its influence on neutron coupled arrays were investigated for several thicknesses of concrete. Also,

the magnitudes of the reactivity response to the removal of concrete separating arrays in a concrete structure and to an hydrogenous reflector external to the concrete were evaluated.

Nonspherical units and noncubic arrays were investigated by changing the shape of the unit and the cell. Within definitive limits, spherical units in cubic arrays are shown to be more reactive than other configurations of units of the same mass.

1. J. T. THOMAS, "Uranium Metal Criticality, Monte Carlo Calculations and Nuclear Criticality Safety," Y-CDC-7, UCC, Oak Ridge National Laboratory (1970).

CRAC—Experimental Study of Criticality Accidents, P. R. Lécroché (CEA-Gif-sur-Yvette), H. F. Revol (CEA-Is-sur-Tille)

It is generally assumed that criticality accidents are most likely to occur during processing of fissile solutions. Thus the Commissariat à l'Energie Atomique has initiated an experimental study of solution excursions.

The experimental assembly is housed in a shielded cell of the Station de Criticité à Valduc. The experimental vessel is enclosed in a confinement vessel providing sufficient volume to prevent hydrogen explosions and to store fission gases for activity decrease before release to the atmosphere. Other components include

TABLE I

Spherical Radius of Fissile Material Allowed in Water-Reflected Storage Arrays Having a Computed Multiplication Factor of 0.93

Number of Units Along Edge of Cubic Storage Array	Radius of Unit (cm) as a Function of Dimension of Cubic Cell (cm) in Storage Array					
	25.40	30.48	38.10	45.72	50.80	60.96
Uranium Metal: H/U = 0; 18.76 g U/cm ³ ; Uranium: 93.2 wt% ²³⁵ U, 6.8 wt% ²³⁸ U						
4	4.95	5.38	6.03	6.40	6.66	6.99
5	4.67	5.10	5.67	6.13	6.36	6.72
6	4.43	4.90	5.40	5.89	6.16	6.56
7	4.22	4.64	5.19	5.66	5.94	6.33
8	4.07	4.48	5.05	5.47	5.81	6.23
9	3.93	4.34	4.86	5.37	5.64	6.07
10	3.81	4.19	4.74	5.24	5.46	5.97
Plutonium Dioxide: H/Pu = 3; 4.708 g Pu/cm ³ ; Plutonium: 94.8 wt% ²³⁹ Pu, 5.2 wt% ²⁴⁰ Pu						
4	6.02	6.66	7.47	7.93	8.34	9.02
5	5.63	6.22	6.98	7.58	7.91	8.58
6	5.37	5.91	6.67	7.29	7.65	8.32
7	5.10	5.66	6.37	7.00	7.34	7.97
8	4.91	5.43	6.15	6.78	7.12	7.76
9	4.73	5.27	5.94	6.56	6.93	7.56
10	4.56	5.11	5.76	6.36	6.73	7.36
Uranium Dioxide: H/U = 20; 1.138 g U/cm ³ ; Uranium: 100 wt% ²³⁵ U						
4	6.23	6.93	7.86	8.52	8.85	9.59
5	5.81	6.46	7.36	7.99	8.37	9.10
6	5.53	6.14	6.99	7.63	8.01	8.76
7	5.23	5.86	6.67	7.32	7.68	8.46
8	5.03	5.65	6.38	7.11	7.53	8.23
9	4.85	5.44	6.21	6.86	7.28	8.02
10	4.69	5.27	6.02	6.63	7.08	7.73

Fig. 1. Typical evolution curves obtained during CRAC experiments.

storage vessels for purified and irradiated solutions, a mixer-settler purification bank, and a concentrator.

A predetermined supercritical volume of 93 wt% enriched uranyl nitrate is pumped at a steady flow rate into the experimental vessel. After allowing the reaction to evolve freely, it is stopped by emptying the vessel.

Different parameters (neutron and gamma fluxes, level, temperature, pressure, etc.) are recorded by means of a computer (capacity 35 parameters every 2 msec).

Experiments have been made using a 30-cm-o.d. (wall thickness 0.3 cm) and an 80-cm-o.d. (wall thickness 0.4 cm) cylinder. Table I shows some typical data obtained in these experiments. Figure 1 shows power and energy evolution during three experiments.

An empirical law, still to be confirmed, has been derived from the first experiments: The number of fissions produced when a plateau is reached is proportional to the volume of solution used and to the calculated corresponding reactivity.

An interesting dosimetry result may be given: Although the energy dissipated may be 10 times higher in the 80-cm cylinder than in the 30-cm cylinder, neutron and gamma fluxes are only twice as high.

No material damage occurred even in case of sharp peaks where some solution was expelled, but there was an important expansion volume above the solution. Vessels vented through a 5-cm-diam pipe and with a small expansion volume will be evaluated during the course of the program.

A Review of the CRAC Experiments, Robert L. Seale* (U of Ariz)

In November 1968, the Service D'Etudes de Criticité of the French Commissariat à l'Energie Atomique initiated a program designed to provide data on the consequences of exceeding criticality with various solutions of uranyl nitrate. This program, designated "Consequences Radiologiques d'un Accident de Criticité" (CRAC), included a series of experiments in which various concentration uranyl nitrate solutions were in-

*This review was largely conducted during the Summer of 1970 while the author was at the Los Alamos Scientific Laboratory.

jected into a large-diameter pipe to heights in excess of the critical height. The results of these experiments were kindly made available to the author by Dr. P. Lécorché.

Experiments with CRAC 29 are discussed here. The concentrations of ^{235}U varied from 48.15 g/liter in CRAC 04 to 298 g/liter in CRAC 25. All these experiments were conducted in a 30-cm-diam pipe with critical heights varying from 190 cm in CRAC 04 down to 28.85 cm in CRAC 25.

The normal experimental procedure is to inject solution into the pipe at a constant rate (varying from 100 liters/h in CRAC 12 to 1870 liters/h in CRAC 25). A constant rate of solution addition results initially in a linear ramp rate of reactivity addition. In all cases after initiation of fission in the solution, power increased rapidly to a peak corresponding to the overriding of excess reactivity by thermal expansion and gas formation. This initial pulse died out and was followed by a series of peaks of generally decreasing size. In all cases, solution was added through the occurrence of the first peak and, in some instances, was continued until after several peaks were observed. Eventually, fissions generated in the solution cause heating, material ejection, and gas formation such that excess reactivity is balanced and a steady power level is achieved. In most experiments, the ramp was initiated with the container holding less than a critical volume of solution. However, in CRAC 15, 16, 18, and 19, a critical volume was initially available and the system at constant low power when the reactivity addition was made. The resulting yields were relatively small. CRAC 14, 20.1, 20.2, and 20.3 were started from below delayed critical in the presence of an external neutron source. As expected,¹ peak fission rate in the first pulse varies linearly with the ramp rate of reactivity insertion.

The remaining CRAC experiments were conducted without an external source present and demonstrate the quantitative behavior one would expect for assembly in the presence of a "weak-source."² At low ramp rates (<0.05 \$/sec), all bursts have peak yields that fall near the standard kinetics family; however, as the ramp rate increases above 0.05 \$/sec, solutions assembled without an external source present show an increasing tendency to have significant delays in the initiation of the burst and resulting larger burst heights. At ramp rates approaching 1 \$/sec, bursts having peak yields ≈ 200 to 400 times that predicted by "standard kinetics" are observed (CRAC 8 and 11). CRAC 20.4 which was the same as CRAC 20.1, 20.2, and 20.3, except for the absence of the source, had a peak yield ≈ 20 times larger than the bursts in the experiments conducted in the presence of the source.

The Fuchs-Nordheim excursion model with simple temperature feedback predicts that, for short periods, peak burst yield is proportional to the square of the inverse minimum period. An excursion model based on the Fuchs-Nordheim model with a negative reactivity coefficient due to temperature, and modified to include radiolytic gas formation (dissolved gas concentration proportional to energy, nucleation rate proportional to power), was developed during the KEWB program³ which predicted that peak power would be proportional to inverse minimum period to the $3/2$ power. The agreement between this model and the CRAC experiments is very good over a range in burst peak size from 10^{16} to 10^{20} fissions/sec.

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2. G. E. HANSEN, "Assembly of Fissionable Material in the Presence of a Weak Neutron Source," *Nucl. Sci. Eng.*, 8, 709 (1960).

3. D. L. HETRICK, *Dynamics of Nuclear Reactors*, University of Chicago Press, Chicago (1971).

Criticality Incident—August 24, 1970, Windscale Works, J. T. Daniels (UKAEA-Risley), H. Howells, T. G. Hughes (UKAEA-Windscale)

At 18.15 on August 24, 1970, the criticality alarms sounded¹ in the Plutonium Recovery Plant, building B.203 and in nearby building B.277, at the Windscale Works of the United Kingdom Atomic Energy Authority, Production Group.

The two buildings were evacuated promptly and the staff assembled in the Criticality Control Center. It was quickly established that building B.277 contained no fissile material at the time and could be discounted as the source of radiation. The two men who had been working in the B.203 plant were monitored immediately on arrival at the Criticality Control Center, and this showed no personal contamination and no indication of significant irradiation. Examination of dosimeters and measurement of ^{24}Na in a whole-body monitor confirmed that neutron doses, if any, were very low. These preliminary measurements were completed by 23.45 on the same day. More extensive whole-body monitoring and subsequent investigations on August 25 confirmed that the whole-body doses received by the two men were <1 and 2 rads, respectively.

Reentry surveys by Health Physics staff commenced at 18.25 on August 24 and by 18.35 radiation levels had been confirmed as normal in building B.277. However, abnormal levels up to 200 mR/h γ had been detected in B.203 at the main control area on the third floor and 40 mR/h γ at the vessel ventilation extract filter on the second floor. Subsequent monitoring showed these levels to be decaying rapidly, indicating the presence of short-lived fission products, thus confirming that a criticality incident had occurred despite the absence of significant radiation exposure. A small release (~ 5 mCi) of gaseous and particulate fission products occurred via the vessel extract filters and stack (400 ft high), but the effects were not detectable at ground level.

The B.203 plant was designed as the Plutonium Purification Section of the original Irradiated Fuel Reprocessing Plant. In 1964 it was adapted to recover plutonium by the TBP/OK solvent extraction process from both solid and liquid residues. These residues include oxides, mixed Pu/U oxides, fluorides, nitrates, slag, and oxalate mother liquors. In addition to the residues, aqueous raffinates, solvent wash liquors, and off-specification product have been recycled from time to time. The plant consists of two parallel lines, North and South, housed in cells with walls ~ 12 in. thick. Extraction from nitrate solution is carried out in 3 pulsed and 1 static column, all of which are geometrically safe. The extraction columns are preceded by dissolver units, conditioners, and constant volume feeders (CVF), all of which are either mass or concentration limited for criticality control. Solutions are transferred from dissolvers to conditioners and conditioners to CVFs by a vacuum lift system via closed transfer vessels. The outlet of the transfer vessels to the CVFs is by a bottom drain, $\frac{1}{2}$ -in. bore, having a 25-ft-deep lute. The 2-ft-diam $\times$ 2-ft, 3-in.-high transfer vessel has convex dished ends.

A recovery program commenced on August 14 and on August 24 oxide material was being processed in the North Cell. The incident occurred as a lift of solution from a conditioner to a vacuum transfer vessel was ending. At this time the transfer vessel contained the maximum amount of solution, as the draining part of the cycle does not commence until the vacuum breaks; i.e., when the conditioning vessel has been emptied.

The excursion was small and there was insufficient irradiation of surrounding materials to permit an estimate of the yield by activation analysis. However, measurements of fission product decay rates in the vicinity and spectrometric analysis of fission products in the aqueous solution led to the conclusion that the total yield was of the order of 10^{15} fissions. Consideration of the response of the alarm systems to an incident of this size led to the conclusion that the duration of criticality was probably from 5 to 10 sec.

Post-incident investigations were conducted with great caution, due to the following:

1. Aqueous solution subsequently syphoned from the CVF contained fission products, but at a plutonium concentration of 6 g/liter this solution could not have been the main source of reactivity.
2. The yield of 10^{15} fissions was too small to have provided an inherent shutdown mechanism; e.g., boiling.

It was decided, therefore, that the bulk of the fissile material that had been critical was still within the transfer vessel and that it was likely it was only just subcritical. It was also believed that as such a small yield could only result from a narrow band of conditions, it was likely there had been some mechanism that had added reactivity incrementally. In these circumstances any small disturbance or increase of reflection could have resulted in another, possibly bigger, excursion.

Subsequent remote investigations, which include gamma and neutron monitoring, gamma radiography, intrascope photography, and location of a liquor surface with a conductivity probe, established that even though the transfer vessel should have been empty it still contained liquor to a depth of $8\frac{1}{2}$ in. associated with kilogram quantities of plutonium. This solution was removed remotely in $2\frac{1}{2}$ -liter aliquots by a suction pipe fitted with a conductivity probe inserted from above through the $\frac{1}{2}$ -in. bore vacuum line. Approximately 40 liters of liquor was removed from the vessel and analysis showed it to be solvent with a plutonium concentration of 55 g/liter. Thus, at the time of criticality, the contents of the transfer vessel were 40 liters of 20% TBP/OK solvent having 55 g/liter plutonium in solution, together with 50 liters of 7 M HNO_3 having 6 to 7 g/liter plutonium in solution. There may have been a small quantity of solids containing plutonium in the vessel but it would not have added significantly to the reactivity. The two phases, solvent and aqueous, were largely separated, the density of solvent phase being 0.96 g/ml and that of the aqueous phase 1.3 g/ml.

The source of the solvent has not been identified positively but there are several operations in which aqueous solutions containing both entrained and dissolved solvent could have been fed into the conditioners. Calculations based on solvent degradation products indicate that the solvent was almost certainly in the transfer-vessel lute system before the August 1970 campaign began, and may have accumulated over a period of one or two years.

Trials conducted on an inactive transparent replica of the system demonstrated that, if solvent is present, as aqueous solution is lifted into the transfer vessel the phases tend to separate out while the lift is in progress, and the aqueous solution forms a layer at the bottom of the transfer vessel. When the vacuum breaks, allowing the contents to drain, only the aqueous liquor drains via the lute to the CVF and the solvent remains trapped as a supernatant layer. The practice of washing out the plant periodically with acid would not have displaced the solvent from the lute but may have reduced its plutonium concentration.

A straightforward mechanism can be postulated for the buildup of plutonium once solvent is trapped in the transfer vessel. Successive batches of conditioned plutonium nitrate at 7 g/liter of plutonium would come into contact with solvent, and plutonium would be extracted into the solvent each time a transfer took place via the transfer vessel. A study of the buildup of plutonium in solvent suggests that at the plutonium concentrations and acidities that existed at the time of criticality, the last cycle probably involved a plutonium transfer from aqueous to solvent as low as 30 g, thus providing a small incremental increase of reactivity. The energy released from as few as 10^{15} fissions spread over several seconds would not have been sufficient to provide a shutdown mechanism due to the phenomena observed in other criticality incidents; viz., boiling, physical expulsion of solution, or microbubble formation. We are led to the conclusion, therefore, that the contents of the vessel passed through transient geometric configurations that involved the increase of reactivity, which produced criticality and also contained a mechanism that shut the reaction down. Observation of the replica system showed that as the aqueous phase flowed into the transfer vessel it fell as a streamlined jet into the solvent layer. In doing so there was some dispersion that generated an interface band of emulsion consisting mainly of globules of solvent in the aqueous phase. This emulsion band was ~5 cm thick below the jet. Calculations² using the MONK Monte Carlo program show that the presence of the jet of aqueous solution, having a plutonium concentration of <7 g/liter, created a "hole" in the middle of the solvent layer and so decreased its reactivity as long as the flow continued. On the other hand, the presence of the emulsion band produced a more reactive system than that which existed in the quiescent state with the phases separated after flow had ceased. Laboratory tests with the solvent removed from the transfer vessel after the incident led to an estimate of between 5 and 10 sec for the collapse of the emulsion band after the cessation of flow. Thus, it seems likely that at the cessation of flow the system became prompt critical and the shutdown mechanism was the collapse of the interface emulsion layer. The separated quiescent phases were shown to be just subcritical.

The post-incident investigation showed there had been no breach of the criticality clearance conditions. The Board of Enquiry recommended, *inter alia*, that means of detecting and removing plutonium accumulations should be provided in all vessels in the plant, other than those that are safe by shape.

The plant has now been recommissioned with neutron monitors installed on all vessels that are not safe by shape. The vessel drain lutes have also been modified to facilitate positive drainage and washout procedures.

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2. J. H. CHALMERS, "Criticality Calculations Relating to an Incident at Windscale," AHSB(S)R.195, UKAEA (1971).

Data for Criticality Safety

Criticality of Plutonium Nitrate Solution Containing Soluble Gadolinium, R. C. Lloyd, E. D. Clayton, L. E. Hansen (PNL)

Criticality data on the effect of soluble poisons are needed for more accurate establishment of nuclear safety limits in the storage and processing of fissile solutions, since few experimental data are available.¹ A research effort to provide data on poisoned systems² is continuing with experiments using gadolinium in plutonium nitrate solutions. Gadolinium was chosen for this study because of its high neutron cross section, high solubility, and compatibility in the separation process. The present effort is oriented toward providing data on the effects of gadolinium additions for nuclear safety guidance and to provide experimental benchmarks for checking calculational techniques and cross-section sets.

A series of criticality experiments was recently completed with plutonium nitrate solution homogeneously poisoned with gadolinium. The plutonium concentration in the solution was 116 g Pu/liter (²⁴⁰Pu content of 8.38 wt%); the acid molarity was 1.85. The gadolinium concentration was varied from 0 to 2.4 g Gd/liter in five equal steps. The experiments were performed in a 61.03-cm-diam, water-reflected, stainless-steel cylindrical vessel (0.079-cm wall thickness). The experimental criticality data are given in Table I. Critical bucklings obtained using an estimated extrapolation length of 7 cm are also included. The results demonstrate the effectiveness of the gadolinium as a neutron poison; the addition of as little as 2.38 g Gd/liter reduces the critical buckling by more than a factor of 3.

Criticality measurements are currently being performed with solutions to determine the relative effectiveness of the gadolinium as the spectrum hardens; the results of these measurements will be discussed also.

Multiplication factors for the experimental assemblies were computed using the KENO Monte Carlo code,³ and are given in Table I. Multigroup cross-section data (18 energy groups) used in these calculations were obtained using the EGGNIT and BRT-I codes.^{4,5} The gadolinium cross-section data, averaged by the EGGNIT and BRT-I codes, were obtained from the ENDF/B-II library. The critical experiments provide an integral check on the gadolinium cross sections.

TABLE I
Criticality of Gd₂O₃-Poisoned Plutonium Solutions*

Gd Concentration (g Gd/liter)	Critical Values				Computed k _{eff}
	Height (cm)	Volume (liter)	Mass (kg Pu)	Buckling ^a (cm ⁻²)	
0	15.44	45.17	5.240	0.017215	1.019
0.479	20.16	58.98	6.842	0.013651	1.026
0.963	25.98	75.99	8.815	0.010952	1.013
1.420	34.48	100.85	11.698	0.008679	1.020
1.922	48.28	141.24	16.383	0.006826	0.996
2.380	80.37	235.10	27.272	0.005287	1.026

*24-in.-diam, water-reflected cylindrical vessel

Chemical analysis: 116 g Pu/liter, 1.85 M acid, 238.9 g NO₃/liter total, specific gravity = 1.2552.

Isotopic analysis (wt%): ²³⁸Pu = 0.044, ²³⁹Pu = 90.677, ²⁴⁰Pu = 8.379
²⁴¹Pu = 0.851, ²⁴²Pu = 0.049

^aλ = 7 cm

The results given in Table I indicate that the effect of homogeneously distributed gadolinium poisons can be computed accurately in this well-moderated plutonium nitrate solution.

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Critical Masses for Slightly Enriched Uranium Rods in Uranyl Nitrate, D. L. Dunaway (Nat'l Lead-Ohio)

Critical parameters for low enrichment uranium rods are usually for lattices with optimum moderation and full reflection by water. When uranium rods are cleaned or dissolved in nitric acid, a more efficient neutron reflector, uranyl nitrate, is produced. This difference (increased reflector savings) should be considered in deriving safe process controls, for example, if additional batches are processed in the same bath.

Reflector savings for uranyl nitrate enriched up to 5.0% ²³⁵U and concentrated up to 1000 g U/liter surrounding spherical lattices of uranium rods of like enrichments were calculated by Battelle-Northwest Laboratory.¹ They verified their calculational techniques by experimental measurements. These reflector savings, along with earlier calculated and experimentally derived material bucklings, extrapolation lengths, and moderation ratios for water-moderated and -reflected spherical lattices of uranium rods,^{2,3} were used to make simple hand calculations of critical mass (Table I). In essence, the uranyl nitrate reflector savings were added to the bare extrapolation lengths yielding progressively lower critical mass values as enrichment and reflector concentration increased. The minimum critical mass for each enrichment and uranyl nitrate reflector concentration is shown in Table II as a percent of the water-reflected value. The effect of the uranyl nitrate in the moderating region was not considered. A study of this effect on the minimum critical volume has been made,⁴ but the effect on the minimum critical mass has not been specifically considered.

Reflector savings for water reach a maximum at ~6 in., but for uranyl nitrate the savings continue to increase significantly with thickness, particularly as enrichment and concentration increase. The results (Tables I and II) are based on 20-in.-thick uranyl nitrate reflectors, but they should be reliable for all thicknesses to 5.0% ²³⁵U with concentrations to 200 g U/liter or to 2.0% ²³⁵U with concentrations to 1000 g U/liter since the

TABLE I

Critical Mass of Uranium Metal vs Uranyl Nitrate Reflector
Concentration for Selected Enrichments and Rod Diameters

% ²³⁵ U	Rod Diameter (in.)	Critical Mass, kg U					
		Reflector Concentration, g U/liter					
		0	200	400	600	800	1000
1.03	0.5	1960	1878	1798	1727	1675	1613
1.03	0.6	1960 ^a	1875 ^a	1793 ^a	1719 ^a	1665 ^a	1602 ^a
1.03	0.75	2010	1919	1833	1756	1688	1634
2.0	0.2	180	164	144	128	112	99.1
2.0	0.22	180 ^a	164 ^a	143 ^a	127 ^a	111 ^a	98.5
2.0	0.25	181	165	144	127	111	96.6 ^a
2.0	0.3	185	168	146	129	112	98.8
3.00	2.0	435	378	300	220	114	29.2
3.00	3.0	725	639	521	395	223	72.6
3.063	0.1	76	64.5	49.9	32.8	13.9	b
3.063	0.12	76 ^a	64.4 ^a	49.7 ^a	32.5 ^a	14.0	b
3.063	0.175	78	65.8	50.5	32.8	13.3 ^a	b
3.063	0.4	118	97.6	72.4	44.0	15.6	b
5.0	0.03	35.3	24.8	4.2	b	b	b
5.0	0.05	34.5 ^a	24.0 ^a	3.5	b	b	b
5.0	0.75	35.2	24.2	3.2 ^a	b	b	b
5.0	0.1	36.8	25.3	3.3	b	b	b

^aMinimum critical mass of metal for the given enrichment and reflector concentration.

^bThe uranyl nitrate reflector would be critical at this concentration and thickness (20 in.).

TABLE II

Relationship of Minimum Critical Mass Values for
Water- and Uranyl Nitrate-Reflected Systems

% ²³⁵ U	Percent of Water Reflected Critical Mass ^a					
	Reflector Concentration, g U/liter					
	0	200	400	600	800	1000
1.03	100	96	91	88	85	82
2.0	100	91	79	71	62	54
3.063	100	85	65	43	18	b
5.0	100	70	9.3	b	b	b

^a% of water reflected Mc

$$= \frac{\text{Mc (reflected by } x \text{ g U/liter)} \cdot 100}{\text{Mc (reflected by } 0 \text{ g U/liter)}}$$

^bThe uranyl nitrate reflector would be critical at this concentration and thickness (20 in.).

Battelle-Northwest study¹ indicated reflector savings approach a maximum under these conditions. Note that at enrichments >2.1% ²³⁵U, the uranyl nitrate (reflector) can attain criticality.

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A Criticality Analysis of Enriched Uranium Metal Spheres Undergoing Dissolution, D. C. Hunt, Robert E. Rothe (Dow-Colo)

Criticality measurements and calculations on enriched (93.2 wt% ²³⁵U) uranium metal spheres symmetrically immersed in enriched (93.2 wt% ²³⁵U) uranyl nitrate solution have been presented previously.¹⁻³ The present interpretation of the same data considers the criticality safety of an enriched uranium metal sphere immersed, initially, in nonfissile liquid (e.g., acid). As the metal dissolves, the fissile concentration of the solution increases until the two interacting fissile regions together achieve criticality. Here, the uranium in solution is assumed homogeneously distributed.

This interpretation of the data was made by calculationally converting each critical experimental system to an ideal one in which the uranium spheres have the full crystal density (18.664 g/cm³) and the solution is a uranium-water mixture. These conversions (and all subsequent calculations) use the DTF⁴ computer code in the S₁₆ approximation with Hansen-Roach⁵ cross sections. Then, two end points are calculated for the data analysis plots: the first is the critical mass of a full metal sphere reflected by water; the second is the critical uranium concentration for a metal/water mixture. An example of the results obtained at this stage of the analysis is pre-

sented by the heavy line in Fig. 1. The dashed curve of Fig. 1 illustrates an application of these results. A 10-kg enriched uranium metal sphere is immersed in zero concentration solution. As the sphere dissolves, its mass decreases and the solution concentration increases accordingly (assuming thorough mixing). Criticality is achieved when the solution concentration reaches 22.7 g/liter, leaving a 9-kg metal sphere.

Fig. 1.

The assumption of a uranium/water mixture is conservative. DTF calculations on simulations of critical experimental systems in which the metal dissolves in strong nitric acid reveal that the presence of the nitrate significantly increases the critical sphere mass.

Finally, calculations (not comparable with experiment) were performed which better approximate the physical dissolution process. In these calculations, the dissolved uranium is allowed to spread through the water solution in layers of increasing radius and decreasing uranium concentration until the condition of total homogenization is reached. It is found that systems are more reactive when the dissolved material is in close proximity to the dissolving sphere.

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Reactivity Measurements on Large ^{235}U -Al Forms in H_2O , R. L. Currie, P. B. Parks, J. L. Jarriel (du Pont-SRL)

Inaccuracies in nuclear safety codes are usually determined by comparing the calculated and measured reactivities of representative lattices. Critical measurements have been reported and compared to calculations for homogeneous solutions^{1,2} and slightly enriched uranium metal and oxide lattices.^{3,4} This paper reports experiments on regularly spaced, rectangular, subcritical arrays of large ^{235}U -Al alloy fuel forms in water.

The fuel pieces used in these experiments were bare, annular U-Al alloy castings (11.21-cm i.d., 20.07-cm o.d.) and Al-clad "logs" (0.125-cm cladding on both sides of 10.22-cm-i.d., 12.12-cm-o.d. annular core) extruded from the castings. The U-Al alloy was 9.985 wt% uranium (92.2% ^{235}U) with an alloy density of 2.94 g/cm³.

The inverse relaxation length, κ , was obtained by static measurements in the Subcritical Experiment (SE) and combined with calculated values of M^2 and the transverse buckling to produce a k_{eff} (static). Pulsed experiments extracted the reactivity in dollars, $\$,$ with Gozani,⁵ Garellis-Russell,⁶ and Sjöstrand⁷ methods of analysis.

Table I summarizes the results of these subcritical experiments. The systematic differences between static and pulsed determinations of k_{eff} are probably due to the use of a calculated infinite-lattice migration area in k_{eff} (static) and the presence of harmonic distortions in k_{eff} (pulsed). These differences decrease as the true critical condition is approached.

The calculational "degree of conservatism" at critical for nuclear criticality safety codes can be determined by extrapolating subcritical measurements. Figure 1 illustrates how this was done for the MGBS-TGAN⁸ codes in these experiments. Codes such as KENO¹⁰, which compute

TABLE I

Pitch	Lattice	k_{eff} (static)	k_{eff} (pulsed)		k_{eff} (MGBS-TGAN) at k_{eff} (measured) extrapolated to 1.0
			A_v^a	Sjöstrand	
Castings (column 113 cm)					
20.1	2 × 2	0.921	0.965	0.903	1.120
	2 × 3	0.978		0.966	
20.7	2 × 2	0.933	0.978	0.913	1.123
	2 × 3	0.983		0.979	
21.3	2 × 2	0.935	0.982	0.920	1.116
	2 × 3	0.987		0.982	
22.0	2 × 2	0.937	0.977	0.930	1.105
	2 × 3	0.983		0.978	
22.6	2 × 2	0.933	0.963	0.925	1.091
	2 × 3	0.973		0.969	
Logs (column 111.8 cm)					
12.3	2 × 3	0.885	0.943	0.865	1.068
	3 × 3	0.952		0.946	
12.9	2 × 2	0.831	0.949	0.951	1.076
	2 × 3	0.892			
13.5	3 × 3	0.960	0.947	0.949	1.078
	2 × 3	0.890			
14.2	3 × 3	0.956	0.933	0.934	1.077
	2 × 3	0.880			
	3 × 3	0.944			

^a k_{eff} (A_v) values were obtained from $\$$ (A_v) (the arithmetic mean of Gozani and Garellis-Russell $\$$ values), the method suggested by Gozani.⁵

Fig. 1. Comparison of calculated and measured effective multiplication constants for castings on 20.7-cm-square pitch.

only k_{eff} should be extrapolated only with the pulsed data, since these values do not rely heavily on calculated parameters and are mostly free of systematic error.

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Critical Experiments with Mixed Oxides of Pu and U Containing 15 and 30 wt% Pu, S. R. Bierman, E. D. Clayton (PNL)

Associated with the development of Liquid Metal Fast Breeder Reactors will be the need to fabricate, process, and recover large quantities of mixed $\text{PuO}_2\text{-UO}_2$ fuels. Because of the plutonium content, these operations will involve a very large number of potential critical masses. This fuel must be handled in such a manner as to preclude criticality under normal and accident conditions, yet criticality considerations will bear heavily on the economics of the entire system. Consequently, it is important that these considerations be based on accurate, reliable data. Also, benchmark-type data on clean homogeneous Pu-U systems in well-defined simple geometries are required for testing cross-section data and evaluating computational techniques for calculating geometry and heterogeneity effects.

To provide these data, a research effort has been initiated at the Critical Mass Laboratory at Battelle-Northwest. Fuels currently of interest in the LMFBR Program Plan involve uranium enriched with between 8 and 30 wt% Pu. Consequently, the initial series of experiments was designed to obtain criticality data on homogeneous mixtures of $\text{PuO}_2\text{-UO}_2$ at concentrations that will provide data over the widest possible range in the 8 to 30 wt% enrichment region.

Calculations^{1,2} have indicated that a homogeneous mixture of PuO_2 and UO_2 containing about 30 wt% PuO_2 should exhibit a nearly constant critical volume at atomic $\text{H}/(\text{Pu} + \text{U})$ ratios below about 50, and that a mixture containing about 15 wt% PuO_2 should experience a minimum critical volume at an $\text{H}/(\text{Pu} + \text{U})$ atomic ratio of about 30. The calculations also indicated that a mixture of PuO_2 and UO_2 containing 8 wt% PuO_2 should have a minimum critical mass in the undermoderated region of the neutron spectrum at an $\text{H}/(\text{Pu} + \text{U})$ atomic ratio of about 50. Criticality measurements have been completed on the

TABLE I
Composition and Atom Densities of Experimental Fuel and Reflector Material

Chemical Composition	at./ (b cm)		
	30 wt% Pu Fuel	15 wt% Pu Fuel	Reflector
^{241}Am	3.5110×10^{-7}	4.0360×10^{-7}	---
^{239}Pu	2.5782×10^{-4}	1.9537×10^{-4}	---
^{238}Pu	0.0	0.0	---
^{240}Pu	2.2570×10^{-5}	1.7024×10^{-5}	---
^{241}Pu	1.7559×10^{-6}	1.2109×10^{-6}	---
^{242}Pu	0.0	0.0	---
O	1.9736×10^{-3}	3.0226×10^{-3}	1.4279×10^{-2}
^{238}U	6.6039×10^{-4}	1.2516×10^{-3}	---
^{235}U	1.0805×10^{-6}	1.6558×10^{-6}	---
H	4.4681×10^{-3}	4.4888×10^{-3}	5.7116×10^{-2}
C	4.5369×10^{-3}	4.4118×10^{-3}	3.5697×10^{-2}

TABLE II
Criticality Data for Homogeneous $\text{PuO}_2\text{-UO}_2\text{-Polystyrene}$ Fuel Mixtures

Reflector	Critical Dimensions, cm ^a			Critical Mass ^a	
	Length	Width	Height	kg of Pu	kg of U
Pu Enrichment = 30 wt%; Density of Fuel = 373 g (Pu + U)/liter; H/Pu + U = 47.4 H/Pu = 158.2; ²⁴⁰ Pu Content of Pu = 8.0 wt%; ²³⁵ U Content of U = 0.151 wt%					
Plexiglas	30.54 ± 0.03	30.54 ± 0.03	30.89 ± 0.02	3.23 ± 0.05	7.53 ± 0.12
Plexiglas	35.63 ± 0.04	35.63 ± 0.04	23.95 ± 0.10	3.40 ± 0.05	7.94 ± 0.12
Plexiglas	40.72 ± 0.04	40.72 ± 0.04	20.22 ± 0.05	3.75 ± 0.05	8.76 ± 0.13
Plexiglas	50.90 ± 0.05	45.81 ± 0.05	17.14 ± 0.04	4.48 ± 0.06	10.44 ± 0.16
Plexiglas	61.08 ± 0.06	50.90 ± 0.05	15.62 ± 0.05	5.44 ± 0.08	12.69 ± 0.20
Plexiglas	61.08 ± 0.06	55.99 ± 0.06	15.12 ± 0.02	5.80 ± 0.09	13.51 ± 0.21
Plexiglas	50.90 ± 0.05	50.90 ± 0.05	16.45 ± 0.04	4.78 ± 0.08	11.14 ± 0.18
Plexiglas ^b	30.60 ± 0.02	30.60 ± 0.02	30.60 ± 0.02	3.21 ± 0.05	7.49 ± 0.12
Plexiglas ^b	—	—	10.90 ± 0.04	—	—
Bare	45.81 ± 0.05	40.72 ± 0.04	37.94 ± 0.06	7.93 ± 0.12	18.51 ± 0.78
Bare	40.72 ± 0.04	40.72 ± 0.04	42.20 ± 0.03	7.84 ± 0.11	18.30 ± 0.77
Bare	45.81 ± 0.05	50.90 ± 0.05	32.46 ± 0.02	8.48 ± 0.12	19.80 ± 0.84
Bare ^b	41.20 ± 0.05	41.20 ± 0.05	41.20 ± 0.05	7.83 ± 0.11	18.28 ± 0.77
Pu Enrichment = 14.62 wt%; Density of Fuel = 580 g (Pu + U)/liter; H/Pu + U = 30.6 H/Pu = 210.1; ²⁴⁰ Pu Content of Pu = 8.0 wt%; ²³⁵ U Content of U = 0.151 wt%					
Plexiglas	30.54 ± 0.03	40.72 ± 0.04	29.81 ± 0.15	3.14 ± 0.04	18.35 ± 0.35
Plexiglas	40.72 ± 0.04	40.72 ± 0.04	23.84 ± 0.10	3.35 ± 0.04	19.57 ± 0.37
Plexiglas	45.81 ± 0.04	50.90 ± 0.05	19.82 ± 0.11	3.92 ± 0.05	22.89 ± 0.44
Plexiglas	50.90 ± 0.05	50.90 ± 0.05	18.92 ± 0.09	4.16 ± 0.05	24.28 ± 0.46
Plexiglas	61.08 ± 0.06	50.90 ± 0.05	17.72 ± 0.09	4.67 ± 0.06	27.28 ± 0.52
Plexiglas	61.08 ± 0.06	61.08 ± 0.06	16.63 ± 0.09	5.26 ± 0.06	30.73 ± 0.59
Plexiglas ^b	33.30 ± 0.17	33.30 ± 0.17	33.30 ± 0.17	3.13 ± 0.05	18.28 ± 0.40
Plexiglas ^b	—	—	11.56 ± 0.09	—	—
Bare	40.72 ± 0.04	40.76 ± 0.17	52.39 ± 0.07	7.37 ± 0.09	43.06 ± 0.82
Bare	40.72 ± 0.04	45.86 ± 0.19	45.10 ± 0.06	7.14 ± 0.08	41.70 ± 0.79
Bare	50.90 ± 0.05	45.86 ± 0.19	36.99 ± 0.05	7.32 ± 0.08	42.75 ± 0.81
Bare ^b	43.78 ± 0.07	43.78 ± 0.07	43.78 ± 0.07	7.12 ± 0.08	41.55 ± 0.80

^aExperimentally determined corrections have been made, accounting for the reactivity effects of the cladding material, the stacking voids, and the structural supports.

^bInfinite slab thickness obtained by extrapolation of data. Cube dimensions obtained by interpolation between critical assemblies.

first of these two fuel mixtures and the experimental results and data analyses are presented in this paper. Experiments are currently being performed on the third.

Experimental data were obtained on both bare and Plexiglas-reflected rectangular parallelepipeds constructed from 2- × 2- × 2-in. $\text{PuO}_2\text{-UO}_2\text{-polystyrene}$ fuel compacts having the compositions shown in Table I. The bare critical assemblies were near-cubic in geometry to permit determining the bare critical cube size for each fuel. The reflected critical assemblies ranged from near-cubic geometry to thin slabs to permit determining the reflected critical thickness for a slab infinite in two

dimensions in addition to the reflected critical cube size of each fuel. The critical dimensions and masses are shown in Table II for the 15 and 30 wt% plutonium-enriched mixtures.

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Cross Section Testing and Measurement

Monte Carlo Analysis of Fast Critical Assemblies Using ENDF/B I and II Data, S. N. Cramer (ORNL)

Evaluation of several benchmark reactors has been made with ENDF/B Phase I and Phase II data. These calculations were made using the ESP¹ code. ESP is a Monte Carlo code designed for general reactor analysis, with an energy range of 10^{-3} eV to 15 MeV and capable of handling complicated reactor and reactor cell geometries. Input data to the code must be in the ENDF/B library tape format, and all Phase I and Phase II data applicable to reactor calculations can be utilized.

The eigenvalues calculated for three critical assemblies are shown in Table I. Diffusion theory (CITATION code) and transport theory (XSDRN code, S_4P_3) results for the spherical model of ZPR-III-48 are also shown in Table I. The ESP results of the spherical model indicate that the detail in the energy structure of the Monte Carlo method produces no significant change in the k_{eff} from more conventional methods. Whereas the unresolved resonance energy range of ^{239}Pu is insignificant in the JEZEBEL calculation, $\sim 30\%$ of the fission production in the ZPR-III-48 occurs here, and an accurate evaluation of the fission cross section of ^{239}Pu in this range is vital to the calculation. Iron data have received wide attention in the literature recently, and ZPR-III-48 has $\sim 8\%$ volume fraction of stainless steel. The iron used in these calculations was material 1020 Phase I data and material 1122 Phase II data.

In an effort to determine if the low k_{eff} for ZPR-III-48 was due to the spherical model, rather than the data, a full-core heterogeneous mockup of this reactor was run on ESP. Calculation of integral quantities such as k_{eff} , leakage, and total flux remained unchanged from those of the spherical model. However, preliminary calculations of heterogeneity effects in the ZPR-III-48 indicate up to a 3% change in k_{eff} due to the degree of homogenization of the voids in the Pu-Al and Pu-U-Mo fuel plates. A version of the ESP geometry package has been developed which will allow an exact description of these plates, as specified in Ref. 4. The results in Table I are from calculations which involve only an approximate representation of the voids and clads in the fuel plates.

All ESP calculations were run on an IBM 360/91 machine. Each JEZEBEL and GODIVA calculation took ~ 2 min for data preparation and 2 min for the random walk and analysis. The ZPR data preparation took 15 to 40 min, depending on the degree of accuracy desired in the unresolved resonance energy ranges and the ENDF/B data set used. The ZPR random walk and analysis took ~ 30 min in both cases to achieve the reported statistical accuracy. The heterogeneous calculation was done with a "fictitious scattering" transport model, which allows collision sites to be selected without the time-consuming tracking process of the neutron history across the many thin plates in the ZPR-III-48 assembly.

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Calculations of Fast Critical Assemblies Using LASL and ENDF/B Version II Data, Thomas J. Hiron, Morris E. Battat (LASL)

Considerable effort has been expended at Los Alamos toward the establishment of a consistent set of uranium and plutonium cross sections that can be used to yield accurate neutronics calculations for a wide range of fast-neutron assemblies. This work has been documented in detail in Refs. 1 through 3, and includes extensive evaluations of ^{235}U , ^{238}U , ^{239}Pu , and ^{240}Pu . The purpose of this paper is to compare neutronics calculations made with data from this LASL library and with data from the ENDF/B Version II file. Significant differences exist between the microscopic data in these two files, and this motivated the comparisons reported in this paper.

The assemblies GODIVA and JEZEBEL were chosen for the calculations; the exact specifications used for these assemblies are given in Ref. 4. The calculations were made with the DTF-IV⁵ S_N code; S_{16} angular quadrature and P_0 transport-corrected cross sections were used. The 26-group structure specified for the ENDF/B benchmark problems was used in these calculations. The LASL data were processed by the SIGMA code which is described in Ref. 3; the ENDF/B data were processed by the MC² code.⁶

Table I shows the calculated eigenvalues and central capture and fission ratios given for both cross-section sets, along with the experimental results. Both calculations predict the GODIVA k_{eff} within experimental error, although two of the three capture and fission ratios are improved with the LASL data. The largest discrepancy in the results occurs in the JEZEBEL k_{eff} calculation; the

TABLE I

Reactor	Reference for Physical Description	ENDF/B Data Phase	$k_{eff} \pm \sigma$
JEZEBEL	2	I	0.990 ± 0.009
	2	II	0.984 ± 0.007
	3	I	1.009 ± 0.006
	3	II	0.979 ± 0.007
GODIVA	2	I	1.038 ± 0.011
	2	II	1.006 ± 0.007
	3	I	1.030 ± 0.006
	3	II	1.027 ± 0.007
ZPR-III-48 (sphere)	2	I	0.964 ± 0.005
	2	II	0.951 ± 0.005
	2	II	0.955
(diffusion)	2	II	0.961
(transport)	2	II	
ZPR-III-48 (heterogeneous)	4	I	0.954 ± 0.005

TABLE I

Calculated and Measured Parameters for GODIVA and JEZEBEL

	ENDF/B Version II	Experimental Values	Los Alamos Data ^{1,2}
GODIVA			
k_{eff} (DTF-IV)	1.00189		1.0006
Corrections:			
$S_{20} - S_{16}$ - 0.0008			
P_3 - Iso. Tr. - 0.0020			
- 0.0028			
Corrected k_{eff}	0.9991	1.000 ± 0.003	0.9978
Central Capture and Fission Ratios:			
$(^{238}\text{U})_f / (^{235}\text{U})_f$	0.155	0.156 ± 0.005	0.147
$(^{239}\text{Pu})_f / (^{235}\text{U})_f$	1.36	1.42 ± 0.02	1.38
$(^{238}\text{U})_c / (^{238}\text{U})_f$	0.55	0.47 ± 0.02	0.50
JEZEBEL			
k_{eff} (DTF-IV)	0.98956		0.99956
Corrections:			
Ga effect + 0.0035			
$S_{20} - S_{16}$ - 0.0008			
P_3 - Iso. Tr. - 0.0020			
+ 0.0007			
Corrected k_{eff}	0.9903	1.000 ± 0.003	1.0003
Central Fission Ratios:			
$(^{239}\text{Pu})_f / (^{235}\text{U})_f$	1.38	1.49 ± 0.03	1.41
$(^{238}\text{U})_f / (^{235}\text{U})_f$	0.184	0.205 ± 0.008	0.183

LASL data calculates $k_{eff} = 1.0003$ (well within the experimental error), while ENDF/B is low by 1%. In Table II comparisons are made of some ^{239}Pu cross sections used in the JEZEBEL calculations; only data for the first 15 groups are given since the percentage of the total central flux is <0.03 below this level. The greatest differences can be observed in the inelastic scattering cross section which has values as much as 2.5 times higher for the ENDF/B library. Also, the LASL group-averaged transport cross sections (not shown in the table) are 5 to 12% higher than those calculated from the ENDF/B data.

1. J.-J. H. BERLIJN, R. E. HUNTER, and C. C. CREMER, "Neutron Cross Sections for ^{235}U and ^{238}U in the

Energy Range 1 keV to 14 MeV," LA-3527, Los Alamos Scientific Laboratory (1968).

2. R. E. HUNTER, J.-J. H. BERLIJN, and C. C. CREMER, "Neutron Cross Sections for ^{239}Pu and ^{240}Pu in the Energy Range 1 keV to 14 MeV," LA-3528, Los Alamos Scientific Laboratory (1968).
3. C. C. CREMER, R. E. HUNTER, J.-J. H. BERLIJN, and D. R. WORLTON, "Comparison of Calculations with Integral Experiments for Plutonium and Uranium Critical Assemblies," LA-3529, Los Alamos Scientific Laboratory (1969).
4. G. H. BEST, M. E. BATTAT, R. J. LABAUVE, and R. E. SEAMON, "Calculations of Jezebel and Godiva with Recent ENDF/B Microscopic Data," Proc. Second Intern. Conf. Nuclear Data for Reactors, Helsinki, Finland, Vol. II, p. 517 (1970).
5. K. D. LATHROP, "DTF-IV, A FORTRAN-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," LA-3373, Los Alamos Scientific Laboratory (1965).
6. B. J. TOPPEL, A. L. RAGO, and D. M. O'SHEA, "MC², A Code to Calculate Multigroup Cross Sections," ANL-7318, Argonne National Laboratory (1967).

TABLE II

Comparison of ^{239}Pu Multigroup Cross Sections Inferred From ENDF/B Version II Data and Evaluation of Hunter et al.² $E_{max} = 10 \text{ MeV}$

GRP	E_{LOW}	% Central Flux		SIG-FISSION		SIG-CAPTURE		SIG-INELASTIC	
		ENDF/B	Ref. 2	ENDF/B	Ref. 2	ENDF/B	Ref. 2	ENDF/B	Ref. 2
1	6.06	2.45	2.42	2.07	2.16	0.0013	0.0042	1.35	0.56
2	3.68	9.08	8.85	1.71	1.85	0.0026	0.0074	2.17	1.60
3	2.23	15.90	15.61	1.84	1.91	0.0053	0.0130	2.39	1.61
4	1.35	18.36	18.26	1.94	1.89	0.0112	0.0225	2.38	1.28
5	0.82	17.26	17.34	1.72	1.67	0.0236	0.0434	2.10	0.98
6	0.50	13.88	14.49	1.60	1.59	0.0509	0.0916	1.67	0.75
7	0.30	10.14	10.17	1.52	1.58	0.1221	0.1427	1.22	0.58
8	0.18	6.23	6.21	1.50	1.53	0.1825	0.1814	0.86	0.48
9	0.11	3.42	3.43	1.50	1.51	0.2104	0.2190	0.68	0.37
10	0.067	1.76	1.74	1.50	1.48	0.2342	0.2496	0.55	0.28
11	0.041	0.85	0.82	1.55	1.46	0.3105	0.3363	0.44	0.24
12	0.025	0.38	0.37	1.63	1.46	0.4748	0.5446	0.38	0.23
13	0.015	0.16	0.16	1.87	1.53	0.6849	0.7224	0.18	0.20
14	0.0091	0.07	0.07	1.90	1.78	0.9663	0.9233	0.04	0.13
15	0.0055	0.03	0.03	2.05	2.17	1.390	1.201	0.002	0.01

Measurement and Calculation of Reactor Parameters

A Basic Experimental Reactor Physics Program on UO_2 - PuO_2 - H_2O Lattices,* J. Debrue, L. Leenders, F. Motte, H. Van Den Broeck (SCK/CEN-Belgium), P. Deramaix, F. de Waegh (Belgonucléaire)

A three-year program has been conducted in the VENUS critical facility at Mol aimed at obtaining reference data for the calibration of computer codes, used for

*Sponsor: J. Goens.

plutonium recycle design calculations.¹ For that purpose, almost 3000 fuel rods were fabricated (Table I),² including about 1000 UO_2 - PuO_2 rods and 130 dismountable rods with tight tolerance pellets. The enrichments are intermediate between those of the Hanford-BNWL and the Saxton programs.

About 130 critical configurations were investigated to study systematically the most important neutronic characteristics. As illustrated by Table I, this work included integral, local, and fine-structure measurements. The available material allowed particularly interesting exper-

TABLE I
Experimental Program

A. AVAILABLE FUEL TYPES (3)					
TYPE	U^{235} ENRICHM.	$\frac{\text{Pu}^{\text{tot}}\text{O}_2}{\text{UO}_2 + \text{Pu}^{\text{tot}}\text{O}_2}$	ISOT. COMP. Pu . 239 / 240 / 241 / 242	RODS AVAILABLE	
				STAND.	DISMOUNT.
4/0	4%	0	—	1760	36
3/1	3%	1.08%	92.08 / 7.29 / 0.6 / 0.03	430	27
2/2.7	2%	2.7%	79.37 / 17.14 / 3.05 / 0.44	470	40
0.7/5	NAT. U	5.04%	79.37 / 17.14 / 3.05 / 0.44	60	5
0.7/4	NAT. U	4.37%	95.65 / 4.10 / 0.24 / 0.01	4	23
B. BASIC EXPERIMENTAL PROGRAMME					
TYPE OF STUDY	CONFIGURATION DESCRIPTION		INVESTIGATED CHARACTERISTICS		
CRITICALITY TESTS	— SQUARE AND/OR CIRCULAR GEOMETRY — 1 AND/OR 2 ZONES — 3 PITCHES — 4/0, 3/1, 2/2.7 FUELS		— CRITICAL MASS — AXIAL BUCKLING — RADIAL BUCKLING		
SUBSTITUTION TESTS	— DIFFERENT Pu FUELS IN CENTRE OF 4/0 (PITCH 13mm) — DIFFERENT FUELS IN CENTRE OF 3/1 (3 PITCHES)		— REACTIVITY EFFECTS		
BOUNDARY STUDY	— 4/0 SLAB NEXT TO 3/1 AND 2/2.7 SLABS — 2/2.7, 0.7/5 AND 0.7/4 CENTRAL SQUARE ZONES IN 4/0		— POWER DISTRIBUTION — POWER SHARING — $\text{Pu}_{239}/\text{U}_{235}$ SPECTRUM INDEX — NON-FISSILE DETECTOR SPECTRUM INDICES		
LOCAL PERTURBATION STUDY	— B_4C , Al , H_2O PERTURBATIONS IN 4/0, 3/1, 2/2.7 FUELS — ID. AT 4/0 - 3/1 BOUNDARY		— REACTIVITY EFFECTS — POWER DISTRIBUTION		
H_3BO_3 POISONED MODERATOR	4/0 SQUARE GEOMETRIES — FROM 0 TO 765 P.P.M. B — 462 P.P.M. B { — WITHOUT PERTURBATION — WITH PERTURBATIONS — WITH CENTRAL Pu ZONE — FROM 766 TO 2025 P.P.M. B		— CRITICAL MASS, β^2 — POWER DISTRIBUTION — REACTIVITY EFFECTS — k_{eff} BY SUBCRIT. MEASUREMENT		

iments, e.g., the 0.7/5 and 0.7/4 fuels, containing the same fissile plutonium amount, but having a different ^{240}Pu content. The program also included boric acid poisoned moderator configurations; for high boron concentrations the configurations were subcritical, so that a pulsed-neutron technique had to be used for reactivity determination.

The attention devoted to fuel, detectors, and all other experimental material involved, allowed achievement of very satisfactory accuracies. Typical figures are: 0.1% Δk_{eff} on critical mass, 0.02% Δk_{eff} on k_{eff} , 1% on B_{ax} with fission chamber and 2% with γ scanning, 3% on B_{rad} (γ scanning), 1.5% on relative power distributions, 2% on power sharing, 3% on spectrum indexes, and 0.03% Δk_{eff} on reactivity effects of perturbations such as absorbing rods or water gaps.

A companion paper³ is devoted to the various fission rate measurements, only results of criticality and reactivity tests are presented here, including poisoned and unpoisoned moderator configurations, and UO_2 and $\text{PuO}_2 - \text{UO}_2$ lattices (Table II). The theoretical values were obtained with Belgonucléaire's PANTHER code chain⁴; in all cases the agreement is better than the target precisions: ± 0.5 to 1% on k_{eff} and $\pm 10\%$ on absorbing rod worth.

The basic program is being completed with studies of approximate mockups of PWR⁵ and BWR plutonium recycle assemblies.

1. H. BAIRIOT and A. LHOST, "Present Status of the Belgian Programme on Plutonium Recycle in Water Cooled Reactors," BN6905-06, *Trans. Am. Nucl. Soc.*, 12, 35 (1969).

TABLE II
Criticality and Reactivity Tests

A. H ₂ O MODERATOR				
CRITICALITY & SUBSTITUTION TESTS			EXPER.	THEORY
- UO ₂ REFERENCE CONFIGURATION (4/0) - WITH PuO ₂ - UO ₂ FUEL ZONE :		1.0002	1.0020
	<div>NUMBER OF RODS TYPE</div>		<div>keff - values</div>	
	14 x 14	3/1	0.9985	1.0025
	14 x 14	2/2.7	1.0021	1.0005
	7 x 7	2/2.7	1.0005	1.0034
	7 x 7	0.7/5	1.0011	1.0014
	5 x 5	2/2.7	1.0009	1.0018
	5 x 5	0.7/5	1.0012	1.0017
5 x 5	0.7/4	1.0007	1.0013	
REACTIVITY EFFECTS (% δk/k)			EXPER.	THEORY
ABSORBING ROD PERTURBATIONS		(REF.: 0 B ₄ C)	
	1 B ₄ C ROD	} IN 4/0 FUEL	0.73 ± 0.01	0.73
	3 B ₄ C RODS		1.67 ± 0.02	1.78
	7 B ₄ C RODS		3.25 ± 0.04	3.43
	1 B ₄ C ROD	} IN 3/1 FUEL	0.49 ± 0.02	0.54
	3 B ₄ C RODS		1.20 ± 0.02	1.26
	7 B ₄ C RODS		2.43 ± 0.04	2.70
	1 B ₄ C ROD	} IN 2/2.7 FUEL	0.44 ± 0.01	0.45
	3 B ₄ C RODS		1.02 ± 0.01	1.04
	7 B ₄ C RODS		2.11 ± 0.02	2.23

B. H ₃ BO ₃ POISONED H ₂ O MODERATOR				
CRITICALITY & SUBSTITUTION TESTS			EXPER.	THEORY
462	4/0 FUEL	1.0016	1.0019
	766	4/0 FUEL	1.0009	1.0016
(ppm B)		<div>keff - values</div>	
	462	4/0 FUEL WITH 19 x 19 PuO ₂ - UO ₂ RODS	1.0000	1.0017
REACTIVITY EFFECTS (% δk/k)			EXPER.	THEORY
462 (ppm B)	1 B ₄ C ROD	0.56 ± 0.01	0.59
		3 B ₄ C RODS	1.28 ± 0.02	1.39
		7 B ₄ C RODS	2.61 ± 0.04	2.77

2. H. VAN DEN BROECK, "Inventory and Main Characteristics of Available UO_2 and UO_2 - PuO_2 Fuel for Critical Experiments," internal SCK/CEN note 63-150/70-26, to be obtained on application (May 1970).
3. J. DEBRUE, et al., "Accurate Experimental Techniques for UO_2 - PuO_2 - H_2O Lattice Evaluation," *Trans. Am. Nucl. Soc.*, 14, 379 (1971).
4. L. BINDLER, L. LEENDERS and H. VAN DEN BROECK, "Experimental and Theoretical Physics Work on Plutonium Enriched LWR Lattices," BLG 440/BN7006-03/EUR 4034 (June 1970).
5. J. DEBRUE, P. DERAMAIX and F. de WAECH, "Plutonium Recycle Studies for the SENA PWR Reactor," *Nucl. Appl. Technol.*, 9, 516 (1970).

Accurate Experimental Techniques for UO_2 - PuO_2 - H_2O Lattice Evaluation,* J. Debrue, M. De Coster, L. Leenders, H. Van Den Broeck (SCK/CEN-Belgium), L. Bindler, A. Charlier (Belgonucléaire)

For plutonium recycle design calculations, the same precisions are wanted as those required and usually achieved for uranium lattices: 0.5 to 1% on k_{eff} , 4 to 5% on power distributions and 10% on control rod worth. To check if they can be achieved by the design codes, experimental techniques should allow the determination of the involved characteristics within error margins substantially smaller than the wanted theoretical precisions.

To obtain such experimental accuracies in the frame of the program on UO_2 - PuO_2 - H_2O lattices realized in the VENUS critical facility at Mol,¹ much care was given to lattice component manufacture (grids and fuel geometry), adequate experimental device design, and precise fuel composition knowledge. When necessary, the measurement techniques used in UO_2 lattices were further developed in order to match the complications inherent to the presence of plutonium in the fuel.² As far as possible, at least two independent techniques were used for the determination of each characteristic; further technique development work was done until the agreement between their results was satisfactory.

Figure 1 gives a few examples of neutronic characteristics measured for various fuel types; the notation x/y means x% ^{235}U enriched uranium and y% PuO_2 in the mixed oxide. The presented results, selected from boundary studies, all concern reactor fuel fission rates of direct interest for code calibration.

The following experimental techniques were used:

1. for the power distribution: activation of fuel rods, fuel pellets, and fissile foils
2. for the spectrum index: activation of fissile foils
3. for the fission rate fine structure: track counting along the radius of solid-state detectors irradiated between fuel pellets and activation of concentric rings cut out from irradiated fissile foil detectors.

Fig. 1. Spatial fission rate and spectrum index dependence.

Figure 1 illustrates the achieved experimental precisions and the good agreement of the theoretical results obtained with Belgonucléaire's PANTHER code chain³: the wanted power distribution precision (Sec. 1) is reached.

To investigate the lattice spectrum in more detail, the fission rate experiments were completed by spectrum index experiments using nonfissile detectors. Before using this technique in the lattice, a calibration experiment was carried out in close contact with Gulf General Atomic laboratories. The activation technique was used to investigate the equilibrium spectrum of an erbium nitrate water solution, previously measured by the time-of-flight technique at GGA.⁴

1. J. DEBRUE, et al., "A Basic Experimental Reactor Physics Program on UO_2 - PuO_2 - H_2O Lattices," *Trans. Am. Nucl. Soc.*, 14, 377 (1971).
2. J. DEBRUE, et al., Paper SM 88/50 to the IAEA-Symposium, Plutonium as a Reactor Fuel, Brussels, STI/PUB/153, p. 85 (1967).
3. L. BINDLER, L. LEENDERS and H. VAN DEN BROECK, "Experimental and Theoretical Physics Work on Plutonium Enriched LWR Lattices," BLG 440/BN 7006-03/EUR 4034 (June 1970).
4. T. GOZANI and G. D. TRIMBLE, "Neutron Spectrum Measurements in a Homogeneous Erbium System and Erbium Cross Sections," *Nucl. Sci. Eng.*, 41, 164 (1970).

Criticality Safety—Data and Analysis

A Los Alamos Monte Carlo Criticality Code for ROVER Reactor Calculations, Clayton W. Watson (LASL)

The Nuclear Furnace (NF) is a small Los Alamos reactor for testing ROVER fuel elements under conditions similar to those in a full-power nuclear rocket reactor (core temperatures of ~ 200 to $\sim 5000^\circ\text{R}$). The water-moderated core contains ~ 50 ROVER fuel elements and is Be reflected. The reactor is neutronically both heterogeneous and three-dimensional.

Homogenized 1-D transport theory calculations¹ predict an unexpectedly large temperature coefficient of reactivity for NF, with a cold-to-hot $\Delta k_{\text{eff}} = -0.0205$ (\$2.85). This is of considerable import for upcoming Nevada tests. To make an independent estimate, and as part of a continuing effort to upgrade Monte Carlo calculational capabilities in the ROVER program, a Monte Carlo criticality code was developed² to estimate, among other things, this same temperature effect.

A generalized, three-dimensional, point cross-section criticality code resulted, based on existing Los Alamos Monte Carlo codes.^{3,4} The eigenvalue, k_{eff} , is estimated via a generation-to-generation tally of fission points, in a manner similar to that of the ORNL multigroup KENO code.⁴ The delayed-neutron β_{eff}/β is also calculated periodically, using an input delayed-neutron spectrum plus a fission-point source distribution from the previous prompt-neutron generation. A free-gas thermalization model is included with moving Maxwellian target nuclei (at a prespecified temperature for each region) and variable scattering cross sections and effective nuclear masses over the thermal range.^{5,6}

Some 30 test problems, covering a wide range of critical systems, were run to check the accuracy of the code and to examine effects on convergence rate of initial source distribution, neutrons per generation, and thermalization parameter variations.

Table I shows representative results for measured critical assemblies (measured $k_{\text{eff}} = 1$). Anomalously high k 's for Be-reflected spheres have been noted also in transport calculations.⁶ This discrepancy was not pursued here. The low k_{eff} for $\text{H}_2\text{O} + {}^{235}\text{U}$ is apparently

associated with uncertainties in the hydrogen thermalization parameters for water. A study of these parameters, based on comparisons of calculated thermal-neutron spectra with experiments, resulted in values giving generally better agreement for water systems.

To examine the NF temperature coefficient, a detailed three-dimensional Monte Carlo model was made of the NF-Zepo (a zero-power LASL critical-assembly mockup of NF). This model contained 370 space cells and 15 material compositions. A hot temperature profile corresponding to that in the transport calculations was used. The Monte Carlo calculations gave $k_{\text{eff}} = 1.000 \pm 0.004$ for a room-temperature core (measured $k_{\text{eff}} = 1.000$) and $k_{\text{eff}} = 0.978 \pm 0.004$ for the hot core, i.e., $\Delta k = -0.022 \pm 0.006$ ($-\$3.1 \pm \0.8). This is consistent with the transport calculations.

1. LASL Group N-2 Quarterly Rover Progress Report for Quarter ending May 31, 1970), Internal Memo N-2-8381.
2. The Monte Carlo codes described herein are the work of E. D. CASHWELL and G. D. TURNER, and their associates of LASL Group TD-6.
3. R. R. JOHNSTON, "A General Monte Carlo Neutronics Code," LAMS-2856, Los Alamos Scientific Laboratory (May 1963).
4. G. E. WHITESIDES and N. F. CROSS, "KENO—A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation, Nuclear Division (September 1969).
5. C. W. WATSON, "Measured ${}^{235}\text{U}$ and ${}^{238}\text{U}$ Fissioning Neutron Fluence Distributions in the Phoebus 2A Shields: Comparison with Monte Carlo Calculations," LA-4615, Los Alamos Scientific Laboratory (March 1971).
6. G. E. HANSEN, Private Communication. (Hansen-Roach 16-group cross sections were used.) Also see H. C. PAXTON, "Los Alamos Critical-Mass Data," LAMS-3067, Los Alamos Scientific Laboratory (May 1964).

TABLE I
Representative Monte Carlo Results for Measured Critical Assemblies

Critical Assembly	Initial Source Distribution	CDC-6600 Running Time (min)	Total No. Histories	Calculated k_{eff}
3-D, ${}^{235}\text{U}$ Metal Pieces (ORNL)	Poor	9.4	59,000	0.999 ± 0.003
${}^{235}\text{U}$ Sphere (LASL Godiva)	Poor	32.4	250,000	0.996 ± 0.002
${}^{235}\text{U}$ Reflected ${}^{235}\text{U}$ Sphere (LASL Flattop)	Poor	89.7	59,400	0.996 ± 0.004
${}^{235}\text{U}$ Sphere with 2-in. Be Reflector (LASL)	Poor	70.0	20,000	1.022 ± 0.002
PHOEBUS-2A ROVER Reactor (LASL)	Poor	110	19,000	(Poor convergence)
PHOEBUS-2A ROVER Reactor (LASL)	Fair	243	52,000	1.002 ± 0.004
Homogeneous H_2O and ${}^{235}\text{U}$ Sphere, H/U = 50 (Based on ORNL solution experiments)	Fair	33.4	71,700	0.983 ± 0.005

Computational Techniques for Monte Carlo and Transport Theory

The Probability Table Method for Treating Unresolved Resonances in Monte Carlo Criticality Calculations, L. B. Levitt (AI)

If, in Monte Carlo criticality calculations, one wishes to have cross-section data sets that are properties of the individual isotopes rather than group-averaged sets, one is faced with a computer storage limitation program. The major obstacle in containing such a library entirely within a high speed memory has been the lack of a method for producing such data sets in the unresolved resonance energy range.

Up to now, two methods that avoid group averaging have been available:

1. Generating a ladder of pseudo-resolved resonances and then preparing a point cross-section set from it.
2. Storing only the parameters and generating the cross sections as needed during the calculation.

The first method is hardly feasible in view of the enormous storage requirements, and furthermore would produce results based on only one ladder.

The second method would require too much computation time in fast reactor calculations since the neutrons would spend a significant fraction of their life in this energy range.

We are proposing a new approach we shall call the probability table method.

These tables, appropriately distributed through the unresolved energy range of a given isotope, consist of a probability distribution of cross sections to be used in an energy range surrounding the table energy.

The tables are generated from point data sets obtained from ladders produced about a small energy range, sufficient to contain 50 to 100 resonances, ensuring an adequate sampling of resonance interference and overlap effects while preventing significant variation in the energy-dependent average parameters. The table size is optional, dependent mainly on the application. The number of tables required for a given isotope is governed by the observed change in the tables as a function of energy. In VIM,¹ for ²³⁹Pu we now use a set of 30 tables with a table size of 10 entries. Only 1230 storage locations are required.

The probability table method² assumes that the resonance energies are so close together that the neutron enters a resonance randomly, with its pre-collision energy playing no role in determining what part of the resonance the neutron sees. This is most true in heterogeneous assemblies where the neutron is likely to have made its last collision with a different isotope. The severest test of the assumption occurs in large regions containing a single heavy isotope, but even there one would expect reasonable results.

The entries to the table ($\bar{\sigma}_J$, P_J) are defined as follows:

Let there be a set of σ_J 's, $J = 1, N$, such that $\sigma_J > \sigma_{J-1}$, $\sigma_1 \geq \sigma_{\text{Min}}$, the lowest value of $\sigma(E)$ between E_{Min} and E_{Max} , and $\sigma_N \geq \sigma_{\text{Max}}$, the highest value of $\sigma(E)$ in this energy range.

Let

$$\int_{E_{\text{Min}}}^{E_{\text{Max}}} \sigma(E) dE = \sum_{i=1}^N \int_{E_{i-1}}^{E_i} \sigma(E) dE = \sum_{i=1}^N I_i,$$

where

$$E_0 = E_{\text{Min}} \text{ and } E_N = E_{\text{Max}},$$

and let each $\sigma(E_i)$, $1 \leq i \leq N-1$, be equal to one of the σ_J 's.

Then $\sigma_{J-1} \leq \sigma(E) \leq \sigma_J$ with probability

$$P_J = \sum_{i=1}^N \Delta E_i \delta_{iJ} / (E_{\text{Max}} - E_{\text{Min}}),$$

where

$$\Delta E_i = E_i - E_{i-1}$$

and

$$\delta_{iJ} = \begin{cases} 0, & \sigma(E_i) \neq \sigma_J \\ 1, & \sigma(E_i) = \sigma_J \end{cases};$$

$$P_J = \sum_{k=1}^J p_k \text{ and } \bar{\sigma}_J = \frac{\sum_{i=1}^N I_i \delta_{iJ}}{\sum_{i=1}^N \Delta E_i \delta_{iJ}}.$$

In a Monte Carlo calculation when the cross section of an isotope in its unresolved range is needed, it is obtained by a random selection from the appropriate probability table as follows:

Select a random number, $0 < r < 1$, and set $\sigma(E) = \bar{\sigma}_J$, where

$$P_{J-1} < r \leq P_J.$$

A test problem, consisting of a 4.5-cm-diam ¹⁸¹Ta cylinder surrounded by carbon, was run on the RABBLE³ code. A table based on 30 ladders was prepared for the energy range 270 to 330 eV. RABBLE calculations were run for each of the 30 ladders, using the generated point data sets, and compared with RABBLE calculations using the probability table. The mean of the RABBLE results based on the probability table method was in statistical agreement with the mean value of the point data set calculations. Table sizes of 45 and 10 were tried with no appreciable difference in results. These results, discussed more fully in the next paper, establish the equivalence of the table method with point data from pseudo-resolved resonances, and suggest use of the method, where possible, in analytic as well as Monte Carlo calculations.

1. L. B. LEVITT and R. C. LEWIS, "VIM-1 A Non-Multigroup Monte Carlo Code for Analysis of Fast Critical Assemblies," AI-AEC-12951, Atomic International (May 15, 1970).
2. J. M. OTTER, "Application of the Probability Table Method of Treating Unresolved Resonances to an Analytical Calculation," *Trans. Am. Nucl. Soc.*, 14, 649 (1971).
3. P. H. KIER and A. A. ROBBA, "RABBLE, A Program for Computation of Resonance Absorption in Multi-region Reactor Cells," ANL-7326, Argonne National Laboratory (1967).

Application of the Probability Table Method of Treating Unresolved Resonances to an Analytical Calculation, J. M. Otter (AI)

The probability table method of treating unresolved resonances, developed for Monte Carlo calculations,¹ has been extended to and tested for use in a one-dimensional integral transport theory slowing down calculation using a modified version of the RABBLE² code.

Although a ladder of resolved resonances can be statistically generated and used directly in the RABBLE code to treat the unresolved resonance region, this procedure is generally impractical, since excessive running times and computer storage requirements are needed to process large amounts of data, large number of fine groups, and averaging of ladders. On the other hand, the probability table method requires only a small number of data storage locations, is pre-averaged over an adequate number of ladders, and allows interpolation over large energy ranges so that the number of fine groups is minimized.

Calculations were made using both the above methods for a 4.51-cm-diam tantalum rod in carbon. Mean values and standard deviations of effective cross sections and percentage cross-section changes with temperature at 300 eV are given in Table I.

Effective cross sections agree with one standard deviation even for the short table. Percentage changes with temperature agree to within $1\frac{1}{2}$ standard deviations. One-third or fewer fine groups were needed to obtain a given precision in the table runs. The number of fine groups needed to obtain 5 to 10% accuracy is economically feasible for users considering such a calculation.

This severely self-shielded test case indicates considerable benefits for this technique in analytical as well as Monte Carlo calculations. No major difficulties are foreseen in extensions to higher energies or other types of isotopes.

TABLE I

Comparisons of RABBLE Results for a Large Tantalum Rod in Carbon Between 330 and 270 eV, Using Pointwise Cross-Section Data or Cross-Section Probability Tables

	Pointwise	Table	Difference
Effective Cross Sections @ 700°K Table Length = 45			
$\bar{\sigma}_c(b)$	5.25 ± 0.20	5.24 ± 0.10	-0.01 ± 0.22
$\bar{\sigma}_s(b)$	10.39 ± 0.20	10.22 ± 0.07	-0.17 ± 0.21
Effective Cross Sections @ 700°K Table Length = 10			
$\bar{\sigma}_c(b)$	5.25 ± 0.20	5.38 ± 0.10	$+0.13 \pm 0.22$
$\bar{\sigma}_s(b)$	10.39 ± 0.20	10.33 ± 0.06	-0.06 ± 0.21
Percentage Change in Effective Cross Sections Between 300 and 700°K Table Length = 43 - 45			
Capture	0.225 ± 0.006	0.236 ± 0.004	-0.011 ± 0.007
Scattering	0.057 ± 0.004	0.057 ± 0.001	$+0.000 \pm 0.004$

Mechanisms of the calculation and the results of further tests are reported.

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Computational Methods and Techniques of Reactor Physics

A Library of Evaluated Integral Neutron Experiments, Viktor E. Hampel (UCLRL-Livermore)

A formalism has been developed for describing integral neutron experiments and corresponding calculational models. Provision has been made for specifying dimensions, material compositions, and relevant bulk parameters, using appropriate assumptions, uncertainties, definitions, and comments. A library of calculational models for 113 evaluated spherical critical assemblies and for 37 subcritical bulk pulsed spheres has been prepared¹ for testing of evaluated nuclear constants and calculational techniques, notably those employing Monte Carlo and S_n methods.¹⁻¹⁰

Integral experiments are selected from a bibliographic library⁸ on the basis of their material composition and accuracy, with the objective of yielding readily interpretable calculated results. Simple, spherical, critical experiments reflected in varying thicknesses by single elements or compounds were evaluated first. These can be described exactly in one-dimensional neutron transport codes. Spatial symmetry of the energy-dependent neutron flux simplifies interpretation. By selecting different fissile species, reflected by the same material, one may discern causes of trends in the comparison of calculations to experiments as the neutron energy spectrum is gradually altered for increasing thicknesses of the reflecting material. Light and heavy reflector materials offer an opportunity to study nuclear reaction rates, although in an integrated manner, under extreme conditions of neutron moderation. Simple spherical subcritical pulsed sphere experiments, containing one dominant material, provide a unique testing ground for the examination of the space- and time-dependent transport of neutrons.

Experimental uncertainties, approximations to reality used in the descriptions, and poor application of calculational models may lead to misinterpretation of calculated results. To avoid inadvertent error, we combine, whenever possible, several experiments from different sources into a coherent series so that singular and systematic errors might be detected more readily from the pattern of calculated data. This has proved to be a practical approach.

Inexplicit or incomplete descriptions of selected integral assemblies are clarified with the assistance of experimenters. When this is not possible, sensitivity studies are used to judge the validity of assumptions necessary during evaluation. Analyses of this type aid the assignment of an overall uncertainty to the calculational model.

Neutronic models of evaluated integral experiments, the neutronic computer program, and different libraries of neutron cross-section libraries are stored on magnetic tape and in the Photo-Store. (Made by IBM, with a total storage capacity of 10^{12} bits. Formally named "Photo-Digital Storage System, Type 1360," this system stores digital information on pieces of film. The device is attached on-line through the PDP-6 computers to two CDC-6600 and two CDC-7600 machines.) These files are created independently and may be combined at will. The computer program used to store and to organize textual and numeric data, and for programmatic operations, is

MASTER CONTROL.¹¹

TART¹² Monte Carlo neutron transport calculations are customarily processed with 160,000 neutron histories yielding a probable error of better than 0.2% in the expectation value of the neutron multiplication factor. ANISN¹³ time-independent S_n calculations are carried out in parallel as needed in the P-3, S-16, 66-group format with a convergence criterion of 0.1%. Calculated results of TART and ANISN are stored by computer to make their correlation and interpretation easier.

We have the capability to translate ENDF/B and AWRE evaluated neutron cross-section libraries into the Livermore format, and, hence, are in a position to use CLYDE⁸ as the common processing code for the generation of group constants from different sources. This allows comparison of evaluations from different libraries in the same format.¹⁴

By choice we do not provide an automatic feedback for the scaling and alteration of neutron cross sections. Our basic axiom is that any computerized analysis of evaluated nuclear constants must not offend the physics of nuclear processes, must take place within the constraints of experimentally measured or inferred uncertainties, be addressable selectively to any reaction and energy domain of an isotope, and be subservient to human judgment.

The testing of cross sections is expected to become more susceptible to interpretation when the linkages between the library of integral neutron experiments and codes are further simplified. Leakage spectra, spectral indices, and reactivity worths are added to the library as required for those experiments where they are available.

Our operational experience with the evaluated library of integral neutron experiments will be demonstrated.

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Criticality Safety—Data and Analysis

Subcritical Experiments with Mixed-Oxide Fuels of Plutonium and Uranium in Poisoned Lattices, C. L. Brown, L. C. Davenport, E. D. Clayton (PNL)

Few data currently exist on the effectiveness of boron and cadmium for criticality prevention in operations external to reactors that may involve fuel elements under conditions of water immersion. Material bucklings and extrapolation distances have previously been measured and reported for 25.2 wt% $\text{PuO}_2\text{-U(Nat)O}_2$ fuel pins in water.¹ These experimental results have also been compared with one-dimensional diffusion theory calculations using ENDF/B version II cross-section data.² The previous measurements have now been extended to include criticality data on these same fuel pins positioned in lattices with boron- and cadmium-poisoned water.

The fuel pins used in this experimental series are of a prototype design of driver fuel pins that are to be used in the Fast Flux Test Facility. This work is part of a criticality measurements and calculational program to provide safe limits for the fabrication, storage, shipment, and reprocessing of FFTF fuels. The results are presented in Table I.

The composition and dimensions of the fuel pins used in the experiments are as follows:

The experiments utilized a 48-in.-diam $\times$ 72-in.-high water tank for the lattice framework. Two different lattice pitches were studied: the 0.600-in. spacing corresponding to the maximum material buckling situation in the unpoisoned lattice, and the 0.900-in. spacing which was very close to the minimum in the critical number of pins, which is 149.5 ± 0.6 occurring at 0.88 ± 0.02 -in. lattice pitch. For five of the measurements at zero and low poison concentrations, neutron multiplication measurements were carried out to within 96% of the critical number of pins. At this point, the neutron source was removed and a "backoff" experiment was performed using neutrons from the spontaneous fission of ^{235}Pu and ^{240}Pu as the neutron source. This technique produced a linear extrapolation to within ± 1 pin. Material buckling and extrapolation distance were then determined by the flux shape method in which the flux distribution is measured perpendicular to the source plane.

Additionally, results were determined for six of the lattices using the exponential pile technique, with the variable loading method, in which flux distributions perpendicular to the source plane were measured for lattices of two different radii. The results are presented in the last three columns of Table I and the systematic differences in the methods can be compared. The critical number of fuel pins deduced indirectly from the poisoned exponential experiments is observed to be slightly higher (7-10%) than those determined directly from the more accurate critical approaches. However, agreement for the unpoisoned lattice on which measurements were made is excellent. In the case of the higher water-to-fuel ratio, as little as 0.4 g natural boron/liter or 0.9 g cadmium/liter is observed to more than double the critical mass. The effects of the boron and cadmium additions as predicted from theory will also be discussed.

Pin Dimensions

Fuel Column	o.d. (in.)		Length (in.)
Cladding (316 SS)	0.195		27.25
0.015 in. wall	0.230		28.70
Fuel Enrichment		Fuel per pin	
25.2 wt% $\text{PuO}_2\text{-U(Nat)O}_2$		$\text{PuO}_2\text{-UO}_2$: 138.4 ± 1.3 g	
Plutonium Composition		Pu: 30.75 ± 0.03 g	
^{238}Pu : 0.04 ± 0.01 at. %		U: 91.16 ± 1.03 g	
^{239}Pu : 86.19 ± 0.06 at. %		O: 16.49 ± 0.17 g	
^{240}Pu : 11.88 ± 0.06 at. %		Fuel Density	
^{241}Pu : 1.73 ± 0.01 at. %		10.35 ± 0.09 g/cm ³	
^{242}Pu : 0.16 ± 0.01 at. %		$(93.34 \pm 0.79\%$ theoretical)	

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A Highly Enriched Uranium Metal Sphere Assembly, J. T. Mihalcz (UCC-OR)

The spherical critical mass of $\sim 93\%$ ^{235}U enriched uranium metal which has been used as a benchmark problem for verification of calculational methods¹ is based on the delayed critical mass measurements made on Godiva corrected to standard conditions.² A near spherical mass of U(93.2), recently surplused from an experimental program examining fast reactor behavior with an accelerator booster,³ was re-machined to a spherical radius of 3.466 in. The device is to be used in a study of neutron kinetic parameters.⁴ The data obtained for the delayed critical configuration confirm the re-evaluated critical parameters recently reported by Hansen and Paxton.⁵

The assembly was constructed in three sections similar to Godiva I.⁶ A photograph of the sphere in the disassembled condition is shown in Fig. 1. The results of the measurements are given in Table I. The reactivity effect of removing material from the surface of the sphere was measured by reducing the radius of curvature of the lower section and repeating the assembly. This result, 0.082 c/g of surface material, and the data of the table can be used to adjust the mass to delayed criticality. To compare this delayed critical mass to those from the Godiva and shell experiments,³ the critical specification of a bare system can be converted to apply to any other density by means of the exact relation that the critical mass is inversely proportional to the square of the density, if shape is preserved.⁶ The adjustment for a moderate change ^{235}U enrichment may be accomplished by the empirical expression that the total critical

TABLE I

Results of Critical Approach and Exponential Experiments with 25.2 wt% $\text{PuO}_2\text{-U(NAT)O}_2$
Pins in Boron and Cadmium Poisoned Water
(Fuel Height: 27.25 in.)

Lattice Pitch (in.)	Water-to-Fuel Concentration (Volume Ratio)	Poison Concentration (g/l)	Critical Approach Experiments		Material Buckling (M^{-2})	Extrapolation Distance (cm)	Exponential Experiments		Using Two Exponentials In Variable Loading Method		
			Number of Pins For Criticality	Critical Mass (Kg Pu)			Number of Pins in Assembly	Relaxation Length (cm)	Number of Pins For Criticality	Material Buckling (M^{-2})	Extrapolation Distance (cm)
0.600	8.60	None	219.2 ± 0.6	6.74	172.7 ± 2.2	7.25 ± 0.20	120	13.71	---	---	---
0.600	8.60	0.542 Natural ± 0.005 Boron	333.4 ± 0.5	10.25	141.8 ± 0.3	6.70 ± 0.20	138 168	11.19 13.21	358.4	129.8	7.21
0.600	8.60	1.098 Natural ± 0.011 Boron	---	---	---	---	198 282	12.18 18.11	483.8	112.8	6.65
0.900	21.01	None	149.8 ± 0.6	4.61	138.5 ± 1.0	6.89 ± 0.20	66 78	12.10 14.27	150.2	136.1	7.07
0.900	21.01	0.410 Natural ± 0.004 Boron	322.8 ± 0.2	9.93	91.7 ± 0.1	5.90 ± 0.20	126 162	12.87 16.35	344.5	84.3	6.52
0.900	21.01	1.103 Natural ± 0.011 Boron	---	---	---	---	246 294	11.78 12.63	∞	7.0	8.24
0.900	21.01	0.879 Cadmium ± 0.008	328.4 ± 0.2	10.10	90.1 ± 0.1	5.99 ± 0.20	120 168	12.52 17.01	359.0	80.2	6.90

(a) Obtained from least-squares fit of "back-off" inverse multiplication measurements.

TABLE I
Description and Reactivity of a Uranium
Metal Sphere Assembly

Uranium density, ^a g/cm ³	18.73 ₂
Wt% ²³⁵ U ^b	93.20 ₆
Radius, cm ^c	8.804 ₆
Assembly reactivity, cents	+70.0
Reactivity corrections, cents	
Support holes ^d	
top	1.3
center	1.4
bottom	2.9
Void within sphere ^e	3.2
Surface mass deficiency ^f	7.6
Lower support stand ^g	-6.3
Reflection from other support structure and the room ^g	-15.0
Corrected reactivity, cents	+65.1
Corrected multiplication factor ^h	1.0043 ± 0.0007

^aFrom hydrostatic density measurements.

^bOther uranium isotopes were present in the following amounts: 0.97% ²³⁴U, 0.24% ²³⁶U, and 5.584% ²³⁸U. The total impurity content was 350 part (mass) per million parts of uranium.

^cThe radius varied from 8.803₁ to 8.807₂ cm.

^dReactivity effect of replacing the support materials within the sphere radius with uranium.

^eThese include the voids around the alignment plug and the plug for the central hole, and the gap between major sections.

^fFlat spots and tapped holes on the surface for installation of uranium mass adjustment buttons.

^gReactivity effect of support materials outside the sphere radius.

^hA value of $\beta_{\text{eff}} = 0.0066$ was used to convert reactivity to multiplication factor.

mass is inversely proportional to the 1.72 power of the enrichment. Thus, from these measurements the critical mass of a 93.8 wt% ²³⁵U enriched uranium metal sphere with uranium density of 18.75 g/cm³ is 52.11 ± 0.08 kg of uranium. This result agrees with that previously given by Hansen and Paxton, 52.28 ± 0.15 kg.

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Fig. 1. Photograph of the sphere assembly in the dis-assembled condition.

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Unreflected ²³⁵U C₁₈₂ Critical Assembly, J. Carlton Hoogterp, Gordon E. Hansen (LASL)

The Los Alamos critical assembly known as Honeycomb is intended to supplement metal assemblies as a source of effective neutron cross sections in the ~0.1-MeV region. It is a 48-in. cube of interleaved U(93.15) foil and graphite plates in a tubular aluminum matrix (78 × 78 × 72 in.), for which the C/²³⁵U atomic ratio is 182. An arrangement for remotely introducing central samples facilitates precision measurement of effective absorption cross sections (central reactivity coefficients).

Several experiments were for the purpose of simplifying the definition of the system. In one case, reactivity effects of a mock concrete wall led to the estimate that total neutron return from the room adds 1.0 ± 0.5 \$. Then, change from gravity-compacted horizontal foil-plate lamination (which left a 0.3-in. void layer at the top of each matrix tube) to loosely arranged vertical lamination increased the reactivity 0.72 \$. From this it was estimated that further removal of void layers (no streaming) would lead to an additional gain of 0.25 ± 0.15 \$. Finally, reactivity changes with changes of U-foil thickness at constant composition indicated a reactivity loss of 0.1 \$ in going from the 0.003-in. foil used to zero thickness.

These corrections and the experimental excess reactivity (1.0 \$) lead to the following idealized system. The multiplication factor $k_{\text{eff}} = 1.001 \pm 0.004$ applies to a homogeneous 48-in. cubic core, unreflected except for Al

extending to the dimensions noted above, which has the following material specifications:

$$\rho_{235} = 0.1456 \text{ g/cm}^3$$

$$\rho_{238} = 0.0107 \text{ g/cm}^3$$

$$\rho_c = 1.353 \text{ g/cm}^3$$

$$\rho_{Al} = 0.165 \text{ g/cm}^3.$$

For comparison, DTK computations with Hansen-Roach and ENDF/B cross sections (both 16-group) gave $k_{eff} = 1.003$ and 0.998 .

Characteristics of this assembly are $\sigma_f(^{235}\text{U})/\sigma_f(^{238}\text{U}) = 110$, and Rossi $\alpha = 948 \text{ sec}^{-1}$ at delayed criticality. Reactivity coefficients available at this time are tabulated in Table I.

TABLE I
Honeycomb Reactivity Coefficients

Material	Central Coefficient ($\beta/\text{g-atom}$)	Average Coefficient ($\beta/\text{g-atom}$)	Remarks
U(93.15)	7.07 ± 0.14	1.31 ± 0.02	2-mil foil
D-38	-8.7 ± 0.2	-1.4 ± 0.02	2-mil foil
C	0.132 ± 0.006	0.044 ± 0.002^a	
Mo	-3.35 ± 0.07	-0.47 ± 0.01	5-mil foil
Nb	-2.42 ± 0.05	-0.340 ± 0.01	5-mil foil
Ta	-33.8 ± 0.7	-5.13 ± 0.1	5-mil foil
W	-13.6 ± 0.3	-2.00 ± 0.04	5-mil foil

^aExtrapolation to zero thickness.

The Nuclear Criticality of Intersecting Cylinders of $\text{U}(5)\text{O}_2\text{F}_2$ Aqueous Solution, E. B. Johnson (UCC-OR)

A program of experiments investigating the critical conditions of intersecting cylinders of fissile material has been inaugurated to further design of processes for fissile material of low ^{235}U content and to further the development of analytical methods applicable to nuclear criticality.

Earlier experiments of this type at the Oak Ridge Critical Experiments Facility¹ and at the Rocky Flats Plant² of Dow Chemical Company were concerned with aqueous solutions in which the uranium was enriched to about 93% in ^{235}U .

Present work reports the results of measurements and calculations of uranyl fluoride aqueous solutions of various chemical concentrations in which the uranium was enriched to 5.0% in ^{235}U . Vessels of two different shapes were investigated. One was constructed of 0.063-in.-thick aluminum and consisted of two 11.0-in.-diam cylinders intersecting at 30 deg, thereby forming a distorted "Y." The other vessel was constructed of Plexiglas, a methacrylate plastic ($\text{H}_5\text{C}_2\text{O}_2$) having a density of 1.2 g/cm^3 . This vessel consisted of two cylinders intersecting to form a cross. The horizontal arms were tipped 1 deg to facilitate draining. Measurements were made with four cylinder diameters.

Various critical configurations of the solution have been investigated by the KENO Monte Carlo code³ by Whitesides⁴ using Hansen-Roach cross sections.⁵ The results are shown in Table I (p. 679).

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Criticality Considerations in the Non-destructive Assay of Nuclear Material, D. B. Smith, R. A. Forster, M. M. Thorpe (LASL)

The nondestructive assay (NDA) of fissile material by either active or passive neutron techniques often involves placing the container into a highly reflecting counting geometry. Of particular concern is the assay of nuclear scrap and waste which may be stored in containers ranging from 1-liter bottles to 55-gal drums. Past experience has indicated that one cannot assume anything about either the amount or enrichment of the fissile material in a container of scrap or the nature of the matrix material present. Clearly, careful consideration must be given to the possibility of criticality for the safe nondestructive assay of nuclear scrap material.

Most references reporting criticality data or calculations give only the configuration necessary to attain criticality and do not indicate the manner in which a fixed volume of material approaches criticality. In addition, applying criticality criteria from documents such as The Nuclear Safety Guide¹ to some NDA systems would be sufficiently conservative as to severely limit the number of scrap containers that could be measured. Therefore, calculations are undertaken for each NDA system to investigate the possibility that a critical system could be formed when a container of scrap of a known volume is placed into the assembly. All criticality calculations are performed using the DTF-IV neutronics code² and the Hansen-Roach 16-group cross sections.³ These cross sections have been used extensively by Stratton⁴ in criticality calculations which agreed well with available experimental data.

In all cases, it is desired that the calculation model the actual physical configuration as closely as possible. Any approximations that are used, such as homogeneity of the fissile material in a highly homogeneous matrix (CH_2), and in some cases, spherical symmetry, are chosen to make the calculational model more reactive than the actual NDA device. These calculations provide the criteria by which standard operating procedures (SOP) are developed. In many cases, the gross weight of the scrap container is a sufficient screening procedure. This weight limit is based on a combination of fissile material and moderator mixed in optimum proportions to yield a calculated $k_{eff} = 0.9$. If the gross weight limit is too restrictive, other screening procedures are developed and added to the SOP.

As an example, consider the active and passive neutron assay of plutonium scrap in one-gallon cans. The

TABLE I

Critical Conditions of Intersecting Cylinders of $U(5.0)O_2F_2$ Aqueous Solution.
The vessels were reflected by an effectively infinite thickness of water except on top.

Inside Diameter of Vessel (in.)	Solution Concentration Density		Liquid Height Above Bottom of Vertical Cylinder (cm)		KENO Calculation of k_{eff}
	(g of U/liter)	(g/cm ³)	Water Reflector	Solution	
30 deg "Y" ^a ; Temperature of Water and Solution ~26°C					
11.0	907.0	2.0289	209.7	128.2	0.9765 ± 0.0059 ^b
			144.5	128.6	
			128.8	129.1	
11.0	884.9	2.0055	209.8	131.4	0.9959 ± 0.0061
			146.8	131.6	
			131.6	132.2	
11.0	877.2	1.9953	209.8	132.1	0.9974 ± 0.0057
			147.1	132.4	
			132.4	132.2	
11.0	825.3	1.9381	209.5	141.7	1.0003 ± 0.0060
			157.1	141.9	
			142.8	142.8	
11.0	801.1	1.9108	209.7	155.3	1.0103 ± 0.0056
			171.5	155.4	
			155.5	155.9	
11.0	796.1	1.9043	209.7	169.1	1.0043 ± 0.0045
			185.1	168.0	
			167.7	168.0	
Plexiglas Cross ^c ; Water Temperature ~23°C; Solution Temperature ~25°C					
10.50	904.85	2.0201	210.5	115.6	0.9924 ± 0.0065
			115.9	115.9	
			117.3	117.2	
			210.5	133.9	
10.75	906.4	2.0256	134.1	134.1	0.9864 ± 0.0061
			210.5	109.6	
			110.6	110.6	
			210.5	115.6	
10.75	857.4	1.9712	116.1	116.1	1.0005 ± 0.0058

^aThe axis of the principal member was vertical; the axis of an equal-diameter member, inclined 30 deg from the vertical, intersected the principal axis 29.9 in. from the bottom of the principal member. Irregularities in the fabrication of the intersection introduced some uncertainties in the calculations.

^bThe uncertainties associated with values of k_{eff} are one standard deviation and correspond to a 67% confidence interval.

^cThe vessel consisted of intersecting cylinders whose axes lie in the same vertical plane. The axes of the side members were inclined 1 deg with the horizontal to facilitate draining. These "horizontal" axes intersected the vertical axis 36 in. from the bottom of the principal member.

active NDA system for such containers consists of a large block of carbon having a steel-Cd-B lined cylindrical cavity. The 4 π passive neutron counter consists of a ring of 18, 6 atm, 1-in.-diam, 24-in.-long ³He proportional counters imbedded in a cylindrical polyethylene moderator lined with cadmium.

Calculations for both counters were performed for a range of ²³⁹Pu masses (100 g to 6 kg) and various enrichments of ²⁴⁰Pu (0 to 6%). An examination of the resultant k_{eff} vs ²³⁹Pu mass curve (with 6% ²⁴⁰Pu) for the active counter showed that a 1-gal can weighing <15 lb and

having 6% ²⁴⁰Pu could be safely placed in the assembly. Plutonium-240 is an important poison as illustrated by the fact that the same 15-lb can with zero percent ²⁴⁰Pu would be critical in the active NDA assembly. Assay in the passive neutron assembly is limited to 500 g PuO₂ (6% ²⁴⁰Pu) for reasons of assay accuracy. Therefore, the counting rate of 500 g PuO₂ outside the counter might be used as the screening criterion. The results of the calculations, which included configurations that could produce a low count rate when unreflected but could yield a critical NDA system, showed that any container having a count rate outside the counter less than that of 500 g

PuO_2 (6% ^{240}Pu) will have a $k_{\text{eff}} < 0.9$ when placed in the counter.

Similar calculations were performed for 97% enriched uranium scrap in 4.75- and 6.00-in.-diam bottles placed in the active neutron counter. The smaller diameter container was found to be considerably less reactive and hence had a much less restrictive weight limit than the larger bottle.

By considering each NDA system separately and using the calculation technique described above, the NDA can proceed in a completely safe manner regardless of what might be labeled as scrap.

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A Difficulty in Computing the k-effective of the World, G. E. Whitesides (ORNL).

In the course of applying Monte Carlo programs in the solution of criticality safety problems, a difficulty in correctly computing certain types of systems has arisen. In view of the increasing use of Monte Carlo-type programs, it is important that some statement be made about the type of systems in which the difficulty is likely to be encountered, since when one attempts to compute the multiplication factor for such a system, the result will almost always yield a low, and hence for criticality safety purposes, nonconservative result. This can occur with no hint that the computed result is in error.

The extreme example, which defines a situation in which this difficulty can exist, is the "k-effective of the world" problem. That is, if one attempts to calculate the k_{eff} of the world using a Monte Carlo calculation, what k_{eff} would be computed assuming that there are several critical assemblies located around the world? The answer would likely be the k_{eff} of the world with no critical assemblies present. The cause of the erroneous result is the fact that the volume of fissile material in the world would be so large relative to the volume of fissile material in the critical assemblies that most commonly

used forms of sampling would almost never "see" the critical assemblies. Hence, this would not reflect their existence in the computed k_{eff} .

A more commonly encountered example in which a user of Monte Carlo programs might observe this difficulty is illustrated by a $9 \times 9 \times 9$ array of plutonium metal spheres with a radius of ~ 4 cm, spaced on 60-cm centers. The array is reflected on all sides by a thick water reflector. The k_{eff} of this system is computed to be 0.93. If the sphere in the center unit of the array is replaced by a sphere of plutonium that is exactly critical as a bare unit and the calculation repeated in the standard fashion using the Monte Carlo method, the calculation will yield a k_{eff} of 0.93, reflecting the same difficulty encountered in the "world k_{eff} " problem.

The erroneous results for these types of problems are the result of the failure of the calculation to converge the source to the fundamental source mode. The difficulty can range from a problem which converges so slowly that the normal number of generations examined are insufficient to assure convergence to problems that perhaps will never converge.

The answer to the problem lies in being sure that source convergence is achieved. Most Monte Carlo programs start all neutrons initially with either a flat or a cosine distribution over the fissile material in the system. For the majority of problems this is adequate and maybe even exact. Unfortunately, the choice of a standard start procedure that is adequate for a system with a region of localized multiplication that is somewhat higher than the multiplication of the major portion of the volume of the system is difficult, if not impossible, to define. One difficulty, which should not be underestimated, is trying to determine if, and where, this situation exists. Because of this, it is not the intention of this paper to offer a solution to the general problem; but only to bring to the attention of the users of Monte Carlo programs the fact that the computed multiplication may be in error for such systems.

If the region of highest multiplication can be established, it appears to be adequate to ensure eventual source convergence to the fundamental mode by starting all neutrons in this region. If there are more than one such regions, it may be necessary to run several cases varying the initial source in order to determine the initial conditions that will lead to a fundamental mode source distribution.

The Monte Carlo method has opened up the path to very precise evaluations of the criticality safety of almost any situation likely to be encountered. Its use, however, should be tempered by the realization that unless the correct fission distribution is achieved, the results will most likely be nonconservative.

Criticality Safety in the Light-Water Reactor Fuel Cycle

Criticality Safety Considerations in the Fabrication of UO_2 and $\text{PuO}_2\text{-UO}_2$ Fuels for Light-Water Reactors, C. L. Brown (PNL), *Invited*

Commercial fabrication of uranium oxide fuels for light-water reactors is the fastest maturing segment of the nuclear fuel cycle. Some ten commercial fuel fabricators now routinely manufacture uranium fuels on a more or less mass production basis. With this maturing comes an increased incentive to increase production rates and thereby reduce fuel fabrication costs. One astutely observes that the only restraints that prevent unlimited fuel throughput are the limits imposed to ensure criticality safety. It, therefore, behooves us to periodically reexamine plant equipment in light of advances in criticality safety technology and to adjust limits wherever possible to enhance the economics of the fuel cycle.

This paper will discuss some of the criticality safety aspects of uranium oxide fuel fabrication, and present new information on plutonium oxide/uranium oxide fuel fabrication, which is now in the developmental production-line stage.

Uranium oxide fuel fabrication plants all use the same basic techniques and processes. Uranium is received as UF_6 ; the UF_6 is converted to UO_2 , most commonly by an aqueous process; the UO_2 is pressed into pellets; the pellets are loaded into rods; and the rods are combined into fuel assemblies.

The most challenging part of the process from the criticality safety viewpoint is the conversion of UF_6 to UO_2 . Most widely used in commercial facilities is the ammonium diuranate (ADU) process.^{1,2} The chemistry of this process involves two steps:

1. hydrolysis of UF_6 with ammonium hydroxide: $\text{UF}_6 + 4\text{NH}_4\text{OH} \rightarrow \text{UO}_2\text{F}_2 + 4\text{NH}_4\text{F} + 2\text{H}_2\text{O}$.
2. precipitation of the hydroxide and formation of ADU: $\text{UO}_2\text{F}_2 + 2\text{NH}_4\text{OH} \rightarrow \text{UO}_2(\text{OH})_2 + 2\text{NH}_4\text{F}$
 $2\text{UO}_2(\text{OH})_2 + 2\text{NH}_4\text{OH} \rightarrow \text{U}_2\text{O}_7(\text{NH}_4)_2 + 3\text{H}_2\text{O}$.

In the subsequent heating step, the ADU undergoes thermal decomposition with simultaneous reduction to UO_2 .

Another production method used for the conversion of the UF_6 to UO_2 is the ammonium uranyl carbonate (AUC) process. The form of the AUC precipitate is probably $(\text{NH}_4)_4[\text{UO}_2(\text{CO}_3)_3]_2$. This compound is calcined to UO_3 or U_2O_5 , which is then reduced³ to UO_2 .

There has been mention, but no detail, of a new shorter and more economical dry process for the conversion of UF_6 to UO_2 , which has been successfully developed in the UKAEA. A 300 ton/year production line is currently under production based on this process.¹ The fact that this method is dry implies a tremendous production rate advantage over the aqueous processes, because a major part of the criticality safe bottleneck is eliminated.

Uranium fuel fabrication covers the full spectrum of criticality safety technology. UF_6 is handled in ton quantities, safe by virtue of undermoderation; UF_6 to ADU or AUC conversion is done in small quantities, safe by optimum moderation; fabrication of pellets, rods, and assemblies takes into account heterogeneity, and again, undermoderation; and neutron interaction in all areas af-

fords a challenge to the analyst. Of these problems, the one that will be stressed in this paper is the importance of density in the ADU process.

Plutonium oxide/uranium oxide fuel fabrication for light-water reactors will require the development of new fabrication equipment and techniques. The fabrication steps involved will be: blending of uranium and plutonium, which could be a dry UO_2 - PuO_2 process or an aqueous mixing or precipitation process; pellet pressing; rod loading; and fuel element assembly. The challenge to the development of equipment will be presented primarily by radiation safety, rather than criticality safety. Radiation safety will require that all operations from mixed-oxide blending to fuel rod loading be carried out in sealed glove boxes. Calculated criticality parameters for the slightly enriched homogeneous $\text{PuO}_2\text{-UO}_2$ system will be presented. Parameters for the similar mixed-oxide heterogeneous rod system have been reported.⁴

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Engineering for Criticality Control in the Recovery of Spent Fuel, J. P. Nichols (ORNL), *Invited*

Near-future, commercial reprocessing plants using aqueous processes will recover plutonium from spent, light-water reactor fuels at rates of several tens to several hundreds of kilograms per day. In the past several years, significant progress has been made in the technology and economics of criticality control for such large-scale operations through the demonstration of the use of soluble and fixed neutron absorbers, low-inventory process vessels, and methods for shipment of large quantities of fissile products. Important problem areas that require continuing investigation include the development of evaluated criticality data for mixed uranium-plutonium systems, methods for management of fissile solids in aqueous systems, and methods for continuous assessment of process inventories.

Soluble Poisons. The use of soluble neutron absorbers (boron, cadmium, and gadolinium) for criticality control in solutions of ^{235}U and ^{233}U has been demonstrated in plant-scale operations by the Idaho Nuclear Corporation, Nuclear Fuel Services, Inc., and the Oak Ridge National Laboratory. This work forms a precedent for use of the method in systems having large volumes and throughputs, and demonstrates procedures for controlling solution stability and concentrations.

Fig. 1. KENO calculated k_{eff} for a water-reflected ($16 \times 16 \times 2$) array of UNC Type-2800 shipping containers, each containing a fully moderated Yankee Rowe fuel element, as a function of water content of the shipping container.

Fixed Poisons. The use of large tanks packed with borosilicate-glass Raschig rings has been demonstrated as an economical method for storing aqueous nitrate solutions of fissile products. This method of storage facilitates the blending of large batches of product to achieve isotopic uniformity. Also, numerous alloys and dispersions of neutron absorbers in metal have been developed for criticality control in environments that involve mechanical shock or high radiation levels.

Low-Inventory Process Equipment. Solvent extraction equipment and dissolvers that have very high throughput but low inventory have been developed. The low-inventory feature enhances neutron leakage and facilitates protection against the gradual accumulation of large masses of fissile material.

Shipment of Solid Products. Recent trends toward shipment of the uranium and plutonium product from reprocessing plants as UF_6 or PuO_2 , as opposed to a nitrate solution, have the promise of significantly increasing the quantity of fissile material per shipment with no compromise in criticality or radiological safety.

Critical Experiments and Analysis. The nuclear safety parameters of hydrogenous systems containing mixtures of uranium and plutonium will be based largely on calculations that are keyed to relatively few critical experiments. We must continue to emphasize the development of reliable computational techniques and evaluated differential cross-section data, and to select critical experiments that will provide for maximal benefits. Credence in computed data must be established by thorough reviews of possible errors and independent assessments by responsible authorities in nuclear criticality safety.

Undissolved Plutonium Solids. Much of the criticality safety of a reprocessing plant depends on complete dissolution of the fuel to produce a homogeneous aqueous solution that can be assayed accurately for fissile content. Mounting evidence indicates that solid residues enriched in plutonium may remain if conventional dissolution prac-

tices are employed for fuels that are made by the preferred technique of mechanical blending individual particles of UO_2 and PuO_2 . Continued work is needed to develop methods that will either ensure complete dissolution or provide for handling and assay of solids containing plutonium.

Process Inventory Control. Spurred by the requirements for safeguarding materials, various techniques (e.g., in-line monitors, tracer techniques, and methods of nondestructive assay) are being developed to promote accurate measurement of the quantity of fissile materials in a process. These developments will provide a basis for future in-line computer systems that will provide a continuous estimate of the fissile inventory in process vessels and forecast the results of process transients and upsets.

Criticality Safety Analysis of an LWR Fuel Assembly Shipping Container, E. Fass (Gulf-UNFC), R. Kropp (UNC-New Haven)

The nuclear evaluation of reload fuel assemblies utilizes methods that lend themselves to evaluating the criticality safety of the shipping containers used in their transport. This discussion concerns itself with the recent analysis of a package designed to carry BWR or PWR fuel assemblies. The analysis begins with a transport and diffusion theory approach and is overchecked using a Monte Carlo method.

The Model UNC-2800 shipping container consists of a fuel assembly support shock mounted to an outer container. The outer container is a 36-in.-diam, 12-gauge steel cylinder, 207 in. long.

Under normal transport conditions, there is no moderating material present and the UO_2 of <5 wt% ^{235}U cannot become critical.¹

For the accident condition, water flooding of the assembly and container vessel is assumed. Macroscopic

data for the fuel assembly are calculated by the LOCALUX² assembly reactivity code. LOCALUX is an improved version of the LASER³ code. Data for extra water regions are calculated from the FORM⁴ and TEMP-EST⁵ codes. Array infinite multiplication calculations were performed using the AIM-6 code⁶ with two energy groups and zero current boundary conditions. Multiplication factors were calculated varying the density of water between the fuel assembly and outer container. Reactivity levels for all finite arrays were calculated with the KENO⁷ Monte Carlo criticality code using the 16-group Hansen-Roach cross sections.⁸

For a PWR analysis applicable to the shipment of a single typical assembly, the maximum k_{∞} of 1.082 occurred with full moderation of the fuel assembly and no water elsewhere. As a check, KENO calculated a $k_{\infty} = 10.77 \pm 0.004$. It was then necessary to establish the number of shipping containers that could be transported safely. The array considered was a $16 \times 16 \times 2$ configuration surrounded by a water reflector. The first type of unit considered in the finite array was the one that gave the highest k_{∞} . Next, a series of KENO problems was run to determine if a more reactive finite array could be made by adding water to the region between the assembly and outer container wall. The results of these calculations are shown in the accompanying figure. The behavior of the k_{eff} curve is due to the absorptions in the steel walls of the outer container. As small amounts of water are added to the system, the k_{eff} decreases because of the additional moderation and absorptions in the added water. As the added water becomes significant, it acts as a reflector about the assemblies and decouples the containers. This behavior was also predicted by k_{∞} curves generated from AIM-6.

In a BWR analysis applicable to the shipment of 4 Dresden I fuel assemblies, the maximum AIM-6 calculated k_{∞} was 0.818. A single KENO calculation done for this most reactive array demonstrated that it met the criteria for a fissile class-I container.

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Criticality Safety Considerations in the Design of a Large Concrete Storage Array for Plutonium Oxide-Uranium Oxide Fuel Subassemblies, C. L. Brown, L. C. Davenport (PNL)

In handling nuclear fuels outside reactors, the most difficult criticality safety problems are usually those involving interacting arrays of nuclear materials. In most instances, the Monte Carlo technique is the only reliable calculational method available for solution. It is also frequently found that insufficient experimental data exist on the type of array under study to provide good checks on the methods used, although criticality safety guidance on a number of standard-type arrays is now becoming available.¹ Typical of these problems, and the method of solution, is the design of a large fuel subassembly storage pit for mixed-oxide fuels at Hanford.

Monte Carlo calculations have established that for 98 Fast Flux Test Facility (FFTF) driver fuel subassemblies (up to 100 kg ²³⁹Pu) in the concrete storage array, $k_{\text{eff}} < 0.85$ (see Table I). Conclusions are: (a) the loaded storage array is critically safe and acceptable for use; (b) under the design conditions specified, there is no credible way criticality can be achieved; and (c) the number of subassemblies moved into or out of the storage array at one time should be limited to one, although a double batch error resulting in two subassemblies side-by-side represents < 0.50 of the number of subassemblies required for criticality.

An FFTF subassembly is comprised of 217 pins in a close-packed (0.056-in. separation) hexagonal geometry. The pins are 0.230-in. o.d. $\times$ 8 ft long (0.015-in. wall,

TABLE I
Final k_{eff} for As-Built FFTF Fuel Subassembly Storage Pit*

	k_{eff}	
	GEM-III Code ^a	KENO Code ^b
Subassemblies Containing 25 wt% PuO ₂ -U(Nat)O ₂ Fuel; 11 wt% ²⁴⁰ Pu		
Dry subassemblies	0.458 $\pm$ 0.017	0.463 $\pm$ 0.018
Water-moderated subassemblies	0.672 $\pm$ 0.012	0.614 $\pm$ 0.006
Subassemblies Containing 31 wt% PuO ₂ -U(Nat)O ₂ Fuel; No ²⁴⁰ Pu		
Dry subassemblies	0.630 $\pm$ 0.024	---
Water-moderated subassemblies	0.800 $\pm$ 0.017	0.765 $\pm$ 0.008

*Fuel Pins: 0.230-in. o.d.; 0.200-in. i.d.; 316 SS cladding; 36-in. fuel length.

Subassemblies: 217 fuel pins; separated by 0.056-in.-o.d. wire wrap; 0.286-in. hexagonal lattice pitch $\pm 5\%$; 4.34-in. distance across flats.

Concrete Storage Array: 98 holes arranged in 7 rows of 14 holes each. Holes are lined with 6-in. Sch 40 PVC pipe. Spacing (c/c) between holes in rows is 10 in.; between rows is 24 in. Hole depth is 8 ft with top of hole (floor level) being 2 ft (minimum).

^aFuel Arrangement: Discrete fuel pins in hexagonal lattice.

^bFuel Arrangement: Fuel, clad, and water homogenized within subassembly boundary.

316 stainless-steel tubing) with a fuel section 3 ft long made up of 0.195-in.-o.d. $\times$ 0.25-in.-long mixed-oxide pellets. Distance across flats is 4.34 in.

The enrichment range for FFTF driver fuel is 18 to 31 wt% PuO_2 in natural or depleted UO_2 , with $\geq 11\%$ ^{240}Pu . In the analysis, two enrichments were assumed: 25 wt% $\text{PuO}_2\text{-U(Nat)O}_2$ with 11% ^{240}Pu , representative of first-cycle FFTF fuel; and 31 wt% $\text{PuO}_2\text{-U(Nat)O}_2$, the maximum enrichment expected. To avoid delays in the analysis due to lack of confirming experimental data, ^{240}Pu content in the 31 wt% PuO_2 enriched fuel was neglected—a conservative assumption.

Design of the storage area was strongly influenced by two factors: available space and required capacity. The desired location was an existing 250-ft² pit area in the FFTF Fuel Fabrication Demonstration Facility. Storage positions were required for about 100 subassemblies. From the criticality safety standpoint, to store up to 100 kg ^{239}Pu in such a confined area required a neutron isolator between subassemblies. Ordinary concrete with a minimum guaranteed density of 150 lb/ft³ was finally selected because of its radiation shielding properties, its relatively low cost, its ease of installation into the existing area, and its long-term stability. The array design selected was 98 holes arranged in 7 rows of 14 holes each; 10-in. center-to-center between holes in rows and 24-in. center-to-center between rows. For a row of 31 wt% PuO_2 enriched subassemblies side by side in air, but concrete-reflected, k_{eff} was found to be near unity (critical). With 3.375 in. of ordinary concrete and two thicknesses of polyvinylchloride plastic pipe (hole linear) between subassemblies in a row, k_{eff} was reduced to 0.63. A neutron poison was found to be unnecessary in the storage array.

Considerable effort was concentrated on evaluating the effect of concrete composition on k_{eff} for the storage array. Local concrete is high in basalt and low in limestone (carbon). The differences in aggregates used for ordinary concrete were found to change k_{eff} by as much as 250 milli-k between extremes.

Calculational methods used included three Monte Carlo computer codes. GEM-III² and GEM-IV³ were used with both an exact representation of the discrete fuel pins in a hexagonal lattice and with a homogenized subassembly. KENO⁴ was used with a homogenized subassembly representation, which utilized fewer regions and required less computational time. The k_{eff} values for the fuel storage pit are summarized in Table I; k_{eff} values for a

single water-reflected subassembly are given in Table II for comparison.

It is recognized that due to the lack of experimental data for direct confirmation, caution must be exercised in accepting a computational analysis. It was for this reason that both the British-developed GEM Code and the US-developed KENO Code were used for the analysis. Results are comparable.

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Criticality Safety Experience Under a Broad License, W. C. Borden, D. M. Dawson (GE-NC)

Effective operation of a high volume nuclear fuel fabrication facility requires a flexible criticality safety program that can respond rapidly to constantly changing process requirements. Optimization of production methods often requires incorporation of improved process techniques without the delays usually associated with the AEC licensing process. Such a criticality safety program has been established by General Electric under the "Broad License" concept developed several years ago by the AEC Division of Materials Licensing. Commission acceptance of such a program requires that five basic principles be established and satisfied:

1. functional independence of the criticality safety staff and operating management
2. a broad spectrum of validated methodology
3. demonstrated proficiency of the criticality safety staff and operating management
4. preestablished parameters within which the nuclear safety program must operate
5. a system of checks and balances to give a continuously high degree of assurance that the program will maintain the desired margin of safety.

Program conservatism should not be so severe that the business cannot remain competitive. Accordingly, the program should be responsive to the day-to-day pressures of production, concurrently maintaining a pre-established margin of safety.

To reconcile these seemingly paradoxical requisites, General Electric has established and operated such a criticality safety program, gaining AEC acceptance and responding successfully to the demands of the business. Assuming that calculational techniques are well standardized, this paper will be devoted to discussion of program structure, limitations, and experience over the past several years.

The program is founded on the competent area-manager concept. In other words, General Electric's safety philosophy assigns principal safety responsibility to the

TABLE II

Calculated k_{eff} for Single Subassembly*
(Water-Reflected Outside Subassembly Boundary)

Conditions	k_{eff}
Dry subassembly (GEM-IV)	0.590 $\pm$ 0.10
Water-moderated subassembly (GEM-III)	
• Hexagonal Lattice	0.732 $\pm$ 0.016
• Homogenized Lattice	0.714 $\pm$ 0.018

*Fuel Composition: 31 wt% $\text{PuO}_2\text{-U(Nat)O}_2$; no ^{240}Pu .

Fuel Pins: 0.230-in. o.d.; 0.200-in. i.d.; 316 SS cladding; 36-in. fuel length.

Subassemblies: 217 fuel pins; 0.286-in. hexagonal lattice pitch.

operating manager with support by staff safety functions. Thus, decisions on matters of safety are made by the area manager with the aid of advice from the safety staff. Therefore, criticality safety specialists are not influenced by production-oriented operational responsibilities.

This program concept necessarily relies heavily on certain preestablished license conditions such as:

1. a stated maximum ^{235}U enrichment
2. stated minimum margins of safety for individual geometries or accumulations of fissile material.
3. statements of policy concerning basic assumptions of analyses such as:
 - (a) degree of moderation
 - (b) number and/or type of contingencies involved in the analysis of the accident case.

Having thus established these basic conditions, which render the program acceptable to the Commission, a "Black Box" type of criticality safety service is integrated into an operating procedural system. Successful operation of this system requires that the following sequence of events be adhered to:

1. recognition by operating management that a proposed design or procedural change holds a potential for influencing inherent or environmental conditions affecting criticality safety
2. formulating the criticality safety evaluation
3. performing a physical audit of the change to assure that all conditions submitted with the request are factual
4. an independent review and approval of the technical analysis to assure proper and accurate application of methodology
5. issuance of a letter noting that the change has been reviewed by the criticality safety staff, and stating special precautions required
6. a continual non-technical audit to assure adherence to general procedural controls
7. a periodic technical audit to assure that all conditions remain as analyzed, and that no further changes have been instigated without benefit of a criticality safety review
8. a formal communication system to establish a continuous operating management awareness of the status or measure of the criticality safety performance in the area.

The program as it exists within General Electric is not faultless. Principally, two general categories of problems plague this type of program:

1. a tendency to implement minor changes, sans benefit of the formal criticality safety request and review system discussed above
2. a rather significant management tendency toward overuse of the nuclear safety service simply because it exists and/or because it requires little effort to initiate.

We are cognizant of these systematic imperfections and are continually improving the administrative aspects of the program. Nevertheless, the system has proved to be technically sound and functions smoothly enough to allow General Electric to meet the needs of its business without compromising criticality safety.

Shipment of 8-in.-i.d. Cylinders of UF_6 at Enrichments to 97.65% ^{235}U , and 12-in.-i.d. Cylinders at Enrichments to 30.00% ^{235}U , at Limited Moderation; and Subsequent Cylinder Cleaning, J. L. Feuerbacher (Goodyear Atomic Corp)

Eight-inch-i.d. $\times$ 48-in. UF_6 cylinders have previously been approved for storage and shipment of 12 $\frac{1}{2}$ % ^{235}U enrichment, and 12-in.-i.d. $\times$ 40-in. cylinders for 5% ^{235}U enrichment. On the basis of experimental data and computer calculations, for a UF_6 purity of 99.5%, 8-in.-i.d. nickel or Monel cylinders are safe at enrichments to

Fig. 1. Effects of number in array, and of spacing between units.

Fig. 2. Effect of thickness of concrete array reflector.

97.65% ^{235}U , and 12-in.-i.d. nickel or Monel cylinders are safe to 30% ^{235}U enrichment. The advantage of such usage, both in storage and shipment, is that 250 lb of UF_6 may be contained in an 8-in.-i.d. cylinder, and 450 lb in a 12-in.-i.d. cylinder, compared to 55 lb in presently approved 5-in.-i.d. $\times$ 30-in. cylinders, thus leading to substantial savings in cylinder inventory, handling, shipping, sampling, and analysis costs. These larger cylinders may be shipped in cylindrical (drum) packages of very nearly the same diameters as for 5-in.-i.d. cylinders. Present shipping criteria are much more stringent than are usually applied to in-plant storage (which can employ fairly small safety margins), requiring proof, instead, that five times as many units (as designated safe) must be subcritical in a cubical array, with optimum interstitial water and full water array reflection.

Proof of individual safety of these cylinders at the higher enrichments is based on comparison with critical experiments at or near the designated enrichments. Proof of array safety is purely computational and should be compared with experimental data when such become available.

Based on the present AEC and DOT criteria, computational proof is presented of the nuclear criticality safety and radiational safety of shipping sixty-two 8-in.-i.d. nickel cylinders or forty-one 8-in.-i.d. Monel cylinders at enrichments to 97.65% ^{235}U , or forty-one 12-in.-i.d. nickel or Monel cylinders at enrichments to 30.00% ^{235}U , all at a moderation limit of H/U atomic ratio ≤ 0.088 (conservatively equivalent to a UF_6 purity of 99.5%) in the previously tested and approved ORGDP packages.¹

The sensitivity of arrays of such packages to (a) small variations in spacing (Fig. 1), (b) increments in numbers (Fig. 1), and (c) concrete array reflection (Fig. 2), was also investigated. The results with the concrete array reflector are of considerable concern, since k_{eff} is markedly increased, as compared to water array reflection (Fig. 2). No distinction could be discerned among concretes containing silica aggregate, calcium carbonate aggregate, and Oak Ridge aggregate. The effects of concrete reflection and interstitial moderation have only been touched on computationally; these should be investigated experimentally and with more extensive computations, for high-enrichment dry UF_6 cases. (AEC criteria are expected to change in view of the findings of a number of investigators concerning concrete reflection and interstitial moderation.)

Safe methods of wet decontamination of these cylinders have also been investigated by means of hand calculations and available computer results, making use of heel size limits, boric acid for neutron absorption, and nonreflection, as controls. While some few additional computer calculations will be required to prove definite safety for the 12-in. cylinders, indications are that safety can be assured for wet decontamination of both sizes of cylinders.

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Fast Reactor Physics

A Comparative Study of Heterogeneity Treatments for ZPR Plate Cells, B. A. Zolotar, M. Salvatores* (ANL), B. M. Bingman, E. E. Lewis (Northwestern Univ)

Heterogeneity calculations for ZPR plate lattices play an important part in the analysis of fast criticals. Several treatments of this heterogeneity have been developed and this work compares the calculation of resonance cross sections by three different methods:

1. *The Standard Argonne Method*¹—The MC² code² is used to obtain homogeneous broad-group cross sections for the system under study. Corrections are then made in the resonance range, using equivalence theory³ and the two-sided E₃ formulation of Meneghetti,⁴ to obtain in-plate capture and fission resonance cross sections for each absorber plate. Broad-group flux shapes are determined from either S_n or integral transport^{5,6} methods and are used to obtain cell-averaged cross sections which preserve reaction rates. The new MC²-2 code⁷ will include the elements of this method.

2. *The RABBLE Method*⁸—This is an integral transport code with numerical integration of the slowing down equation over the resonance range.

3. *The SMC Method*⁹—A modified Monte Carlo treatment is used for the solution of the Boltzmann equation in the resonance region. The energy variation of a neutron history is treated exactly and the space-angle transport is approximated by a flat-source collision probability formulation.

The comparison was made for a realistic representation of the ZPR-6 Assembly 6¹⁰ unit cell as given in Table

TABLE I
²³⁸U Capture Cross Section
(barns)

	Standard Argonne Method	RABBLE	SMC
Group 17			
Infinitely Dilute	—	1.390	—
Homogeneous	1.160	1.129	—
Plate 2 ^a	1.053	1.023	0.998 ± 0.022
Plate 7 ^a	1.002	0.986	0.990 ± 0.040
Plate 10 ^a	1.050	1.024	1.100 ± 0.051
Cell Average	0.972	1.001	—
Group 21			
Infinitely Dilute	—	3.362	—
Homogeneous	1.197	1.109	—
Plate 2 ^a	1.126	1.077	0.97 ± 0.03
Plate 7 ^a	0.885	0.856	0.86 ± 0.03
Plate 10 ^a	1.064	1.043	0.98 ± 0.06
Cell Average	0.983	0.969	—

²³⁸ZPR-6 Assembly 6 Cell Representation

1 2 3 4 5 6 7 8 9 10
M UO Na FO SS U5 U8 FO Na UO

M = Matrix + Drawer

UO = $\frac{1}{8}$ -in. U₃O₈ Plate

NA = $\frac{1}{8}$ -in. Na Plate

FO = $\frac{1}{8}$ -in. Fe₂O₃ Plate

SS = $\frac{1}{8}$ -in. Stainless-Steel Plate

U5 = $\frac{1}{8}$ -in. ²³⁵U Plate

U8 = $\frac{1}{8}$ -in. ²³⁸U Plate

I. An attempt was made to ensure consistency of the cross sections used in the three calculations as far as was possible with the differing code formats. The resolved resonance cross sections were taken from ENDF/B Version I data. Other cross sections were similar but not identical to ENDF/B. The Standard Argonne Method used integral transport rather than S_n methods. Leakage corrections were not included in any of the calculations.

Table I shows some of the results for ²³⁸U capture. The two groups shown are Groups 17 (2.03 to 3.35 keV) which includes the Na resonance and Group 21 (0.275 to 0.454 keV) where the light element cross sections are relatively smooth. The Monte Carlo results are for 1500 histories per group. The agreement between the methods is relatively good with the possible exception of the cell-average value in Group 17. The equivalence theory used in the Standard Argonne Method appears to give reasonable in-plate cross sections even though the three plates containing ²³⁸U differ (as exhibited by the varying in-plate cross sections). The homogeneous and infinitely dilute values are given to provide a measure of the heterogeneity.

Additional results have been obtained for ²³⁵U resonance capture and fission cross sections and for flux shapes in the unit cell.

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A Comparative Analysis of Methods for Calculating Heterogeneity Effects in Fast Critical Assemblies—Part I: Methods, R. E. Mueller, E. M. Page (APDA), M. Komata (CRIEPI-Japan)

A comparative analysis of methods for determining heterogeneity effects in fast plate-type critical experiments has recently been completed.¹ This summary describes the methods and investigates the associated modeling assumptions. Analysis of the computed heterogeneity effects is given in the companion summary.²

The heterogeneity effect is defined as the difference obtained for any neutronic characteristic by an analytical model that accounts for the discrete plates of the assembly and that obtained by a model that assumes complete homogeneity. The homogeneous reference model utilizes multigroup diffusion theory in spherical geometry and incorporates the standard Breit-Wigner resonance self-shielding formulation for a homogeneous mixture of resonant and nonresonant material to obtain group cross sections for the heavy elements in the resonance regions.

The effects of heterogeneity on energy self-shielding of resonances is accounted for through separate self-shielding calculations for each heavy-element plate utilizing the equivalence principle and associated asymmetric Dancoff factors.³ The resulting plate-dependent cross sections for a given isotope are volume weighted over the drawer to form average-group cross sections for the spherical diffusion theory calculations.

The effects of spatial variations in the intra-drawer flux distributions that arise as a result of the heterogeneous plate structure was accounted for by several methods as given below:

1. *DTF-IV Weighted*⁴: Real flux distributions through the drawer were determined by solving the multigroup transport equations for a unit cell in slab geometry with the DTF-IV code using modified S₁₆ Gaussian quadratures. The resulting flux distributions were used to spatially weight the plate cross sections forming averaged values for the subsequent spherical diffusion theory calculation.

2. *Perturbation Approach*⁵: The heterogeneity effect is treated as a first-order perturbation using global diffusion theory fluxes produced by volume-weighted cross sections and perturbation cross sections defined as the difference between DTF-IV flux-weighted and volume-weighted cross sections.

3. *CALHET Weighted*: Identical to the DTF-IV Weighted method except that an escape probability-perturbation technique⁶ as computed by the CALHET code is used to obtain the intra-drawer fluxes.

4. *CALHET*: Direct computation of the flux distribution heterogeneity effect by the CALHET code.

5. *Bilinear Weighted*: As above for DTF-IV Weighted, but spatially weighting the cross sections with both the angular-dependent real and adjoint flux solutions in a bilinear weighting scheme formulated by Nicholson.⁶ A rationale was developed that defends the use of the weighted cross sections in global diffusion theory rather than transport theory calculations as originally developed.

6. *Scalar Bilinear Weighted*: As above for Bilinear Weighted but using instead the scalar real and adjoint flux solutions in an alternate bilinear weighting scheme.⁷ The two bilinear weighting schemes were computed by the MDF-4/BICAL code package, an extensive modification to the S_n code DTF-IV.

Common to all the foregoing methods is the problem of transforming the three-dimensional drawer geometry to a one-dimensional slab unit cell. The reference technique is to assume infinite length plates and to reduce the actual plate densities by the ratio of plate height to matrix tube height, homogenizing upper and lower drawer wall and matrix steel into the adjacent plates. Total cell leakage is accommodated by addition of group-dependent DB²'s to the removal cross section. An alternate calculation in ZPR-III Assembly 48 utilizing full plate densities resulted in an increase in the computed heterogeneity effect on criticality of 0.23% $\Delta k/k$, demonstrating the sensitivity to the geometric model.

Sensitivity to the values of DB² chosen for cell leakage representation was estimated by performing a cell calculation for ZPR-VI Assembly 6 with DTF-IV assuming zero leakage for all groups and comparing the flux distribution with a corresponding calculation employing nominal DB² values. The resulting ratio of average plate flux to average cell flux (peaking factor) for the two cases differed by <1% for all plates in all groups. However, CALHET calculations were performed for ZPR-III Assembly 48 using both the standard DB² effective absorption technique and a technique that incorporates the DB² term in a nonleakage factor that reduces the group-dependent collision probabilities as suggested by ANL.⁸ The latter method reduced the flux distribution heterogeneity effect by 0.12% $\Delta k/k$, a trend similar to that observed in the GEDANKEN experiments.⁹

The reference treatment of scattering anisotropy was to use a transport correction to the total cross section and assume isotropic scattering (P1 option of MC²). Support calculations by ORNL comparing the P1 and P3 scattering options of ANISN for Assembly 48 gave negligible differences, thus defending the reference approach.

Sensitivity to cross-section differences depends on which cross sections are affected. High-energy peaking factors in fissile plates are particularly sensitive to parameters that affect neutron production such as group fission spectra χ_g and neutrons per fission ν_g . Calculations with ENDF/B Version I and a modified library (PUCR)¹ gave high energy peaking factors in the Pu plate of ZPR-III Assembly 48 of 1.25 and 1.30, respectively. The group 1 ratio of χ_1/ν_1 for PUCR to ENDF/B is 1.07.

The bilinear weighted technique formally gives rise to a weighted fission term of the form $\langle \chi \nu \Sigma_f \rangle_g$, a two-dimensional matrix showing dependence on incoming (j) and outgoing (g) neutron energy. The reference approach for a criticality calculation was to select a single j at the

value of an important group and use the resulting χ_g for both the bilinear weighted core and unweighted blanket cross sections. Two sensitivity calculations were performed that (a) allowed a two-group dependence of χ_g on j , and (b) used a single j -value weighted χ_g for the core and the original unweighted χ_g for the blanket. Results for Assembly 48 indicated that use of the single j reference approach for core and blanket introduced a criticality error of only $\sim 0.02\% \Delta k/k$.

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A Comparative Analysis of Methods for Calculating Heterogeneity Effects in Fast Critical Experiments—Part II: Analysis, E. M. Page, R. E. Mueller (APDA)

Heterogeneity behavior for selected plate-type fast criticals has been analyzed in detail.¹ As discussed in the companion summary,² self-shielding heterogeneity effects were determined by a volume-weighting technique, while several methods were employed to determine the flux distribution effects.

Computed heterogeneity effects on system criticality and the sodium void coefficient for a central test region for three assemblies are given in Table I. The net positive self-shielding effect is due primarily to the increased self-shielding of ²³⁵U resonances chiefly in the range 0.1 to 15 keV. Flux distribution effects on criticality by the DTF-IV and CALHET weighted methods agree since the two associated codes compute similar intra-drawer flux distributions. Agreement of the perturbation approach with the DTF-IV Weighted method defends the assumption that the heterogeneity effect on

criticality can be treated as a first-order perturbation. The reason for the larger criticality effect given by the Bilinear Weighted method for Assembly 6 is not known except to note that the maximum high-energy flux advantage factor is large (1.47) compared to that for Assembly 48 (1.30). The Scaler Bilinear method overpredicts the flux distribution effect due to neglect in the weighted cross sections of the effect of net cell leakage of neutron importance.

Heterogeneity effects on the central Na-void coefficients for Assemblies 48 and 6 are extremely sensitive to methods due to the sensitivity of the coefficients themselves as a result of cancellation of energy-dependent effects of opposite sign. This sensitivity can be noted in the large percentage difference between DTF-IV and CALHET weighted results for Na voiding while good agreement was obtained for effects on criticality. The major flux distribution effects are a negative component due to the reduction in high-energy flux peaking in fissile plates upon Na voiding and a positive component from the enhanced high-energy adjoint flux slope due to the above-cited flux peaking. The negative effect dominates in Assemblies 48 and 51, the positive effect in Assembly 6. The Perturbation Approach neglects the adjoint flux slope increase, and thus predicts heterogeneity effects on Na voiding that are substantially too negative, even for the less sensitive void coefficient of Assembly 51.

TABLE I
Heterogeneity Effects on Criticality
and Central Sodium Voiding

Method (Assembly)	Criticality (% $\Delta k/k$)	Central ^a Na Void (lb/kg)
Self-Shielding Effect		
Volume Weighted (ZPR-III, Assembly 48)	0.49	-6.13
(ZPR-VI, Assembly 6)	0.33	-0.26
Flux Distribution Effect		
DTF-IV Weighted (ZPR-III, Assembly 48)	1.38	-0.82
(ZPR-VI, Assembly 6)	0.42	0.28
(ZPR-III, Assembly 51)	1.02	-3.99
Perturbation Approach (ZPR-III, Assembly 48)	1.48	-4.01
(ZPR-VI, Assembly 6)	0.52	-0.32
(ZPR-III, Assembly 51)	1.02	-6.99
CALHET Weighted (ZPR-III, Assembly 48)	1.35	-0.48
(ZPR-VI, Assembly 6)	0.49	0.47
Bilinear Weighted (ZPR-III, Assembly 48)	1.46	0.88
(ZPR-VI, Assembly 6)	1.18	--
Scaler Bilinear Weighted (ZPR-III, Assembly 48)	2.82	--
(ZPR-VI, Assembly 6)	4.65	--
CALHET (ZPR-III, Assembly 48)	1.49	--
(ZPR-VI, Assembly 6)	0.51	--

^aRefers to region A + B for Assemblies 48³ and 6⁴ and to the front 7 in. of the central 9 drawers per half for Assembly 51.⁵

TABLE II

Effect of Plate Configuration Changes on Reactivity (β) and Central Sodium Void Coefficient (β /kg)

Method	ZPR-III, Assembly 48 ³				ZPR-VI, Assembly 6 ^{4,6}			
	Normal to Bunched		Normal to Unbunched		Normal to Bunched		Normal to Isolated SS	
	β	β /kg	β	β /kg	β	β /kg	β	β /kg
Volume Weighted ^a	13.3	-1.3	9.4	-1.7	10.60	-0.64	3.90	-0.13
DTF-IV Weighted	85.4	-3.0	-24.5	-0.2	16.52	-0.80	5.51	-0.16
Perturbation Approach	93.6	-3.8	--	0.1	15.84	-0.76	5.60	-0.16
Experimental	75.2 $\pm$ 1.4	-2.1 $\pm$ 0.4	-20.0 $\pm$ 1.4	0.5 $\pm$ 0.4	11.53	-0.68	4.47	-0.06

^aThe calculations of the effects of plate configuration changes utilize one-dimensional cylindrical geometry for system reactivity effects and one-dimensional spherical geometry for sodium-void coefficient effects.

Table II gives measured and calculated effects of plate configuration changes in test zones on criticality and Na voiding. DTF-IV Weighted results predict the experimental values quite well. Volume Weighted results differ most from experimental values for Assembly 48 since configuration changes for that assembly mainly alter the high-energy flux peaking, while changes for Assembly 6 mainly affect ²³⁸U resonance self-shielding. It should be noted that the agreement between DTF-IV Weighted and Perturbation Approach methods for Assembly 6 void coefficient changes is fortuitous, as evidenced by the lack of agreement between those methods for the absolute heterogeneity effect on Na void given in Table I.

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Neutron Noise, Pulses, and Waves

Experience with Subcriticality Determination by Noise Techniques in the FTR Engineering-Mockup Critical Experiment Program, Charles E. Cohn (ANL)

Noise methods have been applied to the measurement of subcriticality in the FTR Engineering Mockup critical experiment program¹ as a cross-check on subcriticality measurements by rod drop.² Initial efforts used the two-detector cross power spectral density (CPSD) method.³

Equipment for these measurements included, as high-efficiency neutron detectors, two NE-905 glass scintillators, each 1½-in. diam and 1-in. thick containing about 4.7 g of ⁶Li. Each photomultiplier output enters a General Radio multifilter, which allows selection of one of thirty 1/3-octave bands with center frequencies from 100 to 80,000 cycles. Both filters are set to the same frequency. Their outputs enter the two inputs of an analog multiplier. The multiplier output is integrated by a voltage-to-frequency converter, having two pulse outputs, one active on positive excursions of the multiplier output and the other active on negative excursions. These outputs are counted in two separate scalars, and the difference between the scalar readings indicates the CPSD.

The system is calibrated by exposure of one scintillator to white noise in the form of a ⁶⁰Co source. Both multifilter inputs are connected to the one detector output, and measurements are taken as described above. For each band, the reading taken on a reactor run is divided by the white-noise reading, calibrating out equipment frequency response.

With this equipment it was found possible to measure roll-off frequencies to 1 or 2% in 1 h, from near critical down to 2% subcritical. However, the CPSD method gave unsatisfactory results apparently because of departures from point kinetics in the reactor behavior. Near critical, observed roll-off frequency was up to 10% greater with the detectors in the center of the core than with the detectors in the reflector. Furthermore, measured subcriticalities disagreed sharply with rod-drop results, being low by as much as 30% of the latter.

Another method was developed and has given results agreeing with rod-drop measurements within 0.1% Δk. Like the methods of Seifritz⁴ and Albrecht⁵ it measures coherence between signals from two neutron detectors, but unlike their methods it uses the same equipment as for the CPSD measurements. While the CPSD at a given frequency is indicated by the difference between the two above-mentioned scalar readings, the coherence is indicated by their ratio.

This ratio R is given by

$$R = \frac{\int_0^\infty \int_0^\infty xy P(x,y) dx dy + \int_{-\infty}^0 \int_{-\infty}^0 xy P(x,y) dx dy}{\int_0^\infty \int_{-\infty}^0 xy P(x,y) dx dy + \int_{-\infty}^0 \int_0^\infty xy P(x,y) dx dy}$$

where $P(x,y)$ is the bivariate Gaussian probability density function

$$P(x,y) = \frac{1}{2\pi\sqrt{1-\gamma^2}} \exp \left[-\frac{1}{2(1-\gamma^2)} (x^2 - 2\gamma xy + y^2) \right]$$

(in standard measure). Here γ is the coherence or correlation, and ranges from zero with no correlation to unity with complete correlation. We obtain⁶

$$R = \frac{2/\pi(\gamma \sin^{-1} \gamma + \sqrt{1-\gamma^2}) + \gamma}{2/\pi(\gamma \sin^{-1} \gamma + \sqrt{1-\gamma^2}) - \gamma}$$

The value of γ corresponding to a measured R is determined from this formula.

With coherence determined, subcriticality in dollars is given by

$$\$_ = (1 + \$_1) \sqrt{\left(\frac{\gamma_1}{1-\gamma_1} \right) \left(\frac{1-\gamma_2}{\gamma_2} \right)} - 1,$$

where γ_2 is the coherence observed in the subcritical state, γ_1 is the coherence observed at the near-critical reference, and $\$_1$ is the subcriticality of the reference. (This is a variant of a previous formula⁴ that was restricted to a critical reference.) Results are not sensitive to detector location; good results have been obtained with the detectors either in the center of the core or in the axial reflector. The coherence measurements are made at the lowest multifilter frequency; i.e., 100 Hz.

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Education for Nuclear Criticality Safety

Education of Specialists in Nuclear Criticality Safety, D. L. Hetrick (U of Ariz)

Continuing expansion of the nuclear power industry will create an increasing demand for specialists in nuclear criticality safety, with particular emphasis on high-volume fuel fabrication and processing facilities. Operation of large facilities under the "broad license" concept¹ implies the assignment of safety responsibility to operating managers who are guided by a highly competent safety staff. Flexibility of plant operation implies that the safety services might be readily available. The safety specialists must be independent of production responsibilities, and the operating managers must have more than a superficial knowledge of criticality safety problems.

A broad spectrum of technical training will be needed, ranging from the level of technicians to senior engineers. However, the total number of such specialists will probably be small, so universities and technical schools will hardly be eager to offer programs leading to degrees in criticality safety engineering. At the present time, a university course devoted to the topic of criticality safety is quite rare.

A highly developed methodology exists today, but most of the personnel involved grew up with the industry. Where will the next generation of criticality safety specialists learn their trade? A partial answer is that criticality is a fundamental part of nuclear engineering, and that safety is an integral part of all sound engineering education. If, in addition, engineering educators are able to educate for the future (i.e., maintain soundness in a broad range of fundamentals while also teaching the capacity for change and life-long learning), then this may be enough.

The fact remains that nearly all experts in this field are currently employed by national laboratories or industrial organizations. Few university faculties include a specialist in criticality safety. The educational gap could be filled by short courses and institutes. The U.S. Atomic Energy Commission has not supported a proposal for such a course. The Nuclear Criticality Safety Division of the American Nuclear Society has considered the need for such courses, and promotion or possible co-sponsorship should be vigorously encouraged.

The lack of university research in this field is notable. Of 106 papers on criticality safety at a recent meeting of the American Nuclear Society, 5 had authors or co-authors with university affiliations. Universities are not well suited for experimental work in this area, but the dearth of academic analytical work is significant. This would seem to be a fruitful area for future industry-university cooperation.

It is interesting that most of the research effort has been concentrated on the identification of criticality limits rather than on the consequences of criticality accidents. This is due in part to the philosophy that criticality should be identified and then avoided by fiat. However, it does not answer the question: "What happens if the system becomes supercritical in spite of the administrators?" Recent experimental research in France is an important advance in this direction.² Analytical studies of criticality accidents would be highly suitable for university research.

In summary, it seems unlikely that universities or technical schools will offer specialized curricula in nuclear criticality safety. Instead, short courses and institutes for industrial personnel seem highly desirable, possibly as cooperative efforts by university departments, industrial laboratories, and technical societies. Finally, academic research in the field should be encouraged, both as a source of knowledge and as a stimulant for enrichment of the nuclear engineering curriculum in the area of criticality safety.

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Elements of a Good Nuclear Criticality Safety Program, J. D. McLendon (UCC-Nucl Div)

The primary objective of nuclear criticality safety is the avoidance of even a single nuclear criticality accident. The major prize we are working for in preventing such events is the lives of those directly involved.

A survey of the practices that have evolved during the past 25 years shows a number of common elements which may be grouped into (a) those relating to administrative practices, and (b) technical guidelines and general principles.

Administrative practices consist of the following:

1. *Firm Managerial Guidance:* It is imperative that management, beginning at the highest levels, understand and fully support the need and objectives of the nuclear criticality safety program.

2. *Clearly Defined Responsibilities:* Management must clearly establish responsibility for nuclear criticality safety. Supervision is as responsible for this aspect of safety as for production, development, research, or other functions.

3. *Establishment of Well-Defined Controls and Limits:* Important to the determination of appropriate controls and limits is a thorough and complete understanding of the processes, procedures, and credible mishaps. Every operation must be analyzed in detail to determine that the entire process will be subcritical under both normal and credible abnormal conditions.

4. *Precise Communications, Understandings, and Agreements:* The best in managerial guidance, designation of responsibilities, and limits are worthless if not precisely communicated, understood, and agreed upon.

5. *Provision of Technical Support:* Personnel skilled in the interpretation of criticality data and familiar with operations should be provided as advisors to supervision.

6. *Independent Audits and Appraisals:* Operations should be reviewed frequently to ascertain that procedures are being followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluations.

7. *Means to Correct Deficiencies:* Deviations in conditions important to nuclear criticality safety should be investigated and resolved through formally established mechanisms.

A number of general practices and technical guidelines have developed over the years. The following are examples:

1. In problem solving, geometry control is preferred. Reliance on administrative controls should be minimized.
2. The Double Contingency Principle has been employed for many years as a convenient and reasonably uniform method determining acceptability of proposed fissionable materials processes and activities.
3. As an adjunct to the Double Contingency Principle, it is advisable to adopt additional safeguards if impact on productivity and costs is negligible.
4. Whenever possible, subcritical limits should be established on bases derived from experiments, with adequate allowance for uncertainties in the data. As an alternative, limits may be derived from validated calculations, provided sufficient allowances are made for uncertainties in the data and the calculations.
5. The consensual principle should be used in the establishment of credibility of abnormal occurrences. Since credibility is a judgment factor, all interests should be heard; e.g., operations, engineering, safety, etc.
6. Do not compromise nuclear criticality safety for expediency of operation. To avoid this tendency, search for ways to permit the operationally preferred process at minimum cost and maximum safety. Try to make the "right way" the "easy way."

The proper makeup of nuclear criticality safety programs is emphasized by the requirements now a part of every license or contract involving fissionable materials. As a means of contractual understanding, the tendency appears to be to select a one-and-only way for each activity proposed. Unless provisions are made to allow equally acceptable alternative methods, an unfortunate stymie may be generated. Efforts should be continued to search out ways and means to assure reviewers and regulators of the adequacy of nuclear criticality safety and yet provide flexibility to permit effective management. In every way possible, a *good* nuclear criticality safety program must help bridge this gap.

The Competent Nuclear Criticality Safety Specialist in Industry, B. D. Wilson (GE-San Jose)

The criticality safety specialist in industry must be competent to establish practical limitations on the designs of equipment and the actions of men and machines that process fissionable materials so the probability of accidental criticality is infinitely small. Presumably, graduation from an accredited college with a degree in physics or nuclear engineering evidences a certain capacity for nuclear work. Does that education provide the skill and knowledge required by the criticality safety specialist? If not, what is the process by which he becomes competent? By what criteria may one assess competence in this highly specialized field of nuclear endeavor?

Hopefully, answers to some of these questions may become apparent by examining a profile of the competent criticality specialist. Opportunities for enriching the nuclear physics curricula for the student who desires the information may also be revealed. Unfortunately, the experimental data available to the criticality specialist, by which computational techniques may be verified, are not as complete as those available in other disciplines. Accordingly, calculational models are designed to yield conservative results and frequently incorporate the two contingency criteria and "Murphy's Law" to meet the objective of zero-criticality accidents.

In practice, the industrial criticality safety specialist functions as a member of a team, each member playing a well-defined role. Ultimately, of course, safety responsibility rests with line management of the operation. Changes in the facility, process equipment, or procedures that could involve criticality safety are formulated by operations and engineering personnel with appropriate inputs by the criticality safety specialist.

All functions of the criticality safety specialist may be sorted into two major categories: to establish practical limitations, and to maintain surveillance of the total criticality safety program. His limitation-establishing subfunction includes:

1. Advising manufacturing engineers, at the design-concept stage, as to what limitations are likely to be imposed.
2. Analysis of the proposed system design, considering mechanical and human behavior under normal and abnormal conditions.
3. Keeping abreast of emerging technology and methodology germane to the criticality safety function to improve his own capacity.

His "maintenance" subfunction includes periodic audits to determine that actual operations conform to the physical situations on which the criticality limitations have been based and periodic training lectures to reinforce operator discipline in observing established limitations.

In the author's considerable association with successful criticality safety programs and the specialists involved in them, he has formulated the following concept of those educational, professional, and personal factors that are desirable in the competent criticality safety specialist assigned to the large nuclear establishment.

1. Educational Factors

- (a) Academic—There is an unquestionable need for fundamental knowledge of the methods and techniques applicable to problems of reactor physics, reactor shielding, particle interaction and the like which are well covered by the university. Diffusion theory and transport theory, as well as Monte Carlo techniques, are believed essential to an adequate understanding of the suitability and the shortcomings of other semilempirical methods which, to our current belief, are obtainable only upon entry into the profession.
- (b) Industrial—In addition to the semilempirical methodology such as solid angle- k_{eff} , density-analog, and surface-density, industrial training must provide thorough understanding of fuel conversion, fabrication, and recovery processes.

2. Professional Factors

Professional identification and scoping is equally important to problem solving. Often, project delays or premium costs result not from computations of limits, but from system designs that were not analyzed in detail by the specialist or the engineers prior to actual operation. Analysis of priorities in his own work programs and capacity to select practical, economical controls are examples of nontechnical situations in the work place requiring application of subjective judgment.

3. Personal Factors

Contrary to idealized job descriptions, we do not require specialists who walk on water. Certain personal attributes, however, do enhance overall competency. Analytical ability and intuitive reasoning is so essential we have discussed those factors separately above. Self-motivation and initiative is desirable, of course, since the specialist usually works alone. Absolute integrity is an obvious requirement since the laws of physics cannot be negotiated out of existence. Finally, the specialist's communications must be clearly translatable to effective action without disrupting normal relationships within the organization.

Continuing Need for Criticality Data, Hugh C. Paxton (LASL)

In 1964, further need for experimental criticality data for safety guidance was questioned because of abundant information on highly enriched uranium, reasonable coverage of Pu and ^{235}U solutions, and the development of powerful computational techniques. The question was investigated by an *ad hoc* AEC committee headed by Ira Zartman, and results were summarized in a report entitled, "Technical Review of Basic Criticality Studies." This report provides a convenient point of departure for our exploration of the continuing need for criticality data. And because of this session's title, we shall define need in terms of safety applications, which, hopefully, will be of some educational interest.

The Zartman Committee identified a number of criticality areas in which data were considered deficient by AEC contractors and commercial organizations. The needs that were emphasized and subsequent changes of status are summarized in the following table. In addition to the items listed, the increased importance of LMFBR mixed-oxide fuel has led to new experimental data^{14,15} that may be sufficient for small-scale processing but probably will not satisfy ultimate requirements. Criticality problems with ^{238}Pu were afforded low priority by the Committee, and it seems that indirect experimental evidence can continue to suffice until large-scale applications develop. Ultimately, however, experimental checkpoints will be desired for ^{238}Pu , ^{244}Cm , and certain other transuranium actinides.^{16,17}

Needs Emphasized by Zartman Committee

As of 1964	Present Situation
1. <i>Low-Enrichment U</i> Transport of Fernald material. Fabrication and reprocessing of BWR, PWR fuel.	Probably satisfied by current Y-12 program. ^{1,2} Data sufficient for present scale, not for efficient reprocessing on predicted scale. ³ See items 2, 3, and 4.
2. <i>Nuclear Poisons</i> Fixed poisons for large-volume solution storage and processing. Soluble poisons for secondary protection.	ANS-8.5 realizes its glass-ring standard ⁴ is deficient for low-enrichment U, mixed fuel. Data on boron steel remain scant. ⁵ Generally satisfied for LMFBR fuel compositions by Hanford data, ^{6,7} for water-reactor fuel by Y-12 program, if limited to backup protection. But see item 3.
3. <i>Dissolver Problems</i> Emphasis on irradiated reactor fuel.	Present needs apparently satisfied, but predicted large-capacity requirements would require the use of soluble poisons for primary protection, and better data on poisoned lattices. ^{2,8}
4. <i>Air-Spaced Arrays</i> Storage requirements. Additional transportation needs.	ANS-8.7 realizes its storage guide ⁹ requires more confirmation for low-enrichment U, undermoderated mixtures. Effects of interspersed materials are still difficult to generalize. Note suggested needs for transport of low-enrichment UF_6 . ¹⁰
5. <i>Mixtures of ^{235}U and Th</i> Th-cycle breeder fuel.	No additional data, but little present need (should change some-time).
6. <i>Various ^{239}Pu-^{240}Pu Mixtures</i> Power-reactor compositions.	Satisfied reasonably by Hanford data. ¹¹⁻¹³

In 1964, representatives of commercial firms emphasized the need for further safety standards and guides. It appears that they would remain unsatisfied by contents of ANSI N16.1-1969, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," as well as the two proposed standards mentioned in the table.¹⁸ Another standards Work Group, ANS-8.9, on safe pipe intersections for fissile solution, recognizes the need for data beyond results that came recently from Oak Ridge¹⁹ and earlier from Rocky Flats.²⁰

Work Group ANS-8.11, "Validation of Computational Methods for Nuclear Criticality Safety," finds that some experimental parametric surveys that seemed adequate a decade ago are not sufficiently accurate for its purposes. Validation requirements suggested by that Group include:

"Results obtained by the method shall be correlated with relevant experimental data or with the results of a validated method. The correlation shall establish bias over a specified area of applicability."

The sign of sufficient experimental data will be validated computational methods that cover the processing, handling, and storage conditions that can be foreseen.

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Teaching Criticality Safety in the University, Warren F. Witzig, Samuel H. Levine (Penn St U)

The purpose of a university is to provide students with an in-depth understanding of their field of interest so they can (a) find a better job position in their vocation, and (b) attain greater satisfaction and success in their endeavors. University education should also be sufficiently broad that the graduating student will have several options to choose from in seeking a rewarding professional position.

To satisfy these conditions, each educational unit establishes its own nuclear engineering curriculum which may differ from university to university. However, all such curricula should exhibit a similar core, but the details of how the subject of nuclear criticality is taught may vary considerably among them. To investigate this topic, a questionnaire was sent to 28 universities, of which 17 responded. These responses are given in Table I for undergraduate curriculum and Table II for graduate curriculum. In both curricula, the concept of criticality is usually not the sole subject of any one course. Most generally, it is included as a topic in courses on reactor theory, radiological safety, and reactor safety. All undergraduate courses use the four-factor formula as a starting point and many do not go much further. Graduate courses emphasize the eigenvalue concept and give a greater depth of study.

Experimental or laboratory development of the criticality concept is rather complete in the graduate courses using computer simulation, subcritical experiments, or critical mass experiments. In undergraduate laboratories, the subcritical experiment is most frequently used (some 60%) with computer simulation used next often.

Few of the schools responding to the questionnaire have experienced much demand for personnel educated to be criticality specialists, as shown in Table II. It is evident that universities presently consider the field of nuclear criticality to be too narrow a specialty except for graduate students' theses or undergraduate projects.

TABLE I

Partial Tabulation of Undergraduate Curriculum To Teach "Criticality" Concepts

School	General Method			No. of Courses Teaching Criticality	Instructional Approach	Experimental or Lab Techniques
	Four-Factor Formula	Other Approach	Year Instruction Starts			
1. An Eastern University	Yes	--	Senior	2	G,M	A,SCM
2. Univ. Mich	Yes	Multigroup	Junior	3	C,G,M,O	A
3. La.Tech.Univ.	Yes	--	Senior	1	C,G,M	SCM
4. Idaho State U.	Yes	--	Senior	2	C,G,M	None
5. A Western University	Yes	--	Senior	2	C,G,M	A,SCM
6. Univ.of Fla.	Yes	--	Junior	3	C,G,M	SCM
7. M.I.T.	Yes	--	Senior	2	G,M	None
8. Ga. Tech.	Yes	--	Junior	2	C,G,M	CM, SCM
9. La. State	Yes	--	Jr. or Sr.	1	C,G,M,O	None
10. Univ.of Kansas	Subject given introductory treatment in one course.					
11. Kansas State U.	Yes	--	Junior	2	C,G,M,QC	A,CM,SCM
12. Univ. Cinn.	Yes	--	Junior	3	G,M	CM,SCM
13. Penn State U.	Yes	Multigroup	Junior	3	C,G,M,O	A,CM,SCM,O
14. A Mid-Eastern University	Yes	--	Senior	2	C,G,M	None
15. A Southern Univ.	Yes	--	Senior	2	G,M,Q,C	SCM
16. Lowell Tech Ins.	Yes	--	Junior	3	C,G,M,QC	None
17. Univ.of Ariz.	Yes	--	Junior	3	G,C,M,O,QC	S,SCM,O
18. UCLA	Yes	--	Senior	3	C,G,M	A,CM,SCM

C - chemical, i.e., concentration
 G - geometrical, i.e., "safe" geometry
 QC - quality control instruction employed

M - minimum mass
 O - Poison, reflector or other

A - analog or digital simulation
 CM - critical mass experiment
 SCM - subcritical mass experiment

TABLE II

Partial Tabulation of Graduate Curriculum To Teach "Criticality" Concepts

School	General Method			Instructional Approach	Experimental Technique	Number of Requests for Criticality Specialists Per Year
	Four-Factor Formula	Other Approach	Number of Courses			
1. An Eastern Univ.	Yes	--	2	G,M ^a	A, SCM ^a	3
2. Univ. of Mich.	Yes	Multigroup	3	C,G,M,O	A	None on record
3. Univ. of Berkley	Concepts included in two graduate courses, safety and environment.					0
4. Idaho State	Yes	--	1	C,G,M,O	SCM	0
5. A Western Univ.	None	--				0
6. Univ. of Florida	-	Eigenvalue	3	C,G	A, SCM	1-2
7. M.I.T.	Yes	Yes	3	C,G,M,QC	CM,SCM	0-2
8. Georgia Tech.	Yes	--	2	C,G,M,QC	A,CM,SCM	Very few
9. Louisiana Tech.	Yes	Multigroup	2	C,G,M,O	A,SCM	Very few
10. Kansas State	Yes	Multigroup	3	C,G,M,QC	A,CM,SCM	0
11. Univ. of Cinn.	Yes	--	3	G,M,QC	CM,SCM	0
12. Penn State	Yes	Eigenvalue	3	C,G,M,O	A,CM,SCM	0
13. A Mid-Eastern University	Yes	Yes	3	C,G,M	SCM	1
14. A Southern University	Yes	--	1	G,M,QC	SCM	0
15. Lowell Tech.Inst.	Yes	--	2	C,GM,QC	None	0
16. Univ. of Arizona	Yes	Yes	3	C,G,M,O,QC	A,CM,SCM	0
17. UCLA	Yes	--	3	C,G,M,QC	A,CM,SCM	0

^aSymbols defined in Table I.

Criticality Data and Safety Analysis

The Criticality of High Burnup Plutonium,
R. C. Lloyd, E. D. Clayton (PNL)

Plutonium isotopes other than ^{239}Pu are invariably present in all plutonium. The presence of these isotopes can significantly affect the criticality control limits established for ^{239}Pu . The effect on criticality of the even-neutron nuclides such as ^{240}Pu and ^{242}Pu is to cause the critical dimensions and masses for the isotopic mixture to be greater than that for ^{239}Pu alone. The effects are strongly dependent on the neutron spectrum, however, and any moderating diluent that may be present, such as water. The highly fissile odd-neutron nuclide ^{241}Pu will cause significant reductions in the critical dimensions and masses below that of ^{239}Pu . The minimum critical mass for ^{241}Pu has been calculated to be only 260 g for a concentration of 32 g/liter in water, and the limiting critical concentration for this nuclide in water is only about 5 g/liter.

A series of experiments has been completed with plutonium obtained from extreme high burnup fuel to establish the combined effects of the various isotopes of plutonium on criticality. The isotopic composition of the plutonium, as measured on August 11, 1971, was: ^{239}Pu -0.2 wt%, ^{240}Pu -41.4 wt%, ^{241}Pu -42.9 wt%, ^{242}Pu -10.8 wt%, and ^{243}Pu -4.7 wt%. A nonnegligible amount of ^{241}Am also was present to the extent of 1.08 wt% of the total Pu. The experiments were performed in a 24-in.-diam, water-reflected, cylindrical vessel. Ten data points were obtained with the critical heights and masses determined for various plutonium concentrations and nitric acid molarities in the plutonium nitrate solutions. Plutonium concentrations covered the range of about 40 to 140 g Pu/liter with nitric acid molarities ranging from about 1.5 to 5.

The data from the criticality experiments (conducted during August 9-19, 1971) with this plutonium have been reduced, yielding critical volumes for the plutonium nitrate solutions used in the 24-in.-diam, water-reflected, cylindrical vessel. The results are given in the table.

The experimental results were analyzed by means by the KENO Monte Carlo Code, utilizing ENDF/B-II cross-section data. In this case, the epithermal groups were obtained through application of the ETOG and GAM processing codes, and the thermal group data via the FLANGE code followed by the TEMPEST portion of the GAMTEC code. The k_{eff} values are given in the table.

TABLE I
Criticality of High Burnup Plutonium Solutions

Pu Conc (g/liter)	Nitric Acid Molarity	Critical Height in 24-in.-diam (cm)	Critical Volume (liter)	Computed k_{eff} with ENDF/B-II Data and KENO Code
140.0	5.02	54.70	158.3	0.993 $\pm$ 0.007
116.0	4.14	50.55	147.9	1.020 $\pm$ 0.006
99.3	3.67	48.26	141.2	1.012 $\pm$ 0.007
85.5	3.12	47.00	137.5	1.020 $\pm$ 0.005
75.6	2.74	47.29	138.3	1.006 $\pm$ 0.007
65.1	2.41	49.12	143.7	1.014 $\pm$ 0.007
56.3	2.05	52.83	154.5	1.022 $\pm$ 0.007
46.8	1.70	63.47	185.7	1.014 $\pm$ 0.005
40.6	1.46	80.92	236.7	1.007 $\pm$ 0.005

The checks against the integral critical experiments display a somewhat positive bias with the average computed k_{eff} being about 1% above unity.

The effects of the individual isotopes have been separately examined to determine their contribution to the reactivity. The computed effect for the 1.08 wt% ^{241}Am is to lower the value of k_{eff} from 0.8 to 1.2%.

Pu Conc (g/liter)	δk of ^{241}Am
40.6	-0.008
85.5	-0.010
140.0	-0.012

The effects of the other isotopes have been examined for the solution at 85.5 g Pu/liter. The results are:

	δk
^{238}Pu (0.2 wt%)	-0.001
^{242}Pu (4.7 wt%)	-0.009
^{240}Pu (42.9 wt%)	-0.371
^{241}Pu (10.8 wt%)	+0.103
^{241}Pu replaced by ^{239}Pu	-0.036

In the case of ^{241}Pu , the change in k_{eff} is that which would result from adding 10.8 wt% to the solution originally containing no ^{241}Pu , so as to result in the final concentration of 85.5 g Pu/liter. It is more appropriate to ask how much more reactive the ^{241}Pu would be relative to a like addition of ^{239}Pu . This is shown in the last entry. If the 10.8 wt% ^{241}Pu were replaced with a like amount of ^{239}Pu , k_{eff} would be less by about 3.6%; i.e., treating the ^{241}Pu as equivalent to ^{239}Pu would cause a significant error of nearly 4% in k_{eff} , which would be nonconservative (k_{eff} calculated about 4% too small). Thus, these results show the importance of considering the effects of all isotopes that may be present in criticality evaluations pertaining to high burnup fuel.

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Critical Experiments with Mixed Oxides of Pu and U Containing 8 and 30 wt% Plutonium, S. R. Bierman, E. D. Clayton (PNL)

In the development and demonstration of Liquid Metal Fast Breeder Reactors, engineering design decisions will bear heavily on the safety and economics of both reactor and nonreactor facilities. To provide the necessary technical background upon which to base these decisions, experimental criticality data are needed for plant applications which will also serve as benchmark data for calculational checks. A major benchmark program to provide reactor core design data is under way at the Argonne National Laboratory in the form of the ZPR experiments. These experimental assemblies are fast, heterogeneous systems containing essentially no hydrogenous material. However, data are also required over the entire neutron energy spectrum on clean, homogeneous Pu-U systems in well-defined, simple geometries for testing cross-section data and for evaluating computational techniques that are used in calculating geometry

TABLE I
Composition and Atom Densities of Experimental
Fuel and Reflector Material

Chemical Composition	Atoms/(b cm)		
	2.79 Atom H/x 29.3 wt% Pu ^a	51.8 Atom H/x 7.89 wt% Pu ^b	Reflector
²⁴¹ Am	1.017×10^{-5}	1.741×10^{-7}	---
²³⁹ Pu	2.186×10^{-3}	1.5279×10^{-5}	---
²³⁸ Pu	2.288×10^{-6}	0.0	---
²⁴⁰ Pu	2.927×10^{-4}	5.9415×10^{-6}	---
²⁴¹ Pu	5.875×10^{-5}	3.4810×10^{-7}	---
²⁴² Pu	6.751×10^{-6}	0.0	---
O	1.864×10^{-2}	1.8301×10^{-3}	1.4279×10^{-2}
²³⁸ U	6.162×10^{-3}	8.3775×10^{-4}	---
²³⁵ U	9.269×10^{-6}	1.0708×10^{-6}	---
H	2.432×10^{-2}	4.7194×10^{-2}	5.7116×10^{-2}
C	2.660×10^{-2}	4.5403×10^{-2}	3.5697×10^{-2}

^aIsotopic analysis made on 12-2-71. Experiments performed November 1971.

^bIsotopic analysis made on 1-21-71. Experiments performed December 1970-May 1971.

and heterogeneity effects and for studying various operating conditions to be encountered. These data are particularly needed for the nonreactor facilities where, because of the plutonium content and the processing steps associated with the fabrication and recovery of LMFBR fuels, conditions could be encountered that require a small quantity of material for criticality. The quantity of fissile material being handled, however, could be sufficient to potentially constitute large numbers of critical masses.

A research effort is presently under way at the Hanford Critical Mass Laboratory to provide the data on homogeneous PuO₂-UO₂ mixtures in well-defined, simple geometries. Experimental results from the initial two sets of fuels in this program have been reported¹ and consisted of PuO₂-UO₂-polystyrene mixtures containing 30 and 14.62 wt% Pu in the plutonium-uranium and having H/x atomic ratios of 47.4 and 30.6, respectively. Since that time, experiments have been completed with two additional sets of mixed-oxide fuel. One of these again had a plutonium enrichment of about 30 wt% but an H/x atomic ratio of only 3. The other set of fuel contained 7.89 wt% Pu in the plutonium-uranium and had an H/x atomic ratio of 51.8. The composition of the two fuels and the Plexiglas reflector is given in Table I.

Experimentally determined critical sizes and masses obtained from Plexiglas-reflected parallelepipeds, constructed from 2- × 2- × 2-in. PuO₂-UO₂-polystyrene fuel compacts, are shown in Table II. Calculated values of k_{eff} have been obtained for these critical systems utilizing ENDF/B-II cross-section data with diffusion theory, transport theory, and Monte Carlo-type calculations. Typical results obtained were: 1.027 ± 0.007 , with the Monte Carlo code KENO-I,² for the reflected 40.72- × 40.64- × 36.42-cm assembly of 7.89 wt% Pu fuel; and 0.991 with the diffusion theory code HFN,³ for a reflected infinite slab of this same material.

Although the data have been generated in support of the LMFBR program, the experimental results on the 7.89 wt% Pu-enriched fuel will be of use to the light-water reactor community. The plutonium enrichment range applicable to the LWR fuels is between 3 and 8 wt%, but experimental criticality data on homogeneous mixtures have been unavailable for these low enrichment systems.

TABLE II
Criticality Data for Homogeneous PuO₂-UO₂-Polystyrene Fuel Mixtures

Reflector	CRITICAL DIMENSIONS, CM ^a			CRITICAL MASS ^a	
	Length	Width	Height	kg of Pu	kg of U
Pu Enrichment = 7.89 wt%; Density of Fuel = 360 g (Pu+U)/l; H/Pu+U = 51.8 H/Pu = 659; ²⁴⁰ Pu Content of Pu = 8.0 wt%; ²³⁵ U Content of U = 0.151 wt%					
Plexiglas	40.72 ± 0.04	45.72 ± 0.09	32.89 ± 0.02	1.74 ± 0.01	20.29 ± 0.22
Plexiglas	50.90 ± 0.05	50.80 ± 0.10	26.40 ± 0.03	1.94 ± 0.02	22.63 ± 0.25
Plexiglas	61.08 ± 0.06	60.96 ± 0.12	22.66 ± 0.08 ^c	2.39 ± 0.02	27.88 ± 0.31
Plexiglas	61.08 ± 0.06	66.04 ± 0.13	22.09 ± 0.18	2.53 ± 0.03	29.53 ± 0.39
Plexiglas	61.08 ± 0.06	55.88 ± 0.11	23.22 ± 0.01	2.25 ± 0.02	26.26 ± 0.28
Plexiglas	61.08 ± 0.06	50.80 ± 0.10	24.37 ± 0.09 ^c	2.15 ± 0.02	25.06 ± 0.28
Plexiglas	40.72 ± 0.05	40.64 ± 0.08	36.42 ± 0.01	1.71 ± 0.01	19.97 ± 0.22
Plexiglasb	39.12 ± 0.28	39.12 ± 0.28	39.12 ± 0.28	1.70 ± 0.01	19.84 ± 0.32
Plexiglasb	-	-	14.83 ± 0.60	-----	-----
Pu Enrichment = 29.3 wt%; Density of Fuel = 3450 g (Pu+U)/l; H/Pu+U = 3 H/Pu = 10.4; ²⁴⁰ Pu Content of Pu = 11.54 wt%; ²³⁵ U Content of U = 0.151 wt%					
Plexiglas	40.72 ± 0.04	40.66 ± 0.21	28.86 ± 0.05	48.4 ± 0.5	116.5 ± 1.3
Plexiglas	35.63 ± 0.04	35.57 ± 0.18	36.50 ± 0.05	46.8 ± 0.5	112.8 ± 1.2
Plexiglas	50.90 ± 0.05	50.82 ± 0.26	22.10 ± 0.04	57.9 ± 0.6	139.4 ± 1.5
Plexiglas	61.08 ± 0.06	50.82 ± 0.26	20.49 ± 0.09	64.4 ± 0.8	155.1 ± 1.8
Plexiglas	61.08 ± 0.06	60.98 ± 0.31	19.01 ± 0.03	71.7 ± 0.8	172.6 ± 1.9
Plexiglas	61.08 ± 0.06	66.07 ± 0.34	18.49 ± 0.05	75.5 ± 0.8	181.9 ± 1.9
Plexiglasb	35.88 ± 0.25	35.88 ± 0.25	35.88 ± 0.25	46.7 ± 0.7	112.6 ± 1.7
Plexiglasb	-	-	11.97 ± 0.07	-----	-----

^aExperimentally determined corrections have been made, accounting for the reactivity effects of the cladding material and the stacking voids.

^bInfinite slab thickness obtained by extrapolation of data. Cube dimensions obtained by interpolation between critical assemblies.

^cAverage of two experiments.

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Theory-Experiment Tests Using ENDF/B Version II Cross-Section Data, L. E. Hansen, E. D. Clayton (PNL)

The availability of ENDF/B cross-section data, while providing a standard basis for the cross sections needed in criticality calculations, also creates the necessity for reexamining calculational results. This is required to assess the uncertainties in utilizing these data in criticality safety evaluations.

The ENDF/B data have been checked against integral experiments for a number of fast critical assemblies,¹ but little has been reported concerning the use of these

data when applied to the more general cases encountered in nuclear criticality safety; i.e., the conditions to be encountered external to the reactors require consideration of thermal and intermediate, as well as fast systems. Unfortunately, the checks made on fast critical assemblies provide little, or no, assurance of reliability in regard to the first two systems.

This paper describes the results of integral testing against more than 40 experiments selected for their simplicity of geometry, in which the fuel compositions extend over a wide range of moderation from thermal to the highly undermoderated system. The critical experiments examined were all homogeneous in nature, including a number of PuO₂-polystyrene and PuO₂-U(depleted) O₂-polystyrene mixtures, as well as a variety of plutonium nitrate solution experiments and two cases of mixed Pu-U nitrate solutions. The hydrogen-to-plutonium atomic ratios examined ranged from 0 to 1100. The ²⁴⁰Pu content of the plutonium varied between 2.2 and 18.35 wt%.

Since there are several different methods by which the ENDF/B cross-section data may be processed to provide broad-group-averaged cross-section data for criticality calculations, it is desirable that these be specified. It is unfortunate that these codes tend to insert their own bias into the calculated results. For the cases reported here, several different codes were used to facilitate comparisons: The epithermal groups were obtained through ap-

TABLE I
Computed Reactivities for Pu and Mixed Pu-U Systems Using ENDF/B-II Cross Sections

GEOMETRICAL CONFIGURATION	REFLECTOR	H/Pu ATOMIC RATIO	PERCENT ²⁴⁰ Pu	CRITICAL CONC (g Pu/liter)	HNO ₃ MOLARITY	TOTAL NO ₃ CONC (g NO ₃ /liter)	WATER CONC (g H ₂ O/liter)	DERIVED ^a CRITICAL DIMENSION (cm)	HFN COMPUTED MULTIPLICATION FACTORS WITH ENDF/B-II DATA AVERAGED BY	
									(ETOG-GAMTEC FLANGE-TEMPEST)	(ETOE, MC ² FLANGE-TEMPEST)
PuO ₂ -POLYSTYRENE SYSTEMS										
INFINITE SLAB	UNREFLECTED	5.0	11.46	2302	----	----	----	16.83(2)	1.045	1.029
INFINITE SLAB	UNREFLECTED	15.0	2.2	1120	----	----	----	16.09(3)	1.029	1.028
INFINITE SLAB	UNREFLECTED	49.6	18.35	367	----	----	----	23.85(4)	1.045	1.078
Pu NITRATE SOLUTION SYSTEMS (THIN STAINLESS-STEEL VESSELS ENCLOSE SPHERICAL SYSTEMS)										
SPHERICAL	WATER	354	4.6	73.0	0.2	86	971	14.568(5)	1.008	0.988
SPHERICAL	WATER	344	4.6	74.5	0.4	105	963	14.568(5)	1.007	
SPHERICAL	WATER	243	4.6	100.0	1.9	230	898	14.568(5)	1.002	
SPHERICAL	WATER	192	4.6	126.0	2.2	262	892	14.568(5)	1.013	
SPHERICAL	WATER	87	4.6	269.0	1.1	346	868	14.568(5)	1.015	
SPHERICAL	WATER	81	4.6	295.0	0.8	303	893	14.568(5)	1.027	
SPHERICAL	UNREFLECTED	668	4.6	39.0	0.4	64	978	19.318(5)	1.023	
SPHERICAL	UNREFLECTED	125	4.6	172.0	4.9	486	766	19.318(5)	1.030	
INFINITE SLAB	WATER	425	4.6	58.0	2.34	203	909	9.13 (6)	0.999	
INFINITE SLAB	WATER	383	4.6	58.0	5.0	397	791	9.78 (6)	0.988	
INFINITE SLAB	WATER	367	18.4	66.5	2.40	219	897	11.38 (7)	1.004	
INFINITE SLAB	WATER	144	18.4	160.0	3.04	360	839	11.21 (7)	1.024	
INFINITE SLAB	WATER	89	18.4	241.0	4.16	495	770	11.64 (7)	1.016	
INFINITE SLAB	WATER	46	18.4	412.0	4.14	664	684	13.01 (7)	1.043	
Pu-U (DEPLETED) OXIDE POLYSTYRENE SYSTEMS										
INFINITE SLAB	UNREFLECTED	158 (H/X = 47.4)	8.0	111.9 (30 WT% Pu)	----	----	----	21.70 (8)	1.006	1.022
INFINITE SLAB	UNREFLECTED	209 (H/X = 30.6)	8.0	84.8 (14.62 WT% Pu)	----	----	----	23.22 (8)	1.025	1.045
Pu-U (DEPLETED) NITRATE SOLUTION SYSTEMS (THIN STAINLESS-STEEL VESSELS ENCLOSE SYSTEMS)										
SPHERICAL	WATER	322 (H/X = 100)	4.6	70.9 (31.11 WT% Pu)	3.12	365	382	17.87 ^b	1.038	
SPHERICAL	WATER	708 (H/X = 224)	4.6	35.1 (31.65 WT% Pu)	1.49	182	920	19.31 ^b	1.034	1.026

^a RADIUS OF SPHERES AND THICKNESSES FOR SLABS

(2) (3) (4) (5) (6) (7) (8)

REFERENCES FOR APPROPRIATE CRITICAL EXPERIMENT DATA

^b HERETOFORE UNPUBLISHED DATA BY R. C. LLOYD AND S. R. BIERMAN; EXPERIMENTAL DETAILS AND OTHER NEW SOLUTION EXPERIMENTS NOW IN PROGRESS TO BE INCLUDED IN SUBSEQUENT PAPER.

plication of the ETOE and MC² processing codes and the thermal group data from the FLANGE code followed by the TEMPEST portion of the GAMTEC code. A second method for obtaining the epithermal groups involved the application of the ETOG and GAM processing codes. The criticality calculations were then carried out with the HFN multigroup (18 energy groups) diffusion theory code.

The results of some of these calculations are summarized in Table I. As is evident, the criticality factors (k_{eff}) do not reproduce the experimental data with an acceptable degree of accuracy. Also to be noted is the difference in results on using two different processing codes. The results, however, are in most cases conservative; i.e., k_{eff} is computed as larger than measured.

Such comparisons and checks against integral critical experiments will provide the information needed to improve and standardize cross sections and methods of calculation.

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Criticality Safety of Uranium Metal Scrap in Concrete Billets, Hans Toffer, Everett A. Weakley (DUN)

In the production of fuel for Hanford's N-Reactor, large amounts of uranium and zirconium metal wastes are generated in the form of lathe turnings, chips, and filings. Scraps of these metals are highly pyrophoric and require special handling procedures, and storage or shipping methods that preclude contact with air. This is accomplished by placing the materials in barrels filled with water. Since there is no control over the aggregation of uranium in such containers, criticality control considerations have to be based on a most reactive uranium water system for a particular enrichment. This treatment imposes severe restrictions on the quantity of material permitted in shipments to the scrap recovery facilities in the midwest.

After abandoning the oxidation process of such wastes due to an adverse environmental impact, a new method

was developed by Douglas United Nuclear.¹ The lathe turnings, saw fines, and other small pieces of uranium scrap are used as aggregate with masonry cement, and cast into thin-walled metal cans to form concrete billets. This approach represents a simple, but unique, way to prevent uranium or zirconium fires; to provide a structural unit that can withstand certain drop and fire tests; and to guarantee special criticality control safety because of the neutron absorption properties of concrete.

To obtain a Department of Transportation Permit for shipment of uranium-concrete billets in approved steel-strapped boxes (NLO boxes), it was necessary to determine allowable shipping limits.² Since no experimental critical mass data exist for such material aggregates, criticality control was established using well-verified computer codes such as the integral transport theory code; HAMMER³; and the KENO⁴ Monte Carlo code. Most analyses were carried out for uranium scrap enriched to 1.25 wt% ²³⁵U, the highest enrichment currently processed in the fuel fabrication facility. The most reactive uranium water system was compared with equivalent uranium rods latticed in concrete. Homogeneous mixtures of uranium in concrete over a range of uranium concentrations from 1.25 wt% ²³⁵U to 5 wt% ²³⁵U were evaluated and compared with values for homogeneous UO₂ water mixture.⁵ The results of these calculations are listed in Table I. In both cases, the water uranium systems were the most reactive. The material bucklings for homogeneous uranium-concrete systems were almost an order of magnitude less, and for the rod lattices the difference amounted to approximately two orders of magnitude.

Lattices of concrete billets with homogeneous concentrations of uranium were computed over the above-mentioned enrichment range. Inhomogeneous distributions of uranium in the billet (9.875-in. o.d., 10.25-in. height) were considered. The most reactive configuration for such billets was a concentration of the uranium in a ring on the periphery of the cylinder. Critical mass parameters for the most reactive billet lattice were used in the density analog method to compute the safe number of NLO shipping boxes containing uranium-concrete billets. These calculations were verified with the KENO Monte Carlo code.

The use of uranium-concrete billets significantly increased the shipping limits for 1.25 wt% ²³⁵U enriched uranium scrap. An example of the increase is the difference in the minimum critical mass of uranium for the most reactive rods in water, 680 kg U, and 7400 kg U contained in the most reactive-type billets.

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TABLE I
Material Bucklings for Various Cement Uranium Systems

Description	Moderator	V water/V solid	Conc of U g/cm ³	Enrichment wt% ²³⁵ U	B _m ² Max M ⁻²
0.6-in.-o.d. U rods hex lat	Water	2.32	18.9	1.25	64.0
0.6-in.-o.d. U rods hex lat	Concrete	—	18.9	1.25	0.4
Homeg Material UO ₃ ⁵	Water	—	3.0	1.25	17.7
Homeg Material UO ₃ ⁵	Water	—	2.7	2.00	67.0
Homeg Material UO ₃ ⁵	Water	—	2.3	3.00	106.3
Homeg Material UO ₃ ⁵	Water	—	1.9	5.00	155.0
Homeg Material U	Concrete	—	1.32	1.25	-6.5
Homeg Material U	Concrete	—	1.18	2.00	3.2
Homeg Material U	Concrete	—	1.00	3.00	10.5
Homeg Material U	Concrete	—	0.90	5.00	18.0
Homeg Billets	Concrete/water	0.172	3.53	1.25	-1.0
9.875-in. o.d. hex lat	Concrete/water	0.235	5.29	2.00	15.8
	Concrete/water	0.302	7.05	5.00	45.0
U ring on periphery of billet	Concrete/water	0.200	3.53	1.25	11.0

Density Effects on the k_{∞} of Shipping Packages, Ricardo Artigas (GE-San Jose)

Previous studies¹ had indicated that the infinite multiplication constant of certain systems that are representative of packages used for the transportation of essentially dry unirradiated low-enriched uranium compounds, such as the one described below, was practically independent of the density of the fissile material they contained. The results of this work negate the generalization of the above conclusion, since it is shown that the significance of the dependence of k_{∞} on the fissile material density is a strong function of the hydrogen content of the insulating material of the package. The results were obtained by using the KENO² Monte Carlo Code with the Oak Ridge modified version of the 16-group Hansen and Roach³ set of cross sections.

The system considered is composed of two 5-gal cans containing UO₂ of various densities, 5 wt% enriched in the ²³⁵U isotope, and with a hydrogen-to-uranium ratio of 0:45. The cans are contained in a 16-gal drum which is in turn centered and supported within a 55-gal drum by insulating material in which the hydrogen content was allowed to vary for calculational purposes. The zero-current boundary conditions are applied at the surface of the outer drum; that is, assuming an infinite array of such containers.

In a truly homogeneous UO₂-H₂O mixture, of given enrichment and H/U ratio, density variations in the mixture would not affect k_{∞} . A packaging system, such as the one described above, can be considered more nearly homogeneous, in the sense of a flatter thermal flux distribution, as the strength of the slowing down source external to the fissile accumulation decreases in relation to the source strength of the accumulation. The magnitude of this effect explains the relative insensitivity of k_{∞} to UO₂ powder density for the case of low hydrogen content in the mixture, as shown in Fig. 1.

The marked increase in the k_{∞} values as a function of UO₂ density for the higher hydrogen-containing insulating media is also seen from the same Figure. The departure from a truly homogeneous slowing down source indicates that the usage of the Hansen and Roach set of cross sections would tend to overestimate the importance of ther-

mal cross sections of the fuel region, since their averaging process is based on energy considerations and does not account for spatial self-shielding effects.

In the fuel region, the scattering cross section associated with each fuel atom has been maintained constant (H/U = 0.45). The finite dilution correction to the resonance absorptions of ²³⁸U is therefore constant for all densities.

The increase in k_{∞} with increasing density can be attributed in part to the increase in the system's thermal utilization, to the spectral differences of the neutron population, and in part to the conservative usage of the Hansen and Roach set. The relative importance of these

Fig. 1. k_{∞} vs UO₂ powder density for different hydrogen contents of the outer package.

three effects is not immediately evident and has not been assessed.

It can be concluded, however, that a shipping package infinite multiplication constant could change from a value of < 1.0 to a value > 1.0 by changes in the contained powder density (mass not conserved) for certain hydrogen contents of the outer package.

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BIG TEN, a 10% Enriched Uranium Critical Assembly: Kinetic Studies, L. J. Sapir, H. H. Helmick, J. D. Orndoff (LASL)

The Big Ten facility is a clean critical assembly recently installed at Los Alamos Scientific Laboratory. The primary objective of the Big Ten experimental program is the measurement of effective cross sections in a neutron spectrum that, although still energetic, is appreciably softer than the smaller LASL critical assemblies.

The cylindrical Big Ten assembly is mounted on a split table machine. The core is composed of a central section of homogeneous U(10) metal surrounded by a region of alternating 3-mm-thick U(93) and 27-mm-thick natural uranium plates simulating 10% material. A 6-in.-thick uranium reflector surrounds the core and effectively reduces sensitivity to external objects. The overall assembly is 33 in. in diameter and 38 in. long, and contains ~ 250 kg of ^{235}U .

Preliminary reactivity calculations were performed using TWOTRAN,¹ a two-dimensional transport theory code. Sixteen-group Hansen-Roach cross sections² yielded a value of k_{eff} 1.7% lower than experimentally deduced; while 36-group cross sections derived from the ENDF/B Version II data file³ yielded a value 4.7% low.

The experimental program has thus far emphasized kinetic studies; in particular, measurements of Rossi-alpha (decay of prompt-neutron fission chains) and reactivity by means of positive periods. For the Rossi-alpha measurements, a 1-in.-diam $\times$ 1-in.-long ^6Li -loaded glass scintillator was inserted into the core in place of an axial control rod. Data were compiled in a 30-channel time analyzer at various reactivities between -1.7% and $+0.7\%$.

Typical background-corrected, fission-chain-decay data are shown in Fig. 1. The data points were least-square fitted to determine the decay constant, alpha.

In Fig. 2 the resulting values of alpha are plotted against reactivity as determined from an inhour relationship based on Keepin-Wimett delayed-neutron data,⁴ and effective yields obtained from adjoint calculations. The linear relationship between alpha and reactivity and the extrapolation to 100 cents at alpha equals zero support the inhour equation parameters.

Fig. 1. Prompt fission chain decay.

Fig. 2. Big Ten Rossi-alpha versus reactivity.

The value of alpha of $0.116\mu\text{sec}^{-1}$ at delayed critical implies a prompt-neutron lifetime of $0.062\mu\text{sec}$ ($\beta_{\text{eff}} = 0.0072$).

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Nuclear Data Testing

Calculations of Reflected Fast Critical Assemblies and Comparison with Experiment, Morris E. Battat, Thomas J. Hiron (LASL)

Results of intercomparison calculations using the LASL-TD Cross-Section Library and ENDF/B Version II data have been reported in previous papers.^{1,2} In this paper, results of similar calculations using the preliminary ENDF/B Version III data will be presented. The assemblies chosen for this study were JEZEBEL and GODIVA, and two systems not considered previously, FLATTOP-49 and FLATTOP-25. The latter two assemblies are uranium-reflected versions (18- to 20-cm reflector thickness) of JEZEBEL and GODIVA, respectively. Specifications for these assemblies are given in Ref. 3. For the ENDF/B Data, transport calculations were made using an S_{16} angular quadrature, P_0 transport-corrected cross sections, and the 26-group structure specified for the ENDF/B benchmark problems. A correction (-0.002) was applied to the eigenvalues computed with the ENDF/B data to account for the difference between P_0 and the transport correction employed. All the calculations with the LASL-TD library were made with S_8 , P_1 transport-corrected cross sections, and 14 energy groups. As with the ENDF/B data, corrections were also made for the angular quadrature and scattering approximation used. LASL data were processed by the SIGMA code⁴ and the ENDF/B data were processed by the MC² code.⁵

Table I shows the calculated eigenvalues and central ratios for JEZEBEL, FLATTOP-49, GODIVA, and FLATTOP-25, using both cross-section sets. For all cases, experimental values of the ($^{238}\text{U}/^{235}\text{U}$) fission ratios are quoted from the work of Grundl and Hansen⁶; all other experimental ratios are from Ref. 7. The outstanding discrepancy in the eigenvalue calculations is the overprediction (about 1%) of both FLATTOP assemblies using the ENDF/B data. The LASL data calculate all eigenvalues accurately but are consistently low in the prediction of the ($^{238}\text{U}/^{235}\text{U}$) fission ratios. For both libraries the other calculated central ratios are generally in good agreement. The FLATTOP calculations made with the ENDF/B data suggest discrepancies in the ^{238}U reflector cross sections for these particular assemblies. Further

analysis of the data is being performed in an attempt to understand the source of these discrepancies.

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Comparison of ENDF/B Versions I, II, and Preliminary III, R. A. Karam, W. R. Robinson (ANL), M. Salvatores (CNEN-Italy)

ZPR-6 Assemblies 6A and 7,¹ two large and very similar LMFBR criticals fueled with uranium oxide and

TABLE I
Eigenvalues and Central Ratios for JEZEBEL, FLATTOP-49, GODIVA, and FLATTOP-25

	k_{eff}	$(^{238}/^{235})_f$	$(^{239}/^{235})_f$	$(^{238}/^{235})_f$
JEZEBEL				
LASL cross sections	1.0003	0.183	1.41	0.381
ENDF/B-III	1.0020	0.194	1.42	0.373
Experiment	1.000 ± 0.003	0.210 ± 0.005	1.49 ± 0.05	
FLATTOP-49				
LASL cross sections	0.9961	0.153	1.36	0.486
ENDF/B-III	1.0116	0.166	1.37	0.484
Experiment	1.000 ± 0.003	0.176 ± 0.004		
GODIVA				
LASL cross sections	0.9978	0.147	1.38	0.50
ENDF/B-III	1.0038	0.162	1.40	0.48
Experiment	1.000 ± 0.003	0.162 ± 0.004	1.42 ± 0.04	0.47 ± 0.02
FLATTOP-25				
LASL cross sections	0.9999	0.132	1.35	0.58
ENDF/B-III	1.0104	0.147	1.37	0.55
Experiment	1.000 ± 0.003	0.147 ± 0.004	1.40 ± 0.04	0.53 ± 0.02

TABLE I
Comparisons ENDF/B Versions I, II, and Preliminary III
(Assembly 6A)

	Version I	Version II with Version I SS	Version III ^a	Experiment ^b
k_{eff}				
2-D Cylinder ^c 1-D Sphere ^c	0.9818 0.9818	- 0.9698	- 0.9700	1.000
Reaction Ratios				
$^{238}\text{U}_f / ^{235}\text{U}_f$ $^{238}\text{U}_c / ^{235}\text{U}_f$	0.0213 0.149	0.0209 0.149	0.0226 0.144	0.023 0.124
Capture-to-Fission Ratio, α				
^{235}U	0.2823	0.2828	0.2847	-
Central Reactivity Worths, Δk in lh/kg^d				
^{235}U ^{239}Pu ^{10}B Ta Na ^{238}U	+48.99 +62.69 -1189.79 -19.95 +0.10 -4.42	+48.53 +63.32 -1178.32 -19.77 +0.14 -4.41	+53.54 +70.80 - - -0.19 -4.58	+47.9 59.5 -820.4 -12.5 +0.137 -3.50

^aSee text for correction due to error in calculating the nuclear radius in MC².

^bThe experimental results quoted have been reported in Ref. 1.

^cHeterogeneities are not accounted for in any manner. The total heterogeneity effects in Assemblies 6A and 7 were calculated to be 0.95% and 1.35%.¹

^d1% $\Delta k/k$ in Assembly 6A = 458 lh.

mixed plutonium-uranium oxide, respectively, were chosen to compare the three available versions of ENDF/B.

The calculations were performed in the following manner: The MC²-generated ENDF/B Version I homogeneous cross sections were used to calculate k_{eff} of the as-built assemblies¹ with a 2-D diffusion code. The k_{eff} from the 2-D calculation was then used in a 1-D spherical calculation and the radius of the sphere which gave the same k_{eff} as the 2-D calculation was determined. This spherical radius was then held constant in the calculation of the spherical k_{eff} , using the MC²-generated ENDF/B Versions II and Preliminary III. The spherical central fluxes were used, in each case, to calculate the reaction ratios, the capture-to-fission ratio, and the central reactivity worths. The results are summarized in Tables I and II.

The heterogeneity effects in Assemblies 6A and 7 have been calculated to be¹ +0.95 and +1.35% $\Delta k/k$. Additionally, in calculating the potential cross sections and the penetration factor in the unresolved energy range, two different formulae are used to calculate the nuclear radius in Versions II and Preliminary III. Presently, only one nuclear radius formula can be used in the MC² code and consequently the calculations reported here include this systematic error. An evaluation of the magnitude of this error was made by Pennington² who calculated, as a

function of energy for the infinite dilution case, the ratio of the cross section computed with one nuclear radius to that computed with two radii. His results were used to calculate this effect on Assemblies 6A and 7 using Preliminary ENDF/B Version III. The results were an increase in k_{eff} of 0.65 and 0.72% $\Delta k/k$, respectively. The nuclear radius correction does not apply to ENDF/B Version I.

The heterogeneity correction to ENDF/B Version I would make k_{eff} of Assemblies 6A and 7—0.9913 and 0.9751, respectively. The heterogeneity and nuclear radius corrections to ENDF/B Versions II and Preliminary III would make k_{eff} for Assembly 6A—0.9868 and 0.9870, and for Assembly 7—0.9676 and 0.9920, respectively. Based on these results, it is seen that most of the parameters calculated with Preliminary Version III are closer to experiment than the same parameters calculated with Version II; sizable discrepancies remain, however, especially in the reaction rates.

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TABLE II
Comparisons of ENDF/B Versions I, II, and Preliminary III
(Assembly 7)

	Version I	Version II ^a with Version I SS	Version II ^b	Version III ^b	Experiment ^c
k_{eff}					
2-D Cylinder ^d 1-D Sphere ^d	0.9616 0.9616	- 0.9459	- 0.9469	- 0.9713	1.000
Reaction Ratios					
$^{235}\text{U}_f/^{239}\text{Pu}_f$	1.140	1.136	1.140	1.094	1.065
$^{238}\text{U}_f/^{239}\text{Pu}_f$	0.0235	0.0224	0.0230	0.0218	0.0234
$^{238}\text{U}_c/^{239}\text{Pu}_f$	0.1664	0.1688	0.1667	0.1584	0.143
Capture-to-Fission Ratio, α					
^{239}U	0.2446	-	0.2798	0.3070	0.355 ^e
Central Reactivity Worths, Δk in lh/kg ^f					
^{235}U	+172.0	+172.0	+169.9	+168.6	-
^{239}Pu	+202.9	+201.5	+198.9	+204.2	171.5
^{10}B	-3470.0	-3342.0	-3450.0	-	-2868.7
Ta	-66.8	-64.2	-68.0	-	-43.07
Na	-6.21	-6.94	-6.06	-8.63	-6.68
^{238}U	-12.83	-12.48	-12.48	-12.35	-8.72

^aTo compare the effects of including the 665 resolved resonances in Fe, Ni, and Cr in Version II with no resolved resonances in Version I.

^bSee text for correction due to error in calculating the nuclear radius in MC2.

^cThe experimental results quoted have been reported in Ref. 1.

^dHeterogeneities are not accounted for in any manner. The total heterogeneity effects in Assemblies 6A and 7 were calculated to be 0.95% and 1.35%.¹

^eThe measured value of α was provided by M. Bretscher.³

^f1% $\Delta k/k$ in Assembly 7 = 1035 lh.

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Heterogeneous Calculations of ZPR-III Assembly 48 by Monte Carlo, S. N. Cramer (ORNL)

A series of Monte Carlo calculations has been performed on one-, two-, and three-dimensional mockups of the ZPR-III Assembly 48 standard drawer. Full-drawer multigroup fluxes and cross sections have been compiled, as well as individual-plate cross sections and the ratios of the plate flux-to-the average drawer flux. These results are compared with those calculated with one-dimen-

sional discrete ordinates by Page et al.¹ The 1-D results agree in all cases, as do the 2-D and 3-D results for the full drawer. The 2-D and 3-D calculations of individual-plate cross sections reflect the detail in the drawer geometric cross section; however, this detail increased the variance of the flux ratios such that it is difficult to distinguish heterogeneity effects from statistical effects in the 2-D and 3-D cases.

In addition to the calculation of an infinite array of standard drawers, the complete assembly has been mocked up by including the safety, control, and reflector drawers. The eigenvalue calculations in Table I show the effects of geometric detail, ENDF/B data set, and cross-section weighting. These results are an extension of preliminary calculations reported in Ref. 2. The spherical model calculation from Ref. 1 includes a 1.9% $\Delta k/k$ heterogeneity effect.

All calculations presented here were performed with the ESP code utilizing the fictitious scattering model.^{3,4} An outline of the transport and collision selection process is shown with the 2-D drawer cross section in Fig. 1.

TABLE I
Eigenvalue Results for the ZPR-III Assembly 48

Case	ENDF/B Set	Weighting	Eigenvalue
Homogeneous drawers	I	Σ_T^{-1}	0.943 ± 0.005
Heterogeneous drawers	I	Σ_T^{-1}	0.947 ± 0.006
Heterogeneous drawers	I	Constant	0.981 ± 0.007
Heterogeneous drawers	I	$\left\{ \begin{array}{l} (\Sigma_T + \Sigma_e)^{-1}, \text{ core} \\ \Sigma_T^{-1}, \text{ reflector} \end{array} \right\}$	0.979 ± 0.005
Heterogeneous drawers	II	$\left\{ \begin{array}{l} (\Sigma_T + \Sigma_e)^{-1}, \text{ core} \\ \Sigma_T^{-1}, \text{ reflector} \end{array} \right\}$	0.970 ± 0.005
Heterogeneous drawers	II Modified	$\left\{ \begin{array}{l} (\Sigma_T + \Sigma_e)^{-1}, \text{ core} \\ \Sigma_T^{-1}, \text{ reflector} \end{array} \right\}$	0.997 ± 0.005
Spherical homogeneous (diffusion theory)	a	a	0.996^b

^aSee Ref. 1.

^bIncludes heterogeneity and transport theory corrections.

SELECT X FROM $\Sigma_{MAX} e^{-\Sigma_{MAX} X}$ AT A
 $R \leq \frac{\Sigma_B}{\Sigma_{MAX}}$ ACCEPT B AS COLLISION SITE
 $R > \frac{\Sigma_B}{\Sigma_{MAX}}$ NO COLLISION, SELECT NEW X AT B
 FROM $\Sigma_{MAX} e^{-\Sigma_{MAX} X}$

Fig. 1. Plate detail and scattering model.

When a trajectory crosses an outer boundary of a drawer, it is transferred in such a manner that it enters the same drawer at the point and direction corresponding to where it would have entered an adjacent drawer. This model allows the practical mockup of repeating arrays of thin plates in that each unique cell need be described only once, and the tracking process requires no boundary crossings between collisions.

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Effect of ENDF/B-II Data for ^{238}U on Critical Correlations, U. P. Jenquin, B. R. Leonard, Jr., V. O. Uotinen, K. B. Stewart (PNL)

An evaluation has been made of the effects on calculated values of k_{eff} for critical thermal systems when ENDF/B-II cross-section data¹ are substituted for data from the Battelle-Northwest Master Library (BNWML).² The data on the BNWML are similar to ENDF/B-I data.

The calculations were made using the HRG3,³ BRT-I,⁴ and HFN⁵ codes. For Al-Pu systems, the values of k_{eff} (Table I) are nearly the same using the two sets of cross-section data. The values of k_{eff} for the UO_2 and UO_2 - PuO_2 lattices are significantly less using ENDF/B-II cross sections than the values obtained using BNWML cross sections.

Detailed analysis of the components of the calculations reveals that the differences in the results are primarily due to the nonthermal ^{238}U cross sections. McCrosson⁶ observed similar differences in calculated values of k_{eff} for uranium metal systems. The nonthermal cross-section data for ENDF/B-II ^{238}U increases the calculated nonthermal capture rate in ^{238}U by 10%. The infinite dilute capture resonance integrals for ^{238}U are 269.5 and 276.9 b for BNWML and ENDF/B-II data, respectively. The portion of each resonance integral subject to self-shielding is 265.6 and 270.7 b, respectively. The difference makes up the floor in the HRG3 library and is not self-shielded. These values are 3.9 and 6.2 b, respectively. For the UO_2 and UO_2 - PuO_2 systems under consideration, both sets of data give an effective resonance integral of about 15 b excluding the floor. Thus, the total effective resonance integral for ^{238}U is 21 b with ENDF/B-II data vs 19 b with BNWML data. This difference gives lower values of the resonance escape probability and lower values of k_{eff} . The effect becomes larger as the moderator-to-fuel volume ratio decreases.

The ^{238}U data on the BNWML does not contain any p-wave cross sections in the resolved and unresolved resonance regions (1 eV to 2.5 keV). The p-wave cross-section data on ENDF/B-II in this energy range gives a resonance integral of 1.3 b. Thus, about 60% of the difference (2.3 b) between the HRG3 floors using BNWML and ENDF/B-II cross-section data is attributable to omission of p-wave cross-section data in the resonance region on the BNWML. The remainder of the difference is due to different evaluations of the measured data. Most sets of cross-section data have not contained p-wave data in the past.

If the identification of the p-wave resonances in ^{238}U is correct, the description of the ENDF/B-II cross-section data is better than the BNWML cross-section data. Since the values of k_{eff} calculated with ENDF/B-II data are as much as 3.8% lower than those calculated with BNWML data, which generally give good values of k_{eff} , there must be significant errors in the theoretical methods, the assignment of the ^{238}U p-wave resonances, and/or the other cross sections. The components of the floor are generally not self-shielded in slowing down calculations; hence, the calculated values of k_{eff} are very sensitive to this cross-section data.

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TABLE I
Effect of Changing Cross Sections on Calculated Values of k_{eff}

Fuel Type	Lattice Spacing (in.)	Moderator-to-Fuel Volume Ratio	Calculated Value of k_{eff}	
			BNWML	ENDF/B-II
2.734 wt% UO_2	0.405	1.048	1.0035	0.9658
	0.573	3.357	0.9989	0.9850
Al 5 wt% Pu	0.85	1.905	1.0067	1.0065
	1.30	6.173	1.0069	1.0074
UO_2 -2 wt% PuO_2 (8% ^{240}Pu)	0.800	1.515	0.9920	0.9634
	1.386	7.054	0.9979	0.9898
UO_2 -2 wt% PuO_2 (24% ^{240}Pu)	0.800	1.515	0.9942	0.9676
	1.386	7.054	0.9971	0.9893
UO_2 -4 wt% PuO_2	1.050	3.624	1.0035	0.9920
	1.386	7.271	1.0062	0.9991

Computational Methods and Techniques

Application of the Probability Table Method to Multigroup Calculations, D. E. Cullen (BNL)

When considering the transport of neutrons in a given energy interval, a neutron entering the interval (e.g., born or scattered into the interval) has a well defined energy at which there is a well-defined cross section. Once within the interval, a neutron will transport in space and thereby contribute to the flux. This cycle of spatial transport followed by an interaction will continue until either the neutron disappears (e.g., is absorbed) or is transported out of the interval. During this process of transporting across the energy interval, the neutron can encounter a range of cross-section values as illustrated in the upper portion of Fig. 1

In the multigroup approach,¹ the exact model is replaced by an approximation that defines average values over given energy intervals. This introduces two effects. First, there is an uncertainty in the energy of the neutrons within the group. Second, the neutrons are constrained to interact with a single cross-section value (i.e., the group-averaged cross section). In general, the effect of uncertainty in the neutron energy can be compensated for by ensuring that neutrons undergo the correct average number of interactions in transporting

across the group or by using more energy groups. Defining an average cross section is a more difficult problem, particularly in energy intervals such as the unresolved region where increasing the number of groups may not improve the approximation (e.g., the result may simply be a large number of groups all with the same average cross section). Since it is usually possible to define the energy transport in terms of the average number of collisions to transport across the group, the multigroup model may be thought of as illustrated in the middle of Fig. 1.

One possible technique for improvement of the multigroup scheme is to use more than one group-averaged cross section in a group. In this approach, rather than constraining the neutrons within the energy interval to interact with a single cross section they are allowed to interact with a number of cross-section "bands." The lower portion of Fig. 1 illustrates this model.

To use the multigroup, multi-band model, it is necessary to define the band cross sections within each group and the transfer matrix between bands and groups. Starting from the linear neutron Boltzmann equation, these quantities can be defined and the resulting algorithm can be cast into the form of an equivalent set of multigroup equations that can be solved by existing transport codes.² Furthermore, the use of cross-section bands becomes an adaptation of the Monte Carlo probability table method^{3,4} as applied to multigroup calculations in the unresolved region.

The effects of self shielding as a function of probability of survival (i.e., Σ_s/Σ_T) has been studied. A flat flux distribution was considered to be incident on a slab with a cross-section probability given by

$$p(\Sigma_T) = 0.15 - 0.006 (\Sigma_T - 10)^2 \quad \text{for } 5 \leq \Sigma_T \leq 15 \\ = 0 \quad \text{otherwise.}$$

In the case of a totally absorbing medium, a simple exponential attenuation within the various cross-section bands was obtained. The effect is a clustering of the flux in the low cross-section regions. The result is a continuous decrease in the group-averaged cross section from 10 b at the boundary toward a limit of 5 b at infinity. Using mean-free-path (mfp) units as defined by the cross section at the boundary (i.e., 1 mfp = 10 b), the following values of the group-averaged cross section occur:

Distance (mfp)	$\bar{\Sigma}_T$ (barns)
0.0	10.0
0.5	9.75
1.0	9.5
5.0	7.85
10.0	6.75

Calculations show that the difference in the scalar flux at 10.0 mfp units into the slab between using $\bar{\Sigma}_T = 10$ and the true self-shielded cross section is almost a factor of 10.

For the case of a pure scattering medium, a somewhat different result occurs. In this case, a reaction balance $[\Sigma_T(E)N(E)]$ is reached between the bands where the low cross-section bands feed the high cross-section band via spatial transport effects. The effect is to have a group-averaged cross section that varies from $\bar{\Sigma}_T = 10.0$ at the boundary with a flat flux over the group to $\bar{\Sigma}_T = 9.47$ at infinity with a $1/\Sigma_T(E)$ flux over the group.

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Data Applicable to Nuclear Criticality Safety Guidance

Criticality Control Parameters for Discharged Fuel, Hans Toffer, Robert L. Miller (DUN)

Handling and transportation of exposed reactor fuel at most nuclear facilities, including the Hanford N Reactor operated by Douglas United Nuclear, is governed by criticality control parameters associated with unirradiated fuel.¹ On exposure, the net fissile isotope inventory in most thermal reactor fuels is depleted and the reactivity of the fuel is reduced. Therefore, criticality safety based on unirradiated fuel and applied to exposed fuel elements is generally conservative. The margin of conservatism can become very significant, depending on the exposure of the particular fuel in question.

There is a definite lack of direct experimental critical mass data for discharged fuel. Generally, experimental

critical mass values exist for unirradiated fuel of similar geometry. In addition, exposure data for fuel columns or fuel bundles, experimental axial neutron flux profiles along a fuel column in a reactor, and isotopic compositions of discharged fuel can be obtained. The measured operational and isotope production related data have been correlated to within $\pm 5\%$ with the DCODE,² HAMMER,³ and EXTERMINATOR-2⁴ computer codes. Therefore, use of these computational techniques should provide a good prediction of the exposed fuel critical masses.

Critical mass calculations were carried out for exposed tube-in-tube fuel assemblies, and for PuAl cylindrical fuel elements. The isotopic discharge concentrations obtained with DCODE were used as input to the HAMMER code to generate multigroup lattice averaged cross sections and to calculate subsequently spherical critical masses of fuel at optimum lattice spacing

TABLE I
Minimum Critical Masses and Maximum Material Bucklings as a Function of Exposure

Fuel Type	Exposure MWd/T	Min Critical Mass kg, FBM ^a	Bm ² Max M ⁻²	Exposed MCM Unexposed MCM
Mark-IV U Fuel Assembly	0	6 640	21.2	1.00
Outer clad/core o.d. = 6.16 cm/6.03 cm	500	7 410	20.2	1.12
Outer clad/core i.d. = 4.32 cm/4.42 cm	1 000	7 910	19.3	1.20
Enrichment = 0.95 wt% ²³⁵ U	1 500	9 430	17.3	1.42
Inner clad/core o.d. = 3.25 cm/3.10 cm	2 000	11 900	14.6	1.80
Inner clad/core i.d. = 1.22 cm/1.32 cm	2 500	17 900	11.1	2.70
Enrichment = 0.95 wt% ²³⁵ U				
U core density 18.9 g/cm ³				
Zr clad density 6.4 g/cm ³				
Mark IA U Fuel Assembly	0	1 730	44.0	1.00
Outer clad/core o.d. = 6.11 cm/5.98 cm	500	2 020	40.7	1.17
Outer clad/core i.d. = 4.49 cm/4.62 cm	1 000	2 200	38.5	1.27
Enrichment = 1.25 wt% ²³⁵ U	1 500	2 450	36.2	1.42
Inner clad/core o.d. = 3.16 cm/2.96 cm	2 000	2 850	33.3	1.65
Inner clad/core i.d. = 1.24 cm/1.12 cm	2 500	3 370	30.0	1.95
Enrichment = 0.95 wt% ²³⁵ U				
U core density 18.9 g/cm ³				
Zr clad density 6.4 g/cm ³				
PuAl Fuel Elements^b	0	63.7	84.4	1.00
Clad/core o.d. = 3.78 cm/3.29 cm	5 000	77.4	76.3	1.21
PuAl core density 2.88 g/cm ³	10 000	103.2	67.0	1.62
Al clad density 2.7 g/cm ³	15 000	151.6	55.0	2.38
Unexp core comp 6 wt% Pu in Al	20 000	263.2	40.6	4.13
75 wt% ²³⁹ Pu, 20 wt% ²⁴⁰ Pu				
4 wt% ²⁴¹ Pu, 1 wt% ²⁴² Pu				

^aFBM = fuel bearing material or core material.

^bFission products were not included in the calculations of exposed PuAl fuel.

and at full water reflection. The results for Mark-IA, Mark-IV fuel assemblies, and for PuAl fuel elements are given in Table I. As expected, the critical masses of exposed fuel increase with exposure.

The computational study of exposed fuel critical masses included exploring the variation in the critical mass of discharged fuel as a function of time. The effective absorption cross section of long-lived fission products changes with time after discharge, as calculated using RIBD.⁵ A minimum is reached about 10 days after discharge. The effect on reactivity, however, is small compared to the fuel reactivity changes due to ²³⁹Pu buildup from ²³⁹Np. After about 30 days, the discharged fuel reaches the minimum critical mass for exposed fuel, about 10% of the discharge value. The results of the study summarized in Table I are based on isotope inventories 100 days after discharge.

Experimental verification of the exposed critical mass values for the special PuAl test block fuel will be obtained in the near future through a series of approach-to-critical measurements.

The results in Table I indicate that a more realistic assessment of critical control of exposed fuel can result in significant increases in the utilization of exposed fuel handling procedures and storage equipment without compromising criticality safety.

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Criticality of Plutonium-Uranium Nitrate Solutions, R. C. Lloyd, E. D. Clayton, S. R. Bierman (PNL)

Data are required on the criticality of Pu-U nitrate solutions¹ to accurately establish nuclear safety limits for use in processing and handling of LMFBR- and LWR-

type fuels. Since the fuel must be processed both safely and economically, it is necessary that criticality considerations be based on accurate experimental data. No experimental data on Pu-U nitrate solutions with low Pu concentrations (typical of LWR fuels) have heretofore been reported. These data, together with the principal data obtained to date on Pu-U nitrate solutions with Pu concentrations typical of LMFBR fuels, are included in this paper. The experimental program was designed both to provide the data needed for nuclear safety guidance and experimental benchmarks for checking calculational techniques and cross-section sets.

The experiments were performed with Pu-U nitrate solutions in water-reflected spherical and cylindrical vessels. The experiments in spherical geometry involved low ²⁴⁰Pu content (~4.6% ²⁴⁰Pu) material with the Pu concentration in the uranium being ~30 wt% of the total Pu+U concentration in the first two experiments and ~15 wt% in the latter one; those in cylindrical geometry were performed with higher ²⁴⁰Pu content (~23% ²⁴⁰Pu) material in which the Pu content was ~6%.

The experimental data are presented in Tables I and II. Criticality factors have been computed using ENDF/B

TABLE I

Criticality of Pu-U Nitrate Solution in Cylindrical Geometry
(Water-Reflected, Stainless-Steel Cylinder
i.d. = 61.028 cm; o.d. = 61.107 cm)

Plutonium ^{a,c} Conc (g/liter)	Uranium ^{b,c} Conc (g/liter)	Acid ^c Molarity	Critical Height (cm)	Critical Volume (liter)
30.7	395	0.45	50.27	147.03
29.1	399	0.44	54.66	159.89
27.4	403	0.44	61.04	178.56
25.8	408	0.37	70.49	206.20
24.4	412	0.36	84.86	248.23
23.7	413	0.36	94.56	276.62
30.5	395	0.5	51.41	150.38
29.2	378	0.6	52.29	152.98
27.4	355	0.6	56.87	166.36
25.8	335	0.7	60.99	178.39
24.4	316	0.7	67.59	197.71
23.1	300	0.7	76.96	225.13
22.4	285	0.7	89.61	262.13

^{a238}Pu = 0.07 wt%, ²³⁹Pu = 73.00 wt%, ²⁴⁰Pu = 22.80 wt%
²⁴¹Pu = 3.22 wt%, ²⁴²Pu = 0.91 wt%

^{b235}U = 0.6 wt%, ²³⁸U = 99.4 wt%

^cChemical composition data are preliminary.

TABLE II

Criticality of Pu-U Nitrate Solution in Water-Reflected Stainless-Steel Spheres

Approximate Percent Pu	Sphere Wall Thickness (cm)	Plutonium ^a Conc (g/liter)	Uranium ^b Conc (g/liter)	Acid Molarity	Critical Radius (cm)	Critical Volume (liter)	Critical Mass (g/Pu)
30	0.112	70.93	157.1	3.12	17.869	23.9	1695
30	0.122	35.05	75.7	1.49	19.304	30.14	1056
15	0.122	45.6 ^c	265 ^c	1.5	19.314	30.18	1376

^{a238}Pu = 0.01 wt%, ²³⁹Pu = 95.09 wt%, ²⁴⁰Pu = 4.66 wt%, ²⁴¹Pu = 0.22 wt%, ²⁴²Pu = 0.01 wt%

^{b235}U = 0.6 wt%, ²³⁸U = 99.4 wt%

^cChemical composition data are preliminary.

cross-section data in the HFN² or KENO³ codes. Epithermal energy groups were obtained through application of the ETOG and GAM cross-section processing codes, and the thermal-group data via the FLANGE code followed by the TEMPEST portion of the GAMTEC⁴ code. Results of these calculations illustrate the accuracy of these methods to the Pu-U nitrate solutions.

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Critical Parameter Calculations with High-Burnup Plutonium Solutions, R. D. Carter (ARHCO)

A number of different cross-section sets and criticality calculation codes are available for computing plutonium solution parameters. Several compilations of

critical plutonium solution parameters have been published^{1,2} for plutonium with up to 20 wt% ²⁴⁰Pu. The calculations were performed with only two plutonium isotopes, ²³⁹Pu and ²⁴⁰Pu (the other plutonium isotopes were considered as being the same as either ²³⁹Pu or ²⁴⁰Pu). This assumption would be conservative if all isotopes other than ²⁴⁰Pu were considered to be ²³⁹Pu, but if only ²⁴¹Pu is considered as ²³⁹Pu, and ²³⁸Pu and ²⁴²Pu are considered the same as ²⁴⁰Pu, the curves generally indicate too large critical geometries as shown by comparison with recent experiments.³ These experiments used high-burnup plutonium as plutonium solutions where the plutonium consisted of ²³⁸Pu(0.2), ²³⁹Pu(41.4), ²⁴⁰Pu(42.9), ²⁴¹Pu(10.8), and ²⁴²Pu(4.7), the numbers in the parentheses indicating the weight percent of the isotope in plutonium. Americium-241 was present in significant quantities (1.08 wt% of the total plutonium).

Cross-section sets were used from GAMTEC II⁴ (both the original and ENDF/B-II), GEM 4,⁵ HAMMER,⁶ and KENO II⁷ (Hansen-Roach) computer codes. The GAMTEC II cross sections were used in both the HFN² (this combination had been shown to be very accurate for plutonium with low ²⁴⁰Pu content) and KENO codes. Three different combinations of plutonium isotopes were used: (a) all five isotopes; (b) three isotopes with the ²³⁸Pu and ²⁴²Pu combined with the ²⁴⁰Pu; and (c) two isotopes with the ²⁴¹Pu combined with the ²³⁹Pu. The ²⁴¹Am content was included in only one set of calculations since it was only available on that set.

The table lists the k_{eff} values calculated by the various combinations. Those values not including ²⁴¹Am should be reduced by about 1% in k_{eff} as indicated by the ENDF/B II-HFN data. The two-isotope calculation in the first column of k_{eff} values is the same combination used

TABLE I

Cross-Section Source		Calculated k_{eff}						
		GAMTEC II (Original)		GAMTEC II (ENDF/B II)	GAMTEC II (ENDF/B II) W/Am	Hansen-Roach	GEM 4	HAMMER
Code for k calc		HFN	KENO	HFN	HFN	KENO II	GEM 4	HAMMER
Pu g/liter	No. Iso							
140	5	1.0435	1.0410 ± 0.0079	1.0395	1.0280	1.1013 ± 0.0069	---	---
116	5	1.0463	---	1.0406	1.0297	---	---	1.0412
99.26	5	1.0440	---	1.0365	1.0261	---	---	1.0403
75.64	5	1.0412	1.0498 ± 0.0060	1.0308	1.0215	---	---	1.0380
65.12	5	1.0396	1.0460 ± 0.0071	1.0273	1.0183	1.0643 ± 0.0079	---	1.0361
40.58	5	1.0393	1.0405 ± 0.0059	1.0215	1.0138	1.0529 ± 0.0060	---	1.0309
140	3	1.0329	1.0414 ± 0.0060	1.0271	---	1.0849 ± 0.0070	0.9930 ± 0.0131	1.0206
116	3	1.0334	1.0419 ± 0.0074	1.0266	---	1.0549 ± 0.0076	0.9920 ± 0.0094	1.0249
99.26	3	1.0303	---	1.0224	---	---	0.9820 ± 0.0136	1.0237
75.64	3	1.0263	1.0367 ± 0.0068	1.0155	---	1.0289 ± 0.0071	0.9922 ± 0.0141	1.0213
65.12	3	1.0244	1.0195 ± 0.0057	1.0123	---	1.0415 ± 0.0074	0.9794 ± 0.0133	1.0194
40.58	3	1.0239	1.0259 ± 0.0059	1.0057	---	1.0624 ± 0.0064	0.9869 ± 0.0112	1.0150
140	2	0.9770	0.9746 ± 0.0059	0.9872	0.9761	1.0156 ± 0.0064	---	0.9795
116	2	0.9797	---	0.9886	0.9782	---	0.9943 ± 0.0100	0.9854
99.26	2	0.9772	---	0.9856	0.9758	---	0.9789 ± 0.0122	0.9852
75.64	2	0.9735	0.9836 ± 0.0078	0.9798	0.9708	0.9731 ± 0.0065	0.9955 ± 0.0199	0.9836
65.12	2	0.9714	0.9692 ± 0.0070	0.9769	0.9683	1.0023 ± 0.0067	0.9755 ± 0.0107	0.9818
40.58	2	0.9684	0.9743 ± 0.0050	0.9693	0.9620	1.0115 ± 0.0058	0.9670 ± 0.0135	0.9762

to calculate critical solution parameters up to 20 wt% ^{240}Pu and has been shown to be accurate up to 8 wt% ^{240}Pu . For this high-burnup plutonium, it computed k_{eff} low by up to 0.04 Δk , while the five-isotope calculation is high by about 0.03 Δk . Similar variances are shown with other combinations. ENDF/B II cross sections showed a slight improvement. The five-isotope KENO II calculations (no ^{241}Am) were quite high.

These results show that the k_{eff} of high-burnup plutonium critical systems calculated by the use of the two-isotope approximation is too low and generally decreases with decreasing plutonium concentration. The results also show too high calculated k_{eff} values for the exact system. The currently available parameter studies referenced should be modified by the inclusion of an appropriate Δk factor to provide satisfactory critical parameters.

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7. G. E. WHITESIDES and N. F. CROSS, "KENO—A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide and Chemical Corporation (1969).
8. J. R. LILLEY, "Computer Code HFN—Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," NW-71545, General Electric Company (November 1961).

Criticality Experiments on LWR Mixed Oxides at 20 H:Pu+U, S. R. Bierman, E. D. Clayton (PNL)

Light-water reactor fuels normally consist of natural or depleted uranium enriched with up to 8 wt% Pu. Criticality data, even on uranium systems at this low a ^{235}U enrichment, are sparse. For homogeneous uranium enriched with plutonium, data are essentially nonexistent. To obtain criticality data in this area, experiments were recently completed with a $\text{PuO}_2\text{-UO}_2\text{-polystyrene}$ mixture containing 7.6 wt% Pu in the Pu+U. The H:Pu+U atomic ratio of the fuel mixture was 19.5, which is near the optimum concentration for minimum critical volume.

The critical experiments were performed with 2- $\times$ -2-in. fuel compacts, 2 in. and $\frac{1}{2}$ in. thick, assembled on a remotely operated split-table machine. Nuclide compositions and densities are given in Table I for the fuel and the Plexiglas reflector used in the experiments.

Inverse multiplication measurements were made on fully reflected parallelepipeds of the fuel, ranging from

TABLE I
Description of Fuel

Dimension of compacts without cladding	
length $\times$ width $\times$ height (cm)	$5.09 \times 5.09 \times 5.08 \pm 0.02$
Average dimension of stacked compacts with cladding	
length $\times$ width $\times$ height (cm)	$5.15 \times 5.15 \times 5.17 \pm 0.04$
Average thickness of cladding (cm)	
sides	0.015
top	0.030
bottom	0.030
Composition of cladding g/compact	
H	0.21
C	1.76
Cl	1.21
Composition of compacts [at./ (b cm)]	
^{241}Am	3.766×10^{-6}
^{238}Pu	9.535×10^{-7}
^{239}Pu	1.092×10^{-4}
^{240}Pu	3.688×10^{-5}
^{241}Pu	8.945×10^{-6}
^{242}Pu	4.689×10^{-6}
O	4.348×10^{-3}
H	4.155×10^{-2}
C	4.287×10^{-2}
^{235}U	2.897×10^{-6}
^{238}U	1.968×10^{-3}
Composition of reflector [at./ (b cm)]	
H	5.712×10^{-2}
C	3.570×10^{-2}
O	1.428×10^{-2}
PuO ₂ -UO ₂ particle size (mm)	
85%	<0.020
50%	<0.006
5%	<0.002
Polystyrene particle size (mm)	
95%	<0.225
50%	<0.150
5%	<0.050
Plutonium density (g/liter)	64
Uranium density (g/liter)	779
Oxide density (g/liter)	960

near cubic to relatively thin slabs. The critical sizes obtained from these measurements are shown in Table II. An experimentally derived correction, amounting to a 23.8% reduction in mass, has been applied to each assembly to account for the reactivity effects of the cladding material and the voids in the assembly of stacked blocks.

Initial calculations for one of the finite, fully reflected critical assemblies using the Monte Carlo code KENO¹ and the 18-group GAMTEC-II-averaged² ENDF/B-II cross sections have yielded a k_{eff} value of 0.966 ± 0.005 . Neutron transport calculations with 18-group GAMTEC-II-averaged ENDF/B-II cross sections used in the DTF-IV³ code have yielded 0.956 for k_{eff} of the reflected

TABLE II

Criticality Data for PuO₂-UO₂-Polystyrene Fuel Mixtures

843 g (Pu+U)/liter @ 7.6 wt% Pu; H/Pu+U = 19.5 atomic
H/Pu = 259; ²⁴⁰Pu content of Pu = 23 wt%; ²³⁵U content of U = 0.151 wt%

Reflector	Critical Dimensions, cm ^a			Critical Mass ^a	
	Length	Width	Height	kg of Pu	kg of U
Plexiglas	61.08 ± 0.06	66.17 ± 0.07	26.60 ± 0.09	6.87 ± 0.06	83.73 ± 0.72
Plexiglas	61.08 ± 0.06	61.08 ± 0.06	27.75 ± 0.10	6.61 ± 0.06	80.61 ± 0.71
Plexiglas	66.17 ± 0.07	66.17 ± 0.07	25.51 ± 0.10	7.14 ± 0.06	86.99 ± 0.77
Plexiglas	61.08 ± 0.06	50.90 ± 0.05	31.07 ± 0.11	6.17 ± 0.05	75.22 ± 0.66
Plexiglas	61.08 ± 0.06	55.99 ± 0.06	29.00 ± 0.12	6.34 ± 0.06	77.23 ± 0.67
Plexiglas	50.90 ± 0.05	50.90 ± 0.05	35.57 ± 0.13	5.89 ± 0.05	71.76 ± 0.63
Plexiglas ^b	45.17 ± 0.16	45.17 ± 0.16	45.17 ± 0.16	5.89 ± 0.06	71.76 ± 0.71
Plexiglas ^b	∞	∞	13.09 ± 0.04	---	---

^aExperimentally determined corrections have been made, accounting for the reactivity effects of the cladding material and the stacking voids.

^bInfinite slab thickness obtained by extrapolation of data. Cube dimensions obtained by interpolation between critical assemblies.

critical slab infinite in two directions. Whether or not these differences between experiment and theory are caused by the cross-section errors due to high ²⁴⁰Pu content or an overcorrection in the experimental data to account for the cladding and voids is not apparent without further analysis or corroborating evidence. Program plans in support of the Liquid-Metal Fast Breeder Reactor (LMFBR) include criticality experiments with high exposure fuel. Even though these experiments are to be performed at Pu enrichments greater than those encountered in the LWR fuels, they will provide additional data on the effects of ²⁴⁰Pu concentration.

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2. L. L. CARTER, C. R. RICHEY, and C. E. HUGHEY, "GAMTEC-II: A Code for Generating Consistent Multigroup Constants Utilized in Diffusion and Transport Theory Calculations," BNWL-35, Pacific Northwest Laboratory (1965).
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Criticality Safety of Some Water-Sprinkled Noncubic Arrays, G. R. Handley (UCC-OR)

The use of noncubic storage arrays for fissile material is highly desirable since it could result in a better utilization of floor space as well as provide easier access to all material. Therefore, five useful large arrays were investigated calculationaly: square based, vertical racks, rows, planar, and an array of arrays. These arrays show that consideration of optimum moderation will produce results that can be tolerated easily. The basic array unit in the investigation was a 20-kg uranium metal cylinder (H/D = 1), enriched to 93.2 wt% in ²³⁵U and centered in a 20-in. cubic cell to simulate 20-in. birdcage spacing.

A mockup was devised for the KENO¹ code which employed Hansen-Roach 16-energy-group cross sections²

(fission-spectrum-weighted hydrogen).³ All birdcage and other structural materials were neglected, and all voids were filled with water mist, the density of which was varied in a search for the maximum *k_{eff}*. When present, the water mist in the array reflector was at the same density as that which was inside the array. The calculational results and a brief description of each array and its reflector are presented in Table I.

A portion of the reflector for the square-based and the vertical-racks arrays consisted of two 12-ft-thick walls of water mist. They were replaced, in the most reactive case, by 6-in.-thick full-density water closely reflecting the array, and *k_{eff}* did not change. Therefore, since no conservatism results, the other arrays used full-density water walls on all four sides, and the *k_{eff}* increase due only to moderation was more nearly isolated.

The array of arrays on 8-ft edge-to-edge separation had a *k_{eff}* increase, when optimally moderated, of ~3% over the dry case. The maximum *k_{eff}* was not significantly greater than that of the single square array (0.9401 ± 0.0072 vs 0.9336 ± 0.0089). Also, for the dry condition, the maximum *k_{eff}* was not significantly greater than that of a similarly reflected single square array [0.9089 ± 0.0059 vs 0.8944 ± 0.0055 (not presented in Table I)]. However, when the edge-to-edge spacing of the dry array of arrays was reduced to 5 ft (the minimum dimension of the array), *k_{eff}* increased ~4% over the single array to 0.9303 ± 0.0080 (not presented in Table I). Therefore, the dry arrays are detectably interacting at this spacing.

It has been demonstrated that a 12% reduction in the mass of all array units in a reflected cubic array of highly enriched uranium will cancel the increase in *k_{eff}* produced by optimum-density water-mist moderation.⁴ For nearly critical cubic arrays, 12% in array units mass can be converted⁵ to ~4% in *k_{eff}*. The arrays presented with unchanging reflector (no water-mist walls) all had *k_{eff}* increases of 4% or less when optimally moderated.

1. G. E. WHITESIDES and N. F. CROSS, "KENO—A Multigroup Monte Carlo Criticality Program," CTC-5, Union Carbide Corp. (1969).
2. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Crit-

TABLE I
Summary of KENO Code Array Calculations*

Specific Gravity of Water Mist	$k_{eff} \pm \sigma$				
	6 × 6 × 3 High Square-Based Array ^a	Two 1 × 50 × 7 High Parallel Racks With a 5-ft Aisle Between ^a	Two 3 × 50 × 3 High Parallel Arrays With a 5-ft Aisle Between ^b	100 × 100 × 1 Planar Array ^b	Sixteen 6 × 6 × 3 High Arrays in Square Pattern With 8-ft Aisle Between ^b
0.00	-	0.7943 ± 0.0077	0.9093 ± 0.0066	0.9091 ± 0.0085	0.9089 ± 0.0059
0.01	0.8570 ± 0.0087	-	-	-	-
0.02	-	0.8507 ± 0.0076	0.9312 ± 0.0056	-	0.9080 ± 0.0072
0.03	0.8778 ± 0.0078	-	-	-	-
0.04	-	0.8287 ± 0.0068	0.9215 ± 0.0055	0.8757 ± 0.0083	0.9401 ± 0.0072
0.05	0.9003 ± 0.0092	-	-	-	-
0.06	-	0.8301 ± 0.0086	0.9192 ± 0.0078	-	0.9280 ± 0.0059
0.07	0.9225 ± 0.0072	-	-	-	-
0.08	-	0.8316 ± 0.0078	-	0.8481 ± 0.0073	0.9168 ± 0.0065
0.09	0.9336 ± 0.0089	-	-	-	-
0.10	-	-	-	-	0.9166 ± 0.0061
0.11	0.8951 ± 0.0076	-	-	-	-
0.12	-	-	-	0.8374 ± 0.0072	-
0.13	0.8763 ± 0.0080	-	-	-	-
0.16	-	-	-	0.8300 ± 0.0083	-
0.20	-	-	-	-	0.8440 ± 0.0065
0.30	-	-	-	-	0.8247 ± 0.0069

*The array units were 20 kg U(93.2) cylinders (H/D = 1) centered in 20-in. cubic cells of water mist, except for the vertical racks array where the cell height was increased by 4 in. The cells were closely packed except for the aisles where noted.

^aThe array reflector simulated a room with overhead water sprinklers and consisted of a 2-ft-thick concrete floor touching the bottom of the array, a 2-ft-thick concrete ceiling offset 10 ft above the floor, two 6-in.-thick full-density water walls touching the sides of the array, and two 12-ft-thick low-density water-mist walls touching the other two sides of the array. All void was filled with low-density water mist.

^bThe array reflector was similar to the reflector described in Footnote a, except that the two low-density water-mist walls were changed to 6-in.-thick full-density water walls touching the array.

ical Assemblies," LAMS-2543, Los Alamos Scientific Lab. (1961).

3. G. E. WHITESIDES and N. F. CROSS, "On the Treatment of Hydrogen Scattering in Multigroup Monte Carlo Criticality Calculations," *Trans. Am. Nucl. Soc.*, 10, 239 (1967).
4. J. T. THOMAS, Personal Communication.
5. J. T. THOMAS, "The Criticality of Cubic Arrays of Fissile Material," Y-CDC-10, p. 23, Union Carbide Corp. (1971).

Double Batching Cell Loadings in Storage Arrays, J. T. Thomas (UCC-OR)

An item often given consideration in the nuclear criticality safety evaluations for the storage of fissile materials is the contingency of "double batching" a storage cell loading or the loading in some region of a storage array. Monte Carlo calculations, using the Hansen-Roach 16-energy-group neutron cross-section sets,¹ have been made of reflected arrays of spherical units centered in cubic cells in which the mass of one or more units has been increased. The initial storage configurations considered are those recommended² by the ANS Standards Subcommittee 8 on "Fissionable Materials Outside Reactors." It is assumed that operating pro-

cedures are such as to preclude single unit criticality when the mass of a unit is doubled.

The calculated data are summarized in Table I where the initial storage arrangement is given. The succeeding columns describe the perturbation, give an estimate of the neutron multiplication factor of the perturbed region when reflected by water, locate the altered region in the storage array, and give the resultant k_{eff} of the augmented storage array. It may be observed that in almost all cases considered, the array surrounding the perturbed region is no more effective than water reflection. Further, there appears to be no significant dependence on the location of the perturbation in the array, provided the units remain centered in their respective cells. The last entry of the table is noteworthy, representing the replacement of ~184 kg U(93.2) metal in a 2 × 2 × 2 region of the array by an identical-sized 4 × 4 × 4 array having ~620 kg U(93.2) as smaller units which alone would constitute an acceptable storage configuration.

These data support the following conservative generalizations applicable to the referenced storage arrays:

1. If a double-batched cell, reflected by water, can be shown to have a k_{eff} not exceeding 0.93, such a double-batched cell in a storage array ($k_{eff} \leq 0.95$) will not result in array criticality.
2. If a double-batched unit can be shown to be subcritical when water reflected, double-batching 8 or fewer cells in a storage array ($k_{eff} \leq 0.95$) will not result in array criticality.

TABLE I
Monte Carlo Calculations of the Influence of Double-Batched Units
on Storage Array Neutron Multiplication Factors

Recommended Storage Arrangement ^a			Perturbation			Augmented Storage Array	
n_x, n_y, n_z	Spacing (cm)	Mass of Unit kg U(93.2)	n_x, n_y, n_z	Mass of Unit kg U(93.2)	Reflected Cells	Location of Perturbation in Array	Calculated k_{eff}
4,4,4	30.48	16.91	2,2,2	33.83	1.02	Center	0.964
5,5,5	30.48	10.55 •	1,1,1	21.09	0.80	Center	0.939
						Corner	0.935
			3,3,3	21.09	1.03	Center	1.021
						Corner	1.021
5,5,5	60.96	24.25	1,1,1	48.51	1.00	Center	0.981
						Corner	0.980
6,6,6	50.80	18.30	2,2,2	36.59	0.98	Center	0.968
						Corner	0.968
9,9,9	60.96	17.66	3,3,3	35.32	0.98	Center	0.976
						Corner	0.972
			5,5,5	35.32	1.05	Center	1.025
						Corner	1.025
4,4,4	50.80	23.01	4,4,4 ^b	9.63	0.93	Center	0.930

^aNeutron multiplication factor does not exceed 0.95.

^bRepresents recommended storage configuration at center spacing of 25.4 cm, i.e., cubic array of 8 cells with center spacing of 50.80 cm replaced by cubic array of 64 cells with center spacing of 25.40 cm.

Additional calculations of neutron multiplication factors of arrays by matrix methods have yielded parameters interpretable in the concepts of Avery's³ "Theory of Coupled Reactors." It can be shown, for instance, that the above conclusions are consistent with a two-component model of coupling.

1. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Crit-

ical Assemblies," LAMS-2543, Los Alamos Scientific Laboratory (1961).

2. American Nuclear Society Standards Subcommittee ANS-8.7, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials," Draft No. 5 (June 1972).
3. R. AVERY, "Theory of Coupled Reactors," P/1858, Proc. U.N. Intern. Conf. Peaceful Uses At. Energy, 2nd, Geneva, 1958.

LMFBR—Experiment and Analysis

The Seven Critical Configurations of the ZPPR Demonstration Reactor Benchmark Program, R. G. Palmer, W. G. Davey, P. I. Amundson (ANL-Idaho)

The ZPPR Demonstration Reactor Benchmark Program will have included by the time of its completion seven separate critical configurations which will provide an extremely useful series for testing ENDF/B data files and computational models. Five of these critical configurations have already been constructed and are discussed here. The ZPPR Benchmark Program has evolved from a relatively simple two-zone core (ZPPR-2) to a moderately complex one (ZPPR-3) which has a number of B₄C simulated power reactor control assemblies. ZPPR-3 is being studied in three phases; Phase 1 has 19 control rod positions (CRPs), containing sodium and stainless steel, but no B₄C, simulating the end-of-cycle configuration in a power reactor; Phase 2 has about 10% of inserted control rods (CRs), simulating the middle-of-cycle; finally, Phase 3 simulated the beginning-of-cycle with about 20% of inserted control. The loss of reactivity in each case is compensated for by the addition of spiked fuel drawers containing Pu-Al fuel plates in addition to the normal Pu-U-Mo fuel plates. A $\frac{1}{4}$ -in. U₃O₈ plate is replaced by a $\frac{1}{4}$ -in. Fe₂O₃ plate and a $\frac{1}{4}$ -in. Pu-Al plate which is an (inverse) simulation of fuel depletion in the power reactor but with no fission products involved. The core size of ZPPR-3, with one exception, remains fixed in all three phases and is very close to the ZPPR-2 core size. The latter factor is useful in allowing an overlay of the ZPPR-2/ZPPR-3 spatial measurements.

The one exception mentioned above is an intermediate configuration between ZPPR-2 and ZPPR-3, known as ZPPR-3(1A). This is one in which no fuel spiking is used but the core zones are expanded to accommodate the loss in reactivity due to 19 CRPs. Most of the experiments in 1A were subcritical control rod worth measurements which are covered in another paper¹ in these transactions. However, two additional critical configurations were built with different arrangements of six simulated control rods. These were B₄C in CRP numbers 2, 4, 6, 10, 14, and 18 and 8, 10, 12, 14, 16, 18. These CRP positions are shown in a paper given by Grasseschi,² also in these transactions.

These two critical configurations provide an interesting variation on the control rod worth experiments. Being critical, they are more clearly defined than the subcritical experiments. However, they require careful interpretation to give a check of the worth of the rods in terms of reactivity because the core sizes are different. They are clearly separate benchmark experiments.

Analysis is under way to calculate the k_{eff} of the five configurations already built using the SDX³ cross-section preparation package and ENDF/B Version 3. Preliminary calculations with a GE 6-energy-group cross-section set using ENDF/B V2 modified data in 2D X-Y diffusion theory have given the following values: ZPPR-3 (1A), 1.00181, ZPPR-3 (1A) (CRs 2, 4, 6, 10, 14, 18) 1.00168, ZPPR-3 (1A) (CRs 8, 10, 12, 14, 16, 18) 1.00134, ZPPR-3 (1B) (with spiking) 1.00170. Thus it appears there may not be extraordinary model problems introduced by the extra complexity.

1. P. I. AMUNDSON et al., "Subcritical Worth Measurements of Simulated Control-Rod Banks in the LMFBR-Demonstration-Plant Benchmark Critical," *Trans. Am. Nucl. Soc.*, 15, 931 (1972).
2. G. L. GRASSESCHI, "Calculational Support for Subcriticality Experiments in ZPPR-3, The Demonstration Reactor Benchmark," *Trans. Am. Nucl. Soc.*, 15, 937 (1972).
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Calculational Support for Subcriticality Experiments in ZPPR-3—The Demonstration Reactor Benchmark,* G. L. Grasseschi (ANL-Idaho)

A series of calculations has been performed in support of experiments with B₄C single and multiple simulated power reactor control rods in ZPPR Assembly 3, Phase 1A. The sequence of experiments was as given in Table I, with calculations set up to match these configurations utilizing the two-dimensional diffusion theory module of the Argonne Reactor Code (ARC) system DARC2D,¹ and an appropriate model.

The calculational model employed X-Y geometry with half- or quarter-core symmetry as required and a mesh structure of one interval per ZPPR drawer, giving a grid of 5.245×5.7836 cm. Control rod positions as given in Fig. 1 were assigned control rods selected from four unique designs designated Type A, B, C, D, and a sodium/steel design for positions not filled with a B₄C rod. Multigroup cross sections generated from ENDF/B Version II (modified) data by General Electric, in 29 broad

Fig. 1. ZPPR Assembly 3, numbered control rod positions.

TABLE I
Calculated and Experimental Subcriticality Parameters for
the ZPPR 3 Demonstration Reactor Benchmark

Step	Rod Design	Rod Position	Mass B_{4C} In Rods (kg)	Experimental Worth (%)	Calculated Worth (%)	Calculated $\Delta k/k$ kg^{-1} B_{4C}	C/E
1	A	1	4.580	$1.50 \pm 1\%$	1.622	0.3514	1.081
2	B	1	5.544	$1.68 \pm 1\%$	1.834	0.3282	1.091
3	C	1	13.104	$2.69 \pm 2\%$	2.921	0.2210	1.086
4	B	2	5.544	$1.50 \pm 1\%$	1.657	0.2965	1.105
5	B	2	5.544	$1.46 \pm 1\%$	--	--	--
6	C	8	13.104	$1.51 \pm 1\%$	1.776	0.1345	1.176
7	C	8, 14	26.208	$3.45 \pm 1.5\%$	3.966	0.1502	1.150
8	D	9	8.055	$1.25 \pm 1.5\%$	--	--	--
9	B	2,5	11.088	$3.08 \pm 1.5\%$	3.456	0.3091	1.122
10	B	2,4,6	16.632	$4.62 \pm 2\%$	5.100	0.3041	1.104
11	B C	2,4,6 10,14,18	35.944	$9.36 \pm 2.5\%$	10.720	0.1912	1.145
12	B C D	2,3,4,5,6,7 8,10,12,14,16,18 9,11,13,15,17,19	160.218	$26.3 \pm 10\%$	31.941	0.1991	1.214
13	C D	8,10,12,14,16,18 9,11,13,15,17,19	126.954	$17.2 \pm 6\%$	20.898	0.1643	1.215
14	C	8,10,12,14,16,18	78.624	$9.6 \pm 2.5\%$	11.063	0.1395	1.152

groups for self shielding were spatially weighted with CALHET,² and collapsed to six groups. Transverse leakage was accounted for by the addition of group- and region-dependent pseudo absorption terms, obtained from an R-Z calculation, to the macroscopic capture cross section.

The reference calculation for the series was for the critical, un-rodded reactor in X-Y geometry and in both half- and quarter-core symmetry; the obtained k_{eff} was 1.00181. A summary for 12 of the 14 experimental assemblies giving the converged ($\epsilon = 10^{-4}$) subcritical k_{eff} , calculated worth of B_{4C} , and relevant experimental values is listed in Table I.

These results show that the model overpredicts the worth of the B_{4C} rods by roughly 10% for the single, central rod cores, by 10 to 15% for multiple and off-

center configuration, and by 20% for the fully rodded assemblies. Previous experience with similar calculations, however, has indicated that transport and heterogeneity effects can contribute up to 10% to the overprediction of rod worths in critical assemblies.

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Nuclear Data for Reactor Design

Benchmark Tests and Comparisons Using ENDF/B Version-III Data, B. A. Zolotar, E. M. Bohn, K. D. Dance (ANL)

Benchmark testing of critical experiments often employs simplified calculations for reasons of economy and convenience. The usefulness of the results are then dependent on previously calculated corrections or on the assumption that the parameters being calculated are insensitive to the approximations being used. The purpose of this work is to provide detailed benchmark calculations for testing ENDF/B Version-III data as well as to examine the adequacy of the simplified calculational methods.

A series of benchmark calculations has been made for three well-documented critical assemblies: ZPR-6 Assemblies 6A and 7¹ and ZPR-3 Assembly 48.² ENDF/B Version-III data were used throughout the study and the basic 27-group cross sections were determined using the space-dependent generation capability of the SDX³ code. For each assembly, cross sections were determined by breaking the core into three subregions and the blanket into two subregions. The plate unit-cell heterogeneity options present in the SDX code were included in the calculations using equivalence theory for resonance self-shielding and integral transport methods for the spatial weighting factors.

TABLE I
Calculated and Measured Integral Results for
Three Benchmark Critical Assemblies

		Calculated Results			Experiment
		1-Dimensional Homogeneous	1-Dimensional Heterogeneous	2-Dimensional Heterogeneous	
ZPR-6 Assembly 7					
	k_{eff}	0.9772	0.9938	0.9924	1.00
	$28_f/49_f$	0.02304	0.02313	0.02316	0.02336 ± 0.00047^b
	$25_f/49_f$	1.0962	1.1143	-	1.061 ± 0.021^b
	$28_c/49_f$	0.1570	0.1530	0.1531	0.1400 ± 0.0028^b
Central Worths (Ih/kg) ^a	^{239}Pu	198.5	198.2	198.2	155.9 ± 2.0^c
	^{235}U	164.4	165.7	165.8	133.3 ± 3.0^c
	^{238}U	-11.85	-12.65	-12.63	-10.91 ± 0.46^d
	^{23}Na	-8.20	-8.38	-8.24	-6.205 ± 0.315^d
	^{10}B	-3216	-3464	-3472	-2926 ± 58^c
	Δk_{het}	0.0166			
ZPR-6 Assembly 6A					
	k_{eff}	0.9853	0.9916	0.9910	1.00
	$28_f/25_f$	0.02196	0.02187	-	-
	$28_c/25_f$	0.1434	0.1417	-	-
Central Worths (Ih/kg) ^a	^{239}Pu	62.28	63.27	62.65	57.1 ± 2.0^c
	^{235}U	47.24	48.69	48.26	41.59 ± 0.48^c
	^{238}U	-4.193	-4.395	-4.350	-3.29 ± 0.01^d
	^{23}Na	-0.272	-0.651	-0.629	$+0.160 \pm 0.040^d$
	^{10}B	-1167	-1219	-1209	-1231 ± 54^c
	Δk_{het}	0.0063			
ZPR-3 Assembly 48					
	k_{eff}	0.9744	0.9927	0.9927	1.000 ± 0.001^e
	$28_f/25_f$	0.03187	0.03135	-	0.0326 ± 0.0016^e
	$28_c/25_f$	0.1359	0.1285	-	0.137 ± 0.007^e
Central Worths (Ih/kg) ^a	^{239}Pu	566.7	559.8	553.9	445 ± 5^e
	^{235}U	430.4	427.7	-	339 ± 4^e
	^{238}U	-28.42	-31.80	-31.55	-24.8 ± 0.5^e
	^{23}Na	-10.77	-10.68	-10.51	-6.3 ± 0.3^e
	^{10}B	-9162	-10137	-10046	-8926 ± 60^e
	Δk_{het}	0.0183			

^aCalculated: 1% $\Delta k/k = 1007$ Ih for ZPR-6 Assembly 7
1% $\Delta k/k = 449$ Ih for ZPR-6 Assembly 6A
1% $\Delta k/k = 981$ Ih for ZPR-3 Assembly 48

^bG. S. Stanford and W. R. Robinson, Argonne National Laboratory, Private Communication (1972).

^cFrom Ref. 1 including small sample correction factor.

^dFrom Ref. 1.

^eFrom Ref. 2.

The principal results of this study are given in Table I. Especially evident in these results is the improvement over previous ENDF/B versions of the k_{eff} and central in-cell reaction rate ratios. The central worth discrepancy is still present, however, although the agreement with experiment shows a slight improvement. Table I also provides new $\Delta k/k$ conversion factors for the assemblies which are necessary for central worth comparisons. These values are based on the Keepin delayed-neutron data and should be taken to supersede previously quoted values.

The spherical models can be seen to give good agreement with the more detailed calculations and even the homogeneous models in most cases (with the exception of the sodium worth in Assembly 6A and the ^{10}B worth in the three assemblies) can be used to give reasonable comparisons. New values of the Δk due to heterogeneity are also provided in Table I. These values have also changed due to improvements in the heterogeneity calculations as well as because of changes in the ENDF/B data.

The basic calculations used two-dimensional diffusion theory, but heterogeneous and homogeneous spherical calculations using SDX fundamental-mode cross sections were also used to obtain comparative results. Benchmark testing usually employs homogeneous spherical models. These models for Assemblies 6A and 7 were defined in Ref. 1. The spherical model for Assembly 48 was calculated here and defined to give the same hetero-

geneous k_{eff} as the two-dimensional model. The resulting sphere radius was 45.24546 cm with a blanket thickness of 30.0 cm. Compositions and sizes used in the two-dimensional calculations are completely specified in Refs. 1 and 2.

It should also be pointed out that this work is the initial utilization of the new SDX code for benchmark testing. The ability of the code to treat heterogeneity if desired and its economy of use have proved it to be a useful calculational tool for such a study.

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Instruments and Monitors Used for In-Plant Nuclear Criticality Safety

Criticality Problems and Solutions in Nuclear Safeguards, D. B. Smith, J. E. Foley (LASL)

Nondestructive assay (NDA) instrumentation which is routinely used for nuclear safeguards and accountability also is being used to increase the number of individual containers of scrap and waste that can be safely combined into a single batch for processing. This increase is due to the smaller uncertainty with which the fissile content of each container can be determined. The result is more efficient and therefore more economic processing.

For example, a neutron coincidence counter¹ is being used at the plutonium recovery facility of the Los Alamos Scientific Laboratory for the routine assay of plutonium scrap. The plutonium content of a container is determined by coincidence counting neutrons from the spontaneous fission of ²⁴⁰Pu. Since several neutrons are emitted during each fission event and are thus correlated in time, the coincidence counting technique may be used to distinguish between the spontaneous fission rate, which is proportional to the plutonium content of the sample (if the isotopic composition is known), and the (α ,n) neutron rate, which is not necessarily proportional to the plutonium content. Incinerator solids and dissolution residues are now rapidly and more accurately assayed nondestructively than with the earlier time-consuming and expensive chemical methods. Because the plutonium content of each scrap container is measured, maximum batching within criticality limits is possible. In addition, recovery operations can be performed in less restrictive geometries. Thus, considerable savings in time and expense are realized in the recovery operation.

At one of the larger LWR fuel fabrication facilities, very large containers of low enriched uranium contaminated scrap are assayed using gamma rays (from ^{234m}Pa, a daughter of ²³⁸U) with large NaI detectors, prior to volume reduction by incineration. These assays provide a measure of the total uranium input into the incinerator. Thus incinerator cleanout is required only when the uranium content reaches the level specified by criticality procedures. This results in fewer incinerator shutdowns for cleanout.

Improved efficiency in batching for recovery of highly enriched uranium from scrap can be realized by assaying the fissile content of a container using the Random Source Interrogation System.² Here a subthreshold random neutron source is used to induce fission in the fissile content of the scrap. A fast coincidence logic system then separates the time-correlated induced fission neutrons from the random source neutrons with sensitivities equivalent to a few grams of ²³⁵U.

In all the above examples, the use of NDA instrumentation to provide a rapid and accurate measurement of the fissile content of individual containers of scrap or waste

permits maximum critically safe batching of containers for processing, and results in more efficient and economical operation of the plant facility. However, since many of the NDA systems employ a highly reflecting counting geometry, such systems themselves may pose a criticality problem. This is particularly true for the assay of unsegregated scrap for which there is little or no information on the amount or enrichment of the fissionable material in a container or the nature of the matrix material present. Thus, while a container is expected to contain a relatively small amount of fissile material, the assayer must be certain it could not inadvertently contain sufficient fissile material to present a criticality problem when placed into the counting geometry.

In such situations, if the literature reporting criticality data or calculations does not provide sufficient guidance to establish critically safe operating procedures, detailed neutronics calculations are performed to investigate the possibility that a critical system could be formed when a container of scrap of known volume is placed into the counting geometry. The results of these calculations provide the criteria by which standard operating procedures are developed.

In many cases, the gross weight of the container is a sufficient screening procedure. However, if the gross weight limit is too restrictive, additional screening procedures must be developed. For example, the use of an enrichment meter³ measurement to guard against highly enriched uranium will permit a much larger gross weight limit for containers of low enriched uranium scrap. Similarly, gross counting of the neutrons from Pu-bearing samples often can be used to establish an upper limit on the plutonium content of a container before it is placed into the assay system.

Thus, NDA instrumentation often can be used to increase the number of individual units that can be safely batched for more efficient and economical processing. Operating procedures for the critically safe assay of individual units can be established by means of careful neutronics calculations and appropriate screening procedures.

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Criticality Prevention by Continuous Non-destructive Instrumentation Assay in the Plutonium Reclamation Facility,* L. E. Bruns (ARHCO)

Criticality prevention is the number-one priority in the safe operation of the Plutonium Reclamation Facility (PRF), a plutonium scrap recovery facility operated by the Atlantic Richfield Hanford Company for the Atomic Energy Commission. The PRF uses geometrically favorable equipment wherever possible¹; however, added assurance is attained by monitoring the plutonium inputs, outputs, and inventories by continuous or semicontinuous instrumentation. Geometrically favorable equipment aided by instrumentation allows safe processing of large plutonium throughputs with attendant high plutonium inventories in the PRF.

Nondestructive-type instrumentation is used to assay each scrap package before it enters the process, and NaI single-channel analyzers are used to note inventory buildups by measurement of the 414 keV of ²³⁹Pu. Also, continuous specific gravity and nitric acid concentration instruments help assure low plutonium losses to the waste system. Both gamma and neutron counting monitors are used to measure plutonium content of waste streams and waste tanks. Figure 1 shows a simplified sketch of the PRF and its instrumentation for criticality prevention. Figure 2 shows how inventory buildup within the reflux solvent extraction system is monitored by neutron counters.²

The primary plutonium scrap package counter is a rotating-table, movable 3-in.-diam NaI detector system. Packages are periodically checked by a gamma Ge(Li) unit which defines the complete gamma spectrum as well as individual plutonium energies, by a neutron interrogation system using ²⁵²Cf as the source (measures prompt

Fig. 2. CC column Pu profile.

and delayed neutron and gamma counts), by a neutron counter, and by a calorimeter to establish credibility of the NaI system. With this type of instrumentation, accuracies of $\pm 10\%$ are possible. Several portable directional NaI gamma-counting monitors are available for noting process and hood buildup; for example, in batch dissolvers, solids in centrifuges, and hood floor buildups.

*Sponsor: C. L. Brown

Fig. 1. Plutonium Reclamation Facility instrumentation for critical mass control.

A specific gravity meter uses a pneumatic detector and the nitric acid concentration meter is an electrodeless conductivity meter. Eight-inch-long, one-inch-diameter BF_3 detectors and in-line NaI gamma monitors are used routinely as process and waste stream and tank monitors.

Future plans include an improved Ge(Li) gamma counter for packages of scrap and new and improved in-line instruments. Under consideration is a computerized plutonium accountability of the overall Plutonium Reclamation Facility.

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Monitoring for Uranium Accumulations in Diffusion Plant Equipment, R. G. Taylor (UCC-OR)

The most common cause of uranium accumulations in gaseous diffusion process equipment is a malfunction permitting moist atmospheric air inleakage to the sub-atmospheric cascade system, the moisture hydrolyzing the uranium hexafluoride process gas to form solid uranyl fluoride. Other causes include subnormal equipment temperature, resulting in condensation of solid uranium hexafluoride, and unusual mechanical failures resulting in temperatures sufficient to reduce process gas to solid uranium tetrafluoride. For nuclear criticality safety purposes, as well as for optimum production, it is necessary that these accumulations be rapidly located and recovered.

Instrumentation essential for operation warns of potential accumulation whenever abnormal operating conditions such as excessive air inleakage or subnormal temperatures are indicated. Daily input-output balance and periodic accountability inventories aid in determination of material retention in the system. Any of the above observations are considered sufficient to warrant spot-check radiation monitoring using instruments sensitive to the small gamma radiation fields accompanying accumulations. Small, slowly occurring accumulations are not readily detectable by operational instrumentation; hence, a program for routinely surveying the entire cascade with radiation detection equipment has been in effect for many years. Survey data are maintained by computer, with output showing the results for the past ten routine surveys so that any deviation from the trend may be readily detected and suspect locations promptly investigated.

Two versions of the same instrument have been used in the program, both employing the same ionization probe body, 2½-in. diam × 19 in., operated through the highly sensitive circuitry of a battery-powered alpha-particle survey meter.^{1,2}

Probes of the first version are filled with argon at 400 psig and have been shown to be about 2.5 times more sensitive to lower energy gamma radiation, typical of that from ^{235}U , than to higher energy radiations typically from ^{238}U daughter products.³ The second probe version was filled with xenon which gave an increase in the ratio of low energy-to-higher energy gamma sensitivity to about 30; however, the probe sensitivity to ^{235}U gamma radiation was so great, 0.014 mR/h full scale on the least sensitive scale, that it was essentially unusable. Re-

duction of the xenon pressure and probe input resistance reduced the sensitivity by a factor of 5, thus providing a useful accumulation survey instrument.

In operation, both instruments need frequent recalibrations. Maintenance requirements have also gradually increased. Thus, a commercial scintillation probe and associated rate meter have recently come under evaluation. The new instrument has advantages of smaller size and weight, requires only two D-size batteries for power rather than the assortment needed for the argon and xenon monitors, and is more responsive to uranium accumulations.

Gamma radiation sensitive instruments provide the means for rapid monitoring; however, only qualitative judgments about the presence of a uranium accumulation may be made. Once suspected deposits are located, an estimate of the mass is made using specially developed neutron counters which are sensitive to the low fluxes produced by the $^{19}\text{F}(\alpha, n)^{22}\text{Na}$ reaction. There are two types. The earlier model has a 2-in.-diam × 12-in. sensitive volume filled with 96% enriched $^{10}\text{BF}_3$ at 40-cm Hg pressure surrounded by 1¾ in. of paraffin to moderate the 4- to 6-MeV neutrons produced by the reaction. The counter is sensitive⁴ to a flux as low as 0.005 n/(cm² sec). The second model, a later development, employs a sensitive volume filled with ^3He at 6 atm. It has a total weight of 25 lb, half that of the original model, improved temperature tolerance, and two-thirds greater sensitivity.⁵ Estimates with these instruments of uranium-fluorine compound masses in accumulations agree favorably with recovered quantities in ranges from 0.026 lb UO_2F_2 to ~500 lb UF_6 . Conservatively, the estimated upper mass limits have consistently been greater than the mass actually recovered.

The routine radiation monitoring program requires that over 1500 process system locations be checked periodically and has proved highly successful in locating cascade uranium accumulations, particularly those that are occurring slowly. Rapid monitoring of large process areas is feasible and the estimates by neutron monitoring of contained mass provide useful nuclear safety information where equipment removal is involved.

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Criticality Safety Determination by Pulsed-Neutron Source Techniques, S. R. Bierman, E. D. Clayton (PNL)

Determining the reactivity of a reactor core, either by calculation or measurement, is relatively simple compared to establishing the reactivity of a non-reactor

Fig. 1.

system of fissile material. The nuclear reactor is well defined, uniformly arranged, and can experience only a limited number of relatively fixed conditions. The non-reactor facility, however, can generally experience a wide range of varying conditions of moderation and reflection in complex, often undefinable geometries. A reactivity for this type facility cannot be obtained from calculations without imposing overly conservative approximations. Consequently, after these systems are designed, built, and placed in operation, a simple, easily applied method is needed for monitoring their reactivity both to assure criticality prevention and, hopefully, to improve their economy. One such method, although not always applicable or easily applied, is the pulsed-neutron source technique.

In a pulsed-neutron source measurement, bursts of neutrons are repetitively injected into the system for which a knowledge of the reactivity is desired. The time behavior, between bursts, of these neutrons and their progeny is accumulated by means of a series of scalars and a suitable detector properly positioned in or near the system. Such data look, ideally, like that shown in Fig. 1. If data of this nature can be obtained on a system, the degree of subcriticality can be determined without any other knowledge of the system.

The data shown in Fig. 1 can be divided, basically, into two components. The neutron count rates just preceding a new burst at time $1/R$ are due entirely to the background plus the delayed neutrons created by the previous bursts. The neutron count rates following the burst at time zero are due to prompt neutrons created by the burst plus the background and delayed neutrons. By making a background measurement preceding the pulsed-neutron source measurement, the delayed- and prompt-neutron components in Fig. 1 can be segregated.

One method of analyzing data such as shown in Fig. 1 is that developed by Garellis and Russell.¹ They showed that the prompt-neutron count rates multiplied by a factor involving the reactivity of the system must equal the total area under the pulsed-neutron source curve between time zero and $1/R$ following the burst of neutrons.

$$\int_0^{1/R} N_P \exp\left(\frac{k\beta}{l} t\right) dt = \int_0^{1/R} N_P dt + \frac{N_D}{R}, \quad (1)$$

where

N_P = prompt-neutron count rate at time t following the burst

N_D = average delayed-neutron count rate

R = pulse repetition rate

β = effective delayed-neutron fraction

l = neutron lifetime

k = effective neutron multiplication constant.

All the information needed to solve the Garellis-Russell equation (Eq. 1) for $k\beta/l$ is contained in the pulsed-neutron data shown in Fig. 1. Then from the definition of reactivity, ρ , in dollars,

$$\frac{\rho}{\beta} = \frac{k - 1}{k\beta}, \quad (2)$$

the reactivity of the system can be determined from

$$\frac{\rho}{\beta} = -\frac{\alpha - k\beta/l}{k\beta/l}, \quad (3)$$

where

$$\alpha = \frac{1 - k(1 - \beta)}{l}, \quad (4)$$

and is the decay rate of the prompt neutrons. Since β can be estimated fairly accurately, a k value can be obtained for the system by equating Eqs. (2) and (3).

Pulsed-neutron source data, such as shown in Fig. 1, have been obtained at Hanford and analyzed as described above on a number of different systems. These measurements have been made on experimental assemblies about \$100, and less, subcritical, and on plant systems down to about \$50 subcritical. They have included experimental assemblies of PuO_2 and PuO_2 - UO_2 mixtures, low enriched uranyl nitrate, a large underground waste crib, and irradiated fuel elements. Measurements on these systems and others will be described and discussed.

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In Situ Measurements to Assure Criticality Safety During Cask Loading of BAWTR Fuel Elements, M. N. Baldwin, N. H. Shah (B&W)*

During the eight years that the Babcock & Wilcox Test Reactor (BAWTR) operated, five batches of spent fuel elements were shipped to the reprocessing plant at Savannah River. A typical batch contained 25 fuel elements (10 kg of ^{235}U), which were shipped in a lead cask. The inner region of the cask was divided by boral slabs to accommodate six rows of fuel elements.

Since calculations of this arrangement showed the fully loaded cask would be supercritical with fresh fuel, addi-

*Sponsor: F. M. Alcorn

tional boral pieces were added (Fig. 1). Based on a conservative calculation of this arrangement using the LIFE-7, THERMOS, and WANDA computer codes, the k_{eff} of the loaded cask submerged in water was about 0.85.

The casks were loaded under water. For safety, the inverse multiplication was measured and evaluated during the loading of each batch. A suitable neutron source and a pulse-type fission chamber were placed in cask positions 11 and 20. Also, one or more pulse-type neutron detectors were placed outside the cask.

Figure 2 is a typical inverse multiplication curve. Early in the loading sequence, the curve appeared to be disturbingly steep in all cases, and some interpretative allowance had to be made for the addition of elements in positions that significantly increased the lattice surface area near the detector.

Although inverse multiplication measurements are valuable in determining safe loadings, they do not accurately predict critical mass until the lattice is almost critical. The low k_{eff} of the loadings, along with the poor data expected until the lattice between the source and detector is established, leads to considerable scatter of data early in the loading. Based on the later data points,

Fig. 1. Shipping cask for BAWTR elements.

Fig. 2. Relative inverse multiplication(outside detector).

which are regarded as more reliable, widely different extrapolated values were obtained for the five almost identical loadings. However, all channels in all loadings predict a critical mass that is in excess of 50 elements.

Survey of Process Equipment for Criticality Safety on a Periodic Basis to Determine Plutonium Buildup,* M. T. Sind (ARCO)

Nondestructive instrument techniques for the detection and quantitative estimation of plutonium have been developed and used at many atomic facilities for at least 15 years. The principal application initially was for the measurement of plutonium in scrap and waste materials, and more recently has been for accountability and safeguards purposes.

This paper describes some of the methods that have been used in a processing facility to detect and measure plutonium in process vessels and lines, hoods, filters, and other equipment. Continuously operating in-line instruments are being used more and more today, but are not included in this review of nonroutine measurements.

Measurements have been made with BF₃ neutron detectors, gross gamma detectors, and energy discriminating NaI scintillation gamma detectors. These have been used in a passive method similar to that used for waste carton or container measurements. Simplified versions are usually required for easy portability. The NaI scintillation detector with a single-channel analyzer has been the most useful. However, a portable multi-channel analyzer is also often required to properly interpret the readings.

A major question raised in such measurements regards the obtainable accuracy. In a complex equipment arrangement, the geometry considerations of the setup, the variable shielding, and the background problems are often so severe that the usefulness of the readings may seem questionable. Actually, it has been found that very useful measurements can often be made. The measurements have been used in several ways. They warn of conditions that might progress in time to a criticality situation. They check on the possibility of leaks in processing equipment. They indicate when the cleanout of equipment should be done, and subsequently, how effective the cleanout has been.

A number of photographs illustrate typical measurement situations. A collimated detector, detector hoist, and cart-mounted readout were usually used. A compact, hand-held instrument was also used extensively. Some special calibration source arrangements were made to simulate some of the more common situations encountered in the measurements.

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ICPP Soluble Poison Meter, M. A. Georgeson, W. G. Morrison (Allied Chem)

The Idaho Chemical Processing Plant (ICPP) soluble poison meter was developed as an instrument to monitor continuously the amount of nuclear poison entering non-geometrically safe process vessels for dissolution of spent nuclear fuel elements. The nuclear poison (gadolinium or boron) is added to the dissolver with the dissolving acid. The meter detects the poison concentration with an accuracy of $\pm 1\%$, including errors resulting from calibration, analytical techniques, drift, and statistical

variations. The demonstrated effective range of the poison meter is 1.5 to 5 g/liter. Boron is used where possible because of its desirable nuclear properties and low cost; however, gadolinium has chemical properties preferable for a nitric acid system. The poison concentration is determined by measuring the neutron count rate from a neutron source after the neutrons pass through the poison solution. The absorption of neutrons by the solution increases with the concentration of the nuclear poison. Instrument accuracy is enhanced by using digital counting equipment from detector to alarm and for system shutdown. Reliability is provided by system conformance to single-failure criteria.

The instrument consists of a neutron source and detector with appropriate equipment for extending the system into the dissolvent acid feed tank. Neutrons are provided by a 2-Ci ^{239}Pu -Be source. The detector is a proportional counter tube. A preamplifier is located near the detector to increase the signal before transmitting to the control panel located about 250 ft away. The amplifier and discriminator supply a 10-V pulse to the count-rate meter. Visual and audio alarms are actuated when the count rate exceeds (indicating a low poison concentration) the setting of the digital comparator. The count rate is converted to an analog signal, transferred through a diode amplifier, and sent to a digital voltmeter and recorder. The diode amplifier converts the signal from a log-log function to an approximately linear output of count rate vs concentration; this approximation is within 3% over the poison concentration range. However, the recorder cannot be read more accurately. The alarm accuracy, which does not depend on human reading, is still $\pm 1\%$ of true concentration.

Causes of false concentration readings are extraneous gamma fields, air sparging of the solution being measured, acid level with respect to location of the source and detector, variation in concentration of acid, temperature of solution, and source-detector separation. The acid level must cover the tubes and the neutron source to a minimum depth of 6 in. Sparge air in the acid tank effectively changes the source-detector tube separation. The separation of source and detector tubes is optimum at ~ 3.1 cm. Lower separation does not permit thermalization of neutrons required for absorption, while a larger separation causes a degradation of statistical accuracy.

This poison-meter system, developed at the ICPP, is part of a system used to maintain nuclear criticality safety in nongeometrically safe vessels.

A Passive Gamma, Neutron Monitor for Fissile Material in Scrap and Waste,* Paul Goris (West-Hanford)

Hanford Engineering Development Laboratory (HEDL) operations include a Plutonium Fuels Laboratory for development of PuO_2 - UO_2 fast breeder-type reactor fuels, and Chemistry Laboratories that process numerous mixed-oxide fuel samples. Kilogram quantities of fissile scrap and waste generated in these operations require analysis for criticality safety and safeguards purposes. Certain chemical and physical forms of this scrap and waste including grinder sludge, contaminated glove-box equipment, dry concentrated analytical residues, plastic-mounted mixed-oxide pellets, etc., cannot be sampled and analyzed by standard chemical techniques. In many

cases, the fissile content can only be estimated in these species unless some type of nondestructive testing is used. HEDL has developed a passive gamma, neutron monitor to measure fissile content in these scrap and waste forms.¹ Features of this monitor are as follows:

1. rapid, simultaneous gamma and neutron counting analysis of Pu in small containers or waste drums up to 55 gal in size
2. applicability in the range ~ 0.01 to 130 g per container. The range may well extend to the kilogram level.
3. geometrical arrangement to test the full volume of a 55-gal drum without vertical movement of detectors
4. container rotation to minimize errors due to spatial distribution of the fissile material
5. horizontal movement of detectors to increase sensitivity, or optimize the count rate for macro levels of Pu.

Four 3- $\times$ -3-in. NaI gamma detectors and 10 BF₃ neutron proportional counters are used in this monitor. Figure 1 depicts configuration of the electronic components, the detector to waste drum geometry, and a functional block diagram of the system. A support table rotates the scrap or waste container around its vertical axis at speeds up to 7 rpm. Measurements are accomplished in 200 sec with an estimated standard error of about 0.6% in counting at the 1-g level.

The BF₃ neutron detectors are shown in test position in Fig. 1. Each bank of neutron detectors contains five proportional counters that swing into measuring position. The NaI gamma detectors are moved to their most sensitive position with the neutron detectors moved out of their measuring position. Each pair of NaI crystals is supported on roller-bearing runways. The crystals can easily be pushed from the outer position through a distance of about 30 in. to the most sensitive position adjacent to the rotating unit. In practice, the gamma and neutron detector positions are compromised so both systems can be operated simultaneously. No undue loss of detectability occurs in either system with this mode of operation.

The two banks of five neutron detectors are each contained in a 1-in.-thick polyethylene envelope that thermalizes fission spectrum neutrons for counting by the BF₃ thermal-neutron detectors. A 0.20-in.-thick cadmium sheet provides isolation for fissile material in containers to prevent induced fissioning by moderating material needed for the neutron detectors.

Detector linearity was tested with a set of PuO_2 standards ranging from 0.01 to 130 g of Pu containing 88% ($^{239}\text{Pu} + ^{240}\text{Pu}$). The PuO_2 was encapsulated in small plastic vials spaced throughout a simulated glove-box waste matrix of paper, plastic, and metal. The choice of PuO_2 rather than mixed oxide, and the spacing used provided minimum self-absorption for the purpose of evaluating system linearities. According to Cline,² the attenuation of gamma emission due to 200 g of Pu evenly distributed over an entire drum is $< 2\%$. Although the PuO_2 in the type standard prepared for these linearity tests was spaced throughout a 6- $\times$ -6-in. container, no significant self-absorption effects were noted up to the 130-g level. It was concluded that both the gamma and neutron detector systems exhibit good linearity. A detectability of about 0.01 g of Pu per container was noted for the gamma monitoring system, whereas the neutron monitoring system detectability was established at about

*Sponsor: C. L. Brown

Programs and Plans of Critical Mass Laboratories and Recent Experiments Performed for Criticality Safety—I

Criticality Research at the Oak Ridge Critical Experiments Facility, E. B. Johnson (UCC-OR)

The work of the Oak Ridge Critical Experiments Facility (CEF) was summarized four years ago by Thomas¹ at the Washington Winter Meeting of the American Nuclear Society. During the time since that presentation, several areas of uranium criticality have been probed more deeply, both experimentally and computationally. This paper examines the present status of criticality studies.

The experimental work in Oak Ridge continues to be confined to the criticality of the uranium isotopes ²³⁵U and ²³³U. In the past it has often been necessary and desirable to perform a certain number of brief experiments that were intended only to answer the question of whether a certain piece of apparatus or material or configuration was critical under the conditions of its intended use. This type of experiment has become increasingly infrequent.

Studies of basic criticality may be divided into two broad, overlapping categories: in one category are those generating data for which there is obvious and immediate application to nuclear criticality safety; in the other are those generating data fundamental to the understanding of the fission process, often with no recognized application to "practical" problems. Concurrent with both types of experimentation is a vigorous computational program through which calculational methods may be evaluated—and often validated—and the applicability of measurements extended beyond those systems amenable to experimentation.

Typical of the first category are the experiments in which moderators, insulating materials, and packaging materials were interspersed between units of U(93)² metal in three-dimensional arrays, reflected and not.³ This work was an extension of earlier measurements by Thomas² that have been the basis for the development of a calculational method of determining the criticality of arrays significantly larger than practical for experimentation.^{4,5}

Also in this category falls most of the work with uranium enriched to 5% in ²³⁵U, both as metal rods and as aqueous solution of uranyl fluoride. The criticality of single and interacting units of aqueous solutions of U(5)O₂F₂ have been reported in previous papers.⁶⁻⁹ More recently¹⁰ this material has been made critical in two types of intersecting water-reflected cylinders or pipes, one a 30-deg lateral and the other a cross. The solution concentration was varied between about 900 and 740 g U/liter. The diameter of the cylinders forming the lateral was 11 in.; that of the cross was increased from 10.5 to 11.25 in. for the experiments. Although these are

few data, they are infinitely more than had existed previously for solutions containing uranium of low ²³⁵U enrichment and have served to validate a calculational method for this material in intersecting pipes, a practical problem for chemical plants.

In like manner, the program with U(5) rods of several diameters has been extended, beyond the arrays made critical with water^{11,12} and with dilute boron solution¹³ as moderator and reflector, to arrays critical in dilute (about 300 g U/liter) aqueous uranyl fluoride solution, to submerged arrays reflected by lead on four sides, and to submerged arrays reflected by depleted uranium on one side. Some of these lattices also contained a solid neutron absorber, either stainless steel, Boral, or cadmium. The latter experiments have applicability to shipping casks for reactor fuel elements for which validated calculational methods are needed.

An example of basic or fundamental criticality physics is the work of Mihalcz¹⁴ in which β_{eff} has been derived from measurements with a sphere of uranium highly enriched in ²³⁵U.

The Criticality Data Center has supported the development of generalizations of experimental data of array criticality, resulting in the publication of five reports in its series^{1,5,15-17} and in a Proposed Guide for Nuclear Criticality Safety in the Storage of Fissile Materials. The Guide is now published by the American Nuclear Society¹⁸ on instruction from American National Standards Committee N16 for "trial use and comment for a period of one year." The proposal was developed within Subcommittee 8 of the Standards Committee of the American Nuclear Society.

The Data Center was fortunate in being permitted to publish a translation of the French document reporting their definitive critical experiments with U(93)F₆ moderated by HF in spherical reflected geometry.¹⁸ Now in process of publication is a summary, prepared at the University of Arizona, of the French CRAC experiments that studied the behavior of supercritical uranium solutions in cylinders.¹⁹ These data are important to accident analyses of solution systems.

By comparison with previous years, the Oak Ridge CEF has performed many fewer experiments than during any similar interval in its history. This is, as all of you are aware, not because "all the problems have been solved," but because of reduced staff and because the equating of safety in experiment facilities to power reactor safety results in an increased load of paper now necessary to initiate experiments. Consequently, the cost of measurements has increased appreciably and will, unfortunately, probably continue to do so.

²U(93) designates uranium enriched to 93% in ²³⁵U.

¹⁸Available from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521.

In addition to those who obtain the data, those attempting to write standards, and those who are responsible for safety and economy in industrial processes recognize there are a number of areas in criticality, even of uranium isotopes, that still need attention. The Oak Ridge Staff hopes to begin work soon in some of these areas and continue work in others for which it feels there is more than adequate justification. Among these is the exploration of critical parameters of uranium enriched to 10 to 15% in ^{235}U , for which there are no data; this program would follow essentially that almost completed with the 5%-enriched uranium. For economic and safe fuel processing, the use of neutron absorbers, such as boron in Raschig rings or gadolinium in stainless steel or an absorber dissolved in the process solution, appears quite attractive but requires experimental measurements to at least provide base points for the validation of calculational methods. Also an investigation of the criticality of uranium oxides in the forms that are currently handled can lead directly to the specification of conditions necessary for criticality safety.

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Criticality Experiments for Criticality Safety at Pacific Northwest Laboratory, E. D. Clayton, S. R. Bierman, R. C. Lloyd (PNL)

Since beginning operations in 1961, hundreds of criticality experiments have been performed at the Plutonium Critical Mass Laboratory of the Pacific Northwest Laboratory. The current research emphasis at this Laboratory is on the procurement of criticality data for plutonium-uranium mixtures that will serve as the basis for handling criticality problems subsequently encountered in the development of technology for the LMFBR community—FFTF, demonstration plants, and commercial LMFBRs.

The more recent experiments performed at the CML are listed in Table I which includes all the experiments performed on homogeneous plutonium-uranium mixtures. Included in this list are experiments on high-burnup plutonium, the effect of soluble poison (gadolinium), and the effect of boron-containing Raschig rings on criticality. Certain of the experiments are suggested for consideration as possible benchmarks for checking computational methods and nuclear data against integral critical experiments.

Figures 1 and 2 show the regions of the curves where experimental data are applicable, and indicate regions of the curves in which applicable data are being obtained in the current year. For purposes of clarity, the data with 5.4 to 7.3 wt% Pu in a narrow concentration region about 400 g oxide/liter are not indicated.

Various methods of calculation are available for use in checking ENDF/B data against integral critical experiments. The results to date have been somewhat inconsistent, particularly for the plutonium-uranium mixtures.⁷ Analyses of the experimental data are proceeding, but the insufficiency of experimental data makes it difficult, at present, to delineate sources of error in the nuclear data input. A point of concern is the difference in results on using two different processing codes. In some cases the results between ENDF/B-II and -III data differ less than those caused by selecting one processing code over another.

Data from the experiments on mixed oxides are indispensable as the basis for the preparation of standards

TABLE I
Summary of More Recent Significant Critical Experiments

Material Form	Isotopic Composition of Pu	Fuel Concentration	Critical Geometry	Number of Data Points Taken
<u>PLUTONIUM NITRATE SOLUTIONS⁽¹⁾</u>				
Plutonium Nitrate Solution (²³⁹ Pu = 41.4 wt%, ²⁴⁰ Pu = 42.9 wt%, ²⁴¹ Pu = 10.8 wt%, ²⁴² Pu = 4.7 wt%)	High Burnup Pu:	40.6 to 140 g Pu/liter	Water-Reflected Cylinder ^a	9
<u>PLUTONIUM NITRATE SOLUTIONS AND SOLUBLE POISON⁽²⁾</u>				
Plutonium Nitrate Solution Containing Soluble Gadolinium (0 to 2.4 g Gd/liter)	²⁴⁰ Pu Content in Pu: 8.38 wt%	116 g Pu/liter	Water-Reflected Cylinder ^a	6
<u>PLUTONIUM NITRATE SOLUTIONS AND RASCHIG RINGS⁽³⁾</u>				
Plutonium Nitrate Solution Containing Borated Raschig Rings	²⁴⁰ Pu Content in Pu: 8.32 wt%	63 to 412 g Pu/liter	Water-Reflected Cylinder	12
<u>PLUTONIUM-URANIUM NITRATE SOLUTIONS⁽⁴⁾</u>				
Plutonium-Uranium Nitrate Solutions (~30 wt% Pu in Depleted U - 0.6 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 4.66 wt%	71 g Pu/liter; 157 g U/liter	Water-Reflected Sphere ^a	3
Plutonium-Uranium Nitrate Solutions (~30 wt% Pu in Depleted U - 0.6 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 4.66 wt%	35 g Pu/liter; 76 g U/liter	Water-Reflected Sphere ^a	4
Plutonium-Uranium Nitrate Solutions (~15 wt% Pu in Depleted U (0.6 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 4.66 wt%	46 g Pu/liter; 265 g U/liter	Water-Reflected Sphere ^a	6
Plutonium-Uranium Nitrate Solutions (5.4 to 7.3 wt% Pu in Depleted U - 0.6 wt% ²³⁵ U) ^b	²⁴⁰ Pu Content in Pu: 22.8 wt%	22 to 31 g Pu/liter; 285 to 413 g U/liter	Water-Reflected Cylinder ^a	13
<u>HOMOGENEOUS MIXED OXIDES^(5,6)</u>				
Mixed Oxides of PuO ₂ and UO ₂ in Polystyrene (7.6 wt% Pu in Depleted U - 0.15 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 23 wt%	90 g Pu/liter; 1099 g U/liter	Plexiglas-Reflected Rectangular Parallelepipeds (Evaluation on Reflected Cube and Infinite Slab) ^a	6
Mixed Oxides of PuO ₂ and UO ₂ in Polystyrene (7.89 wt% Pu in Depleted U - 0.15 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 8.0 wt%	38 g Pu/liter; 446 g U/liter	Plexiglas-Reflected Rectangular Parallelepipeds (Evaluation on Reflected Cube and Infinite Slab) ^a	7
Mixed Oxides of PuO ₂ and UO ₂ in Polystyrene (14.62 wt% Pu in Depleted U - 0.15 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 8.0 wt%	116 g Pu/liter; 667 g U/liter	Bare and Plexiglas-Reflected Rectangular Parallelepipeds (Evaluation on Bare and Reflected Cubes and Reflected Infinite Slab) ^a	10
Mixed Oxides of PuO ₂ and UO ₂ in Polystyrene (30.0 wt% Pu in Depleted U - 0.15 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 8.0 wt%	159 g Pu/liter; 370 g U/liter	Bare and Plexiglas-Reflected Rectangular Parallelepipeds (Evaluation on Bare and Reflected Cubes and Reflected Infinite Slab) ^a	11
Mixed Oxides of PuO ₂ and UO ₂ in Polystyrene (30.0 wt% Pu in Depleted U - 0.15 wt% ²³⁵ U)	²⁴⁰ Pu Content in Pu: 11.5 wt%	1408 g Pu/liter; 3399 g U/liter	Bare and Plexiglas-Reflected Rectangular Parallelepipeds (Evaluation on Bare and Reflected Cubes and Reflected Infinite Slab) ^a	9

^aSuggested as possible benchmark for checking computational methods and nuclear data against integral critical experiments.

^b5.4 wt% Pu in U represents lower bound of Pu concentration in Pu-U fuel mixtures of which data are known; experiments on reflected cylinder constitute only measurements to date.

and codes pertaining to criticality control and safety. The current American National Standard on handling fissionable materials outside reactors contains no information directly applicable to plutonium-uranium fuel mixtures.

Criticality experiments being considered for the CML in the near-term include measurements on both heterogeneous and homogeneous assemblies with Pu-U fuel compositions typical of those from LMFBRs. Included

are further measurements to obtain data on the effect of Pu enrichment and moderation on criticality (firmly establish the curves in Figs. 1 and 2), and the effect of higher concentrations of ^{240}Pu ; also to be included are measurements on a heterogeneous assembly of mixed-oxide fuel pins (FFTF) immersed in Pu-U nitrate solutions (dissolver problem). Data are available from previous experiments on the criticality of these fuel pins in water.^{8,9} The effectiveness of fixed poisons

Fig. 1. PNL data relevant to evaluation of reflected critical volumes for $^{239}\text{PuO}_2 + \text{U}(\text{Nat})\text{O}_2\text{-H}_2\text{O}$ mixtures typical of mixed-oxide fuels.

Fig. 2. PNL data relevant to evaluation of reflected critical masses for $^{239}\text{PuO}_2 + \text{U}(\text{Nat})\text{O}_2\text{-H}_2\text{O}$ mixtures typical of mixed-oxide fuels.

(boron-containing Raschig rings), soluble poisons (boron-containing Raschig rings), and soluble poisons such as gadolinium and boron (or mixtures of these) on the criticality of Pu-U mixtures, is of considerable interest and is to be studied.

The planned experimental program, together with problem areas in nuclear criticality safety of Pu-U mixtures, will be discussed.

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Experimental Programs at the Dow Rocky Flats Nuclear Safety Laboratory, C. L. Schuske (Dow-Colo)

The Dow Nuclear Safety Facility has the primary goal of supporting the Dow-AEC production complex at Rocky Flats. This is accomplished by performing pertinent experiments to provide checks for our computational methods. These computational methods are then used to design chemical and metallurgical processing equipment and fissile storage facilities. However, some of the experiments have direct application to specific plant problems.

The lab began operations in September of 1965. Since that time, we have performed approximately 1000 criticals and critical approaches with plutonium and enriched uranium. A large number of measurements were performed with close-fitting hemispherical assemblies. These assemblies were reflected with oil and steel and oil combinations.¹⁻³ A description of the apparatus in which these experiments were done is given. These same hemispherical components were also used in a study in which they were immersed in tanks containing fissile solution. These experiments were performed to simulate

the dissolution of metal in an acid solution.^{4,5} The assembly area and a typical assembly and fissile solution vessel are described. These measurements were followed by a series in which outer shells of boron stainless steel (~1.2 wt% boron) were placed around the fissile metal, thus somewhat decoupling the metal from the aqueous fissile solution. The experiments were performed to check computations and determine if economies could be realized if inner dissolution baskets of boron stainless could be developed, thus increasing plutonium metal batch sizes for dissolution.

Another series of measurements involved arrays of intersecting pipes or cylinders.⁷ We referred to this series of assemblies as "trees," because there were as many as 16 arms or branches, 4 arms per layer, intersecting a central column. The central column had a square cross section ~100 in. tall and the branches were cylinders of diameters ranging from 4.5 to 6.5 in. Each of these branches was 54 in. long. This work was correlated into a simple engineering model and presented and later extended by Dickinson and Schuske to cover a much wider range of applications.⁸ The information gained in this work was used to design a tree tank which would hold ~1000 liters of solution.⁹ The tree tank is described.

The criticality of arrays of parallel cylinders sitting upon a fissile solution slab were investigated.^{10,11} These experiments were performed under minimum reflector conditions as well as various reflector thicknesses of Plexiglas. The number of cylinders in arrays varied from 1, 4, 9, and 16. Cylinder diameters ranged from 4.5 to 8 in.

Since all our chemical processing is done in hoods or other enclosures, these experiments simulate solution leaks which could result in slabs of solution accumulated beneath process equipment.

The last experimental series discussed consists of the use of boron stainless plates to poison a large volume of fissile solution.¹² The plates contain ~1.02 wt% normal boron. The experimental setup is described. The general vessel dimensions are 59 in. tall x ~42-in. i.d. Since the boron steel plates are 47 in. long, this permitted an unpoisoned solution slab to be developed above the plates. The solution concentrations used are 50 g U/liter, 140 g U/liter, and 450 g U/liter. The work on these systems continues. One interesting point observed was that for a 400-g U/liter $\text{UO}_2(\text{NO}_3)_2$ solution, 28 vol% boron steel in the container lowered k sufficiently that the effect of this poisoned tank of solution upon the unpoisoned slab in contact at the top was about the same as a thick water reflector.

A series of measurements involving 3-kg cylinders of high-density plutonium fully water reflected and moderated is being planned.¹³ The arrays will contain upwards of 200 kg of Pu metal.

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Critical Mass Laboratory Program in France, P. R. Lécorché (CEA-Saclay)

Experimental nuclear criticality safety studies started in Saclay in 1960 with ALECTO experiments using nitrate solutions of ^{239}Pu , ^{235}U , and ^{233}U (Ref. 1).

Since 1963 they have been conducted mainly at the VALDUC Critical Mass Laboratory. Plutonium nitrate solutions were used in various geometries: slabs, cylinders, and annular cylinders. The effects of acid molarity and ^{240}Pu content were studied; ^{240}Pu content varied from 1.5 up to 10 wt% (Refs. 2 and 3). Some experiments were made with a small amount of plutonium containing 37 w% ^{240}Pu .

The criticality of uranium at low H/U atomic ratios (5.7 to 82) has been investigated by mixing liquid $\text{U}(\text{O}_2)_2$ and HF in water-reflected spheres.⁴ These experiments showed the important effect of scattering materials on critical masses at low H/U ratios.⁵

Some interaction studies have been made using 294-mm-i.d. cylinders containing either uranyl or plutonium nitrate solutions and 250-mm-i.d. cylinders with plutonium nitrate solutions.⁶ Neutronic isolating properties of different materials have been studied in the course of these experiments.

It has been shown, for example, that lead, while being a good reflector, has very poor isolating properties. The insertion of a slab of lead even 20 cm thick between two interacting cylinders decreases the critical height.

Experiments on heterogeneous poisoning of plutonium nitrate solutions by borosilicate glass pipes or Rashig rings have been reported.⁷

More recent studies have been related to the fast reactor and light-water reactor fuel cycles.

A series of experiments has been made using variable pitch lattices of Rapsodie fuel pins either in water, in nitric acid, or in plutonium nitrate solution. They showed or confirmed that:

1. The same lattice is more reactive in water than in nitric acid.
2. The same lattice is more reactive in water than in concentrated plutonium nitrate solution provided the pitch is less than or equal to the optimum pitch in water.

The criticality of 4 w% enriched UO_2F_2 aqueous solution has been investigated in three geometries:

1. a stainless-steel 500-mm-i.d. cylinder (wall thickness 3 mm) either bare or fully reflected, or bottom and side reflected
2. a stainless-steel 792-mm-i.d. cylinder (wall thickness 4 mm) bottom and sides reflected
3. a Monel 540-mm-i.d. sphere (wall thickness 4 mm) fully reflected.

Uranium concentration was varied from 0.548 up to 1.084 g/cm³. Main results are:

1. 500-mm-i.d. cylinder:
 - minimum critical mass -
 - fully reflected: 79 kg - U concentration: 1.010 g/cm³
 - no top reflector: 84 kg
 - minimum critical concentration -
 - 0.630 g/cm³
2. 792-mm-i.d. cylinder:
 - minimum critical mass -
 - no top reflector: 136 kg - U concentration: 0.800 g/cm³
 - minimum critical concentration -
 - 0.450 g/cm³
3. 540-mm-i.d. sphere:
 - critical concentration -
 - fully reflected: 0.953 g/cm³ corresponding to a 77-kg critical mass.

In additional experiments, the 500-mm-i.d. cylinder was surrounded by a 150-mm-thick lead reflector (no top or bottom reflector). At 0.800 g/cm³ concentration, the critical mass was the same as with a full water reflector.

Planned experiments are:

1. tests on poisoning effects of iron
2. studies on 80% fissile (^{239}Pu + ^{241}Pu) plutonium nitrate solutions either in annular cylinders or in a parallelepiped vessel having a square base 1.300 × 1.300 mm. Experiments with low-concentration solutions in this last vessel are intended to determine more precisely the influence of finite dimensions in big vessels. Temperature and acid molarity effects shall also be investigated.
3. studies of lattices of fast reactor fuel rods in water, nitric acid, and fissile solutions.

As for the future, it seems that experimental nuclear criticality safety studies with Pu , ^{235}U , and ^{233}U should be limited to specific experiments related to peculiar uses or processes. Experiments with fissile actinides should

be planned when these isotopes will be produced in sufficient amount.

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Programs and Plans of Critical Mass Laboratories and Recent Experiments Performed for Criticality Safety—II

Experimental Criticality in the United Kingdom,* J. H. Chalmers (Nucl Installations Inspectorate-England)

At the present time, there are no experimental criticality facilities operating in the UK; nor, indeed, are any programs of critical mass measurements currently planned. It is not clear how much the organizational changes that have taken place over the last two years have contributed to this state of affairs. Certainly, prior to the reorganization of the UKAEA, criticality experimentation had been restricted for some time to those areas where there was a need for confirmation of computational methods. There is no reason to believe that the approach to nuclear safety problems will not continue to be predominantly calculational, but new computer codes have been developed which are not yet fully tested, and there is an obvious need to reconsider the situation and determine the direction of future effort.

The last experiments conducted investigated the critical parameters of mixtures of plutonium and uranium for 30 wt% Pu, over the range H/Pu 18.6 to 1461.

The measurements at H/Pu = 18.6 reported by Lane¹ were made with oxide/polythene compacts, having a plutonium concentration of 0.950 g/cm³, Pu/U atomic ratio of 0.335, and a ²⁴⁰Pu content of 5.9%.

The derived polythene-reflected critical cube has a 27.7-cm side and contains 20.2 kg of plutonium. Two interesting features were observed in reflector replacement measurements:

1. The effect of a polythene reflector of less than 6-cm thickness was relatively less than is the case for plutonium or uranium cores, the difference being greatest at 2.5 cm.
2. A cadmium sheet, 0.05 cm thick, placed between core and reflector, has an effect equivalent to reducing the reflector thickness from 20 to <2 cm.

Solution measurements were made by Lane and Parker² with mixtures of Pu(NO₃)₄ and U₃Na₂O₂(NO₃)₂ at H/Pu atomic ratios of 224, 795, 1372, and 1461. The ²⁴⁰Pu content was 5.63%; other properties are given in Table I.

The stainless-steel core vessels were 25.4-, 30.6-, 38.0- and 50.7-cm diam. The wall thickness for the smallest vessel was 0.2 cm, and for the remainder, 0.25 cm.

Steel cylindrical shells, capable of being clamped to any vessel, were available to enable estimates to be made of the effect of vessel walls on critical size.

In the assembly, the appropriate core vessel was situated within a water reflector vessel of 84-cm diam. This vessel was built up from aluminum-alloy ring sections,

*Sponsor: C. L. Brown.

TABLE I
Analytical Data

Solution	A	B	C	D
H/Pu (atoms)	224	795	1372	1461
Pu/(Pu + U) %	30.7	30.7	30.6	30.6
Density (g/cm ³)	1.524	1.184	1.1185	1.113
Conc (g/liter) Pu	101.3	31.58	18.61	17.50
U	228.5	71.3	42.2	39.6
NO ₃	319	132	97.5	94.7
H ₂ O	857	949	962	963
Normality	1.60	1.10	0.95	0.93
N/Pu (atoms)	12.1	16.1	20.2	20.9
Atoms/cm ³ × 10 ²⁰				
²³⁹ Pu	2.396	0.747	0.44	0.414
²⁴⁰ Pu	0.144	0.045	0.026	0.025
²⁴¹ Pu	0.011	0.0033	0.002	0.0018
²³⁵ U	0.042	0.012	0.0076	0.0076
²³⁸ U	5.739	1.791	1.060	0.995
H	573	633	643	644
N	31.0	12.8	9.5	9.2
O	386	362	349	349

TABLE II
Critical Parameters

Solution	Diameter (cm)	Height (cm)	Volume (liters)	Plutonium Mass (g)
A	25.425	56.31	28.59	2896
	30.62	29.89	22.01	2230
	37.99	21.17	24.00	2431
	50.72	16.05	32.43	3285
B	30.62	46.18	34.01	1074
	30.62	46.29	34.10	1077
	37.99	28.24	32.01	1011
	50.72	20.39	41.21	1301
	50.72	20.35	41.11	1298
C	37.99	72.86	82.61	1537
	50.72	33.59	67.86	1263
D	50.72	37.16	75.08	1314

bolted to a stainless-steel base section which provided a 15-cm water reflector below the base of the core vessel. Top reflection was provided by a 15-cm-thick polythene plug, suspended inside the core vessel by two independent chains. Results are given in Table II.

Additional measurements were made using a mockup of the dump valve and a water-filled top reflector plug to determine the reflector saving difference between the experimental conditions and 15-cm water reflection.

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Critical Lattice Data in Support of Safe Fuel Handling Criteria, P. G. Lacey (W-NES)

This paper discusses critical experiment lattice data used in support of developing safe fuel handling criteria outside reactors for both uranium- and plutonium-bearing nuclear fuel. From these and other supporting data, applicable criteria for safe handling of powders, pellets, fuel rods, and fuel assemblies were established.

Standard reactor design codes and buckling-to-shape conversion formulae have been used to determine minimum critical dimensions for optimally moderated and fully reflected units, including spherical, planar, and cylindrical geometries. From these minimum critical mass values and dimensions, safe mass limits and dimensions for isolated units were determined by using generally accepted safety factors taken from the literature as discussed in Ref. 1. Through the use of appropriate critical experiment lattice configurations it was possible not only to verify minimum critical dimensions for several configurations, but to make comparisons between the calculated limiting-safe parameters and the actual configurations required to produce criticality.

Inherent in these calculations of critical dimensions is the use of a reflector savings fit applicable for water-reflected slightly enriched uranium- and plutonium-bearing systems over a wide range of moderating ratios. A fit to experimentally determined reflector savings data for a range of enrichments up to and including 5.7 wt% ^{235}U in UO_2 and 6.6 wt% PuO_2 in $\text{PuO}_2\text{-UO}_2$ lattices² was

Fig. 1. Effective extrapolation distance.

determined as shown in Fig. 1. In addition, data from uranium rod lattices³ enriched to 3.0 wt% ^{235}U were also compared. Use of this reflector savings fit,

$$\lambda = 4.62 \exp(-0.0096 H/X) + 6.12 \text{ cm}$$

with the computed material bucklings over a range of H/X (hydrogen to fissile atom ratios), leads to satisfactory and often conservative minimum critical dimensions as shown in Fig. 2.

Coupled core critical experiment lattice data were used to verify the accuracy of standard reactor design codes used in determining safe separation distances for fuel assembly handling and storage.

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Fig. 2. Critical number of rods for 6.6 wt% $\text{PuO}_2\text{-U}^{235}$.

Summary of Measurements Needed for Nuclear Criticality Safety—An Industrial Point of View, Ricardo Artigas (GE-San Jose)

A survey¹ of a cross section of the nuclear community was made in an effort to identify criticality measurements that are still needed or will be needed by the commercial nuclear industry for nuclear criticality safety evaluations. In this paper, the term "commercial nuclear industry" applies primarily to that segment of industry involved in the manufacture and reprocessing of fuel for nuclear power reactors.

Experimental data obtained from critical or subcritical systems are the bases for determining the accuracy and, in a broad sense, establishing the range of applicability of the methods used in designing and operating critically safe plants. Although we are in an era of sophisticated and accurate computational methods, they are not always readily applicable to particular industrial plant situations. In such instances, the criticality safety specialist is forced into making simplifying and always conservative assumptions and into using conservative semiempirical methods such as solid angle, density analog, or surface density in solving complex plant problems. Criticality data also are used directly as the bases for design.

The commercial nuclear industry, then, needs additional criticality data primarily to better identify the conservatism inherent in the critically safe design, operation, and spacing of equipment and items containing fissile material in industrial plants, with possible beneficial effect on the cost of building and operating such plants.

Criticality measurements are needed to provide for experimental verification of calculated data points and for supplementing the criticality data already available for mixed-oxide systems with plutonium concentrations and enrichments in the range of interest to light-water reactor applications and for low enriched uranium systems. These needs, as identified by the survey, are summarized below.

Criticality data are needed for:

1. Lattices of mixed-oxide rods with PuO_2 concentrations ($\text{PuO}_2/\text{PuO}_2 + \text{UO}_2$) in the range of 1 to 10 wt%, and plutonium with ^{240}Pu contents ($^{240}\text{Pu}/\text{Pu}$) in the range of 5 to 20 wt%, with pellet diameters ranging from 0.3 to 0.60 in.

2. Homogeneous mixed-oxide systems. The greatest need exists for PuO_2 concentrations in the range of 1 to 10 wt% with plutonium having ^{240}Pu contents ranging between 5 and 20 wt%.

3. Mixed-oxide rods undergoing dissolution in nitrate solutions containing fissile material or soluble poisons such as $\text{Gd}(\text{NO}_3)_3$ and $\text{Cd}(\text{NO}_3)_2$. The PuO_2 concentration of the mixed-oxide ranging between 1 and 5 wt% and the ^{240}Pu content of the plutonium ranging between 10 and 28 wt%.

4. Determining the effect of fixed poisons (such as borosilicate Raschig rings) in the critical plutonium concentration of nitrate solutions of mixed oxides.

5. Determining the minimum plutonium concentration needed to achieve criticality in a mixed-oxide ($^{239}\text{PuO}_2 - \text{UNatO}_2$) system.

6. Determining the minimum hydrogen content required to achieve criticality in a mixed-oxide system with $^{239}\text{PuO}_2$ concentrations in the range of 5 to 10 wt%.

7. Precise definition of the minimum critical ^{235}U enrichment for unmoderated UO_2 systems. Criticality data are also needed for systems of low-enriched uranium with small amounts of hydrogenous materials added.

8. Specific compounds of low enrichments such as ADU and $([\text{NH}_4]_4 \text{UO}_2[\text{CO}_3]_3)$.

9. Random distribution of low enriched uranium oxide pellets and pellet chips in water.

10. Large reflected arrays consisting of very squat-shaped units, and for arrays composed of low enriched uranium units (not exceeding 5%), with particular emphasis on low-density interunit moderation and reflection effects.

TABLE I
Summary of CRAC Experimental Studies

Experiment No.	01	08	09	12	15	17	18	26	37	38	43	59
Vessel diameter, mm	300	300	300	300	300	300	300	300	800	800	800	300
U_t concentration, g/liter	52.0	202	78.3	77.9	82.3	79.6	79.6	166	21.8	21.4	202	221
Critical volume, liter	130	18.9	29.5	29.9	28.3	28.8	28.8	19.7	222	231	70.9	17.9
Flow rate, liter/h	1,440	1,620	1,590	100	1,490	500	100	1,520	450	1,320	1,410	1,290
Final volume, liter	251.5	22.6	37.8	42.9	44.5	42.5	69.7	27.9	256.9	259	76.4	22.2
Duration of experiment, min	260	10	5	20	5	6	360	3	7	4	3	3
Peak fission rate in the first pulse	1.1×10^{18}	3.0×10^{18}	2.9×10^{17}	1.0×10^{18}	1.1×10^{17}	2.0×10^{18}	7.7×10^{15}	2.6×10^{18}	8.6×10^{18}	2.8×10^{17}	5.6×10^{18}	2.8×10^{18}
Number of fissions in the first pulse	2.7×10^{17}	8.0×10^{18}	4.4×10^{18}	3.7×10^{18}	3.7×10^{18}	3.7×10^{18}	3.7×10^{18}	3.9×10^{18}	3.4×10^{17}	3.2×10^{17}	1.3×10^{17}	3.4×10^{18}
Doubling time first pulse, sec	2.9	0.0007	0.015	0.275	0.033	0.177	0.52	0.0027	0.345	0.102	0.0050	0.0022
Total number of fissions	5.0×10^{17}	2.7×10^{17}	2.9×10^{17}	4.8×10^{17}	5.5×10^{17}	4.9×10^{17}	5.0×10^{17}	3.0×10^{17}	3.2×10^{18}	2.7×10^{18}	1.05×10^{18}	2.7×10^{17}
Final temperature, °C	34	96	76	103	102	97	-	102	110	105	104	96
Total potential reactivity (calculated) 10^{-3}	850	5,100	3,100	4,300	5,450	4,650	7,500	7,760	2,160	1,760	5,210	5,300

11. Evaluating the effect of unmoderated fissile reflectors on the critical parameters of Pu- and/or ^{235}U -bearing systems.

12. Evaluating the effect on critical systems of thick slabs or rods of poisons, such as cadmium, boron, or boron carbide as used in irradiated fuel casks.

It is well understood that some of the foregoing needs of industry, especially those concerned with low enriched and undermoderated uranium compounds, may be extremely difficult to satisfy because of practical limitations. It is hoped that in those cases the realistic limits of experimentation can be investigated so that more meaningful extrapolations become possible.

1. Private Communications from F. M. Alcorn (Babcock & Wilcox), H. K. Clark (E. I. Dupont de Nemours & Co.), D. M. Dawson (General Electric Co.), D. L. Dunaway (National Lead Company of Ohio), W. S. Geiger (Westinghouse Electric Co.), T. Gutman (Combustion Engineering Inc.), R. E. Kropp (Gulf United Nuclear Fuels Corp.), R. Nilson (Jersey Nuclear Company, Inc.), R. L. Stevenson (USAEC), E. E. Walker (General Electric Co.), and F. G. Welfare (Babcock and Wilcox).

CRAC Experimental Study of Criticality Accidents, P. R. Lécorché (CEA-Saclay)

First results of CRAC (experimental study of criticality accidents), have been reported previously^{1,2}; the program was terminated in December 1972.

The solution used was 93 wt% enriched uranyl nitrate. Three series of experiments have been completed in a 30-cm-diam cylinder:

1. 400 cm high using low concentration solutions
2. 100 to 200 cm high with open or closed top (vented through a 5-cm-i.d. pipe) using medium and highly concentrated solutions
3. 50 cm high with closed top (vented through a 5-cm-i.d. pipe) using highly concentrated solutions.

In the fourth series, an 80-cm-diam cylinder was used which permitted study of criticality accidents in large volumes of low concentration solutions or in slab geom-

etry with concentrated solution. In some experiments, concentrated solution was pumped either from the top or through the bottom into the vessel already containing a large volume of nonfissile solution, creating an initial heterogeneous system. Criticality occurred by natural or forced mixing of the two phases. In one case, the system was built up in such a manner that the layer of fissile solution reflected by the nonfissile solution was critical, but, after the accident the mixed solution was subcritical. Tables I and II give some results obtained during the most typical experiments.

Some practical conclusions may be drawn from this experimental study:

1. In all homogeneous cases, when no boiling occurs, the ratio

$$R = N_f / (V_f \times \Delta k_p) ,$$

where

N_f = total number of fissions

V_f = final volume of solution

Δk_p = calculated introduced reactivity

is a constant for a given diameter.

For the 30-cm-diam cylinder: $R \approx 2.3 \times 10^{14}$ fissions/cm³
For the 80-cm-diam cylinder: $R \approx 5.7 \times 10^{14}$ fissions/cm³.

2. The maximum number of fissions produced in the first 2 min was 4.1×10^{17} for the 30-cm cylinder and 2×10^{18} for the 80-cm cylinder.

3. The ratio of dose-to-number of fissions depends on the dimensions of the bursting solution and on the location of the point where the dose is measured relative to the source. For example, at the same location it is smaller for the 80-cm cylinder than for the 30-cm cylinder. In the case of the 80-cm vessel with concentrated solution forming a slab, at the same distance from the solution, it is higher on the vertical axis than on the side.

4. Mechanical effects have generally been trivial, although vibrations have been experienced and solution has sometimes been expelled. However, in the case of experiment No. 44 with a heterogeneous system, the insertion of reactivity by mixing a large quantity of uranium solution with nonfissile solution was rapid; a steel support fixed across the upper edge of the vessel was bent by the

TABLE II
Heterogeneous Systems

Experiment No.	44	46	47
Vessel diameter, mm	800	800	800
Nonfissile solution volume, liter	200	120	96
Concentrated solution added	through the bottom	from the top	through the bottom
Concentrated solution added volume, liter	46	21.2	146
U_t concentration, g/liter	200	211	31.6
U_t mass, kg	9.20	4.47	4.60
Peak fission rate in the first pulse	3.1×10^{19}	2.6×10^{17}	3.5×10^{17}
Number of fissions in the first pulse	1.9×10^{17}	1.4×10^{17}	1.7×10^{17}
Doubling time first pulse, sec	0.0014	0.061	0.055
Total number of fissions	1.7×10^{18}	2.0×10^{18}	3.4×10^{17}
Remarks	Part of the solution expelled and material damages	-	Subcritical after mixing of the two phases

expelled solution, the vessel supports were heavily warped, and the vessel itself was slightly distorted.

1. P. R. LÉCORCHÉ and H. F. REVOL, "CRAC—Experimental Study of Criticality Accidents," *Trans. Am. Nucl. Soc.*, 14, 33 (1971).

2. R. AGAISSE, F. BARBRY, J. C. BOULY, P. LÉCORCHÉ, D. MANGIN, L. MAUBERT, and H. REVOL, "Le Programme C.R.A.C.—Etudes d'Accidents de Criticité en Solution, Commissariat à l'Energie Atomique," *Bulletin d'Informations Scientifiques et Techniques* n° 172 (July-August 1972).

Fuel Processing, Spent-Fuel Transportation, Storage and Waste Disposal

Licensing of Packages for the Transport of Radioactive Materials, C. E. MacDonald, R. H. Odegarden (USAEC), *Invited Paper*

Both national and international regulations pertaining to safety in the transport of radioactive materials rely principally on the integrity of the packaging and the proper preparation of the packages for shipment. Accordingly, it is the policy of the Regulatory organization of the AEC to make an independent review of all container designs submitted by licensees, to assure that the containers meet package standards contained in AEC regulation 10 CFR Part 71. The AEC and DOT recently signed a new Memorandum of Understanding. According to this memorandum, the AEC will continue to regulate the handling and shipping of radioactive material, while the evaluation and approval of certain packages will be transferred to the AEC.

On receipt of an application pursuant to Part 71, the Transportation Branch, Directorate of Licensing, performs a preliminary review to determine whether the application provides a reasonably complete presentation of the information required by Part 71 which forms the basis for the findings that must be made to support the issuance of a license. If the application is not complete, further review of the application will not be conducted until the deficiencies have been rectified.

Guides are being prepared to define the format and contents of applications for licenses authorizing the use of packaging for transport of radioactive materials. Their objective is to inform applicants as to the type, detail, and format of the information to be provided in support of applications so that good quality applications are received. As an integral part of the licensing process, the Transportation Branch performs an in-depth technical review of the design and use of packages for shipment of radioactive materials.

Because of the tremendous growth and development of the nuclear industry, the economic pressures to minimize shipping costs are causing licensees to more efficiently utilize shipping packages. This, coupled with the changing nature of the materials to be transported (e.g., fuels irradiated to higher burnups and recycled Pu) and greatly increased transport needs, is causing designers to match actual operating conditions as closely as possible in the design.

We are alert to this trend and are refining our analytical techniques for package evaluation.

With the assistance of consultants in some cases, the Branch develops calculational methods (including complex computer programs) for analyzing the structural, thermal, shielding, and criticality aspects of package designs. The Branch staff uses these methods to perform independent analyses of package designs to verify the adequacy of the applicant's analyses and results.

Currently the Branch is using the ORNL Isotope Generation and Depletion Code¹ to determine source terms for our shielding analyses. Shielding analyses are being

performed using ANISN, DOT, or MORSE programs depending on the modeling requirements of the problem. A 22-18 coupled-neutron and gamma-ray cross-section library² is used in the shielding calculations. Criticality analyses are performed almost exclusively with the KENO Monte Carlo program although ANISN and DOT may also be used. The cross-section sets used in criticality calculations are the 16-group Knight-modified (ORNL) Hansen-Roach for homogeneous systems and the 123-group GAM. The latter set is resonance-corrected, cell-weighted, and reduced to 16 groups by the XSDRN program for heterogeneous fuel regions.

Heat transfer analyses are performed using the HEATING4³ program and structural analyses can be performed using the STRUDL and NASTRAN program.

It is planned in the near future to revise the present general requirements for quality assurance (QA) in design, fabrication, use, and maintenance of shipping containers in Part 71 to include more specific definition of the QA program and acceptance tests. In addition to incorporating the criteria of Appendix B of Part 50, requirements for leak-tightness, structural integrity, shielding, and thermal tests will be specified. Currently the Branch is requiring QA for irradiated fuel casks.

When a new package has been determined to be acceptable for licensing, a determination is made whether the package may be used under the general license provisions of Paragraph 71.9(b) of 10 CFR 71. In such cases a license is issued to a single licensee. Additional users of the package must then fulfill the following requirements:

1. Have a copy of the specific license authorizing use of the package and all documents referred to in the license or application.
2. Comply with the terms and conditions of the specific license and the applicable requirements of 10 CFR Part 71.
3. Submit in writing to the Director of Licensing prior to the first use of the package, his name and license number, the name and license number of the holder of the specific license authorizing use of the package, and the model number of the packaging.

In the near future the Branch is planning to certify certain container designs rather than license each package. This will further simplify the administrative procedures prior to using the package for the first time.

1. M. J. BELL, "ORIGEN-ORNL Isotope Generation and Depletion Code," ORNL-4628, ORNL Chemical Technology Division (1973).
2. G. W. MORRISON et al., *Trans. Am. Nucl. Soc.*, 15, 535 (1972).
3. W. D. TURNER and M. SIMAN-TOV, "HEATING4-An IBM 360 Heat Conduction Program," ORNL-TM-3208, ORNL Mathematics Division (1971).

Analysis of Critical Assemblies

The Calculation of ZPR-6-6A and 6-7 with Karlsruhe Data and Methods, W. J. Oosterkamp (KFZ-Karlsruhe)

Cross-section sets and methods have been developed and used for the analysis of various SNEAK assemblies at Karlsruhe.¹ It seemed worthwhile to analyze some foreign assemblies on the same basis to obtain an independent check. The ZPR-6 assemblies 6A and 7 were chosen. They represent the inner zone of the demonstration type of fast reactors and have been well described.² The results of the calculations are presented in Table I.

Two-dimensional diffusion calculations were made in R-Z geometry. Heterogeneity calculations to obtain a correction in k_{eff} were performed with the ZERA code.

For the reaction rate fine structure in ZPR-6-7, the cell code REAC was used (Fig. 1). REAC is a collision probability code using the probability table concept. Care was taken to represent the actual thickness of the activation foils in the calculation since the capture rate of ^{238}U , especially, is strongly influenced.

Fig. 1. Reaction rate fine structure in ZPR-6-7.

Agreement is reasonable, with the exception of the ^{238}U capture rate in the center of the uranium oxide plates. The perturbation phase of REAC was used to calculate the reactivity of material worth samples. We represented the experiments in one-dimensional slab geometry. We assumed that the structure of the sample holder is equivalent to 2-mm stainless steel. This has an effect of 10% on the worth of ^{238}U compared to a homogeneous case. The normalization integral was taken from a 2-D diffusion calculation.

The agreement is of the order of 10%. The large normalization integral discrepancies as reported in Ref. 2 are not evident. The use of the β evaluation of Evans et al.³ will reduce the ZPR-6-7 values by 5% and the ZPR-6-6A values by 2%.

1. R. BÖHME et al., "Progress of Fast Reactor Physics in the Federal Republic of Germany," KFK-1632, Karlsruhe (1972).
2. Applied Physics Division Annual Report 70-71, ANL 7910, Argonne National Laboratory.
3. A. E. EVANS, M. M. THORPE, and M. S. KRICK, "Revised LASL Delayed Neutron Yield Data," *Nucl. Sci. Eng.*, 50, 80 (1973).

TABLE I

Results of the Calculations with the KFKINR Set

	ZPR-6-7		ZPR-6-6A	
	Exp.	Cal. Exp.	Exp.	Cal. Exp.
$k_{2D-Diff.}$		0.992		0.996
k_{HET}		0.997		0.993
σ_{f8}/σ_{f9}	0.0234	1.03		
σ_{f5}/σ_{f9}	1.065	1.03		
σ_{c8}/σ_{f9}	0.143	1.05		
σ_{f8}/σ_{f5}			0.0229	0.97
σ_{c8}/σ_{f5}			0.134	1.08
ρ^{10B}	-2900	1.12		1.04
ρ^{235U}	131	1.14	41.8	1.17
ρ^{238U} in sample holder	-9.8	1.09	-3.29	(0.73)
ρ^{239Pu}	159	1.16	57.0	1.12
$\rho^{235U} - \text{Fuel}$			48.0	0.94
$\rho^{239Pu} - \text{Fuel}$	169	1.11		
$\rho^{238U} - \text{Fuel}$			-3.5	0.98
$\rho^{238U} - \text{U}_2\text{O}_8$	-8.5	1.04		

Experimental reactivities in lh/kg

ZPR-6-7 1% EQ 958 lh
ZPR-6-6A 1% EQ 469 lh

Analysis of High ^{240}Pu Zoned Critical Experiments in Support of LWR-Grade Pu Utilization in FTR, S. Ramchandran, R. W. Rathbun (WARD)

The first two FTR cores are to be fueled with a low ^{240}Pu content fuel, commonly called "clean" Pu; whereas, the follow-on cores will progressively contain high ^{240}Pu content fuel.¹ Because the present FTR/Engineering Mockup Critical (EMC) experiments were performed² using clean Pu, the impact of using high ^{240}Pu content fuel on the neutronics characteristics and performance has to be evaluated by analyzing other critical experiments.^{3,4}

The currently available high ^{240}Pu zoned experiments are in blanketed systems such as ZPR-3/Ass'y 48B (Ref. 3) and ZPR-6/Ass'y 7 (Ref. 4). These experiments were analyzed using the FTR design methods and the reference FTR Set 300 cross sections. Heterogeneously resonance self-shielded, flux-weighted, and cell-averaged cross sections were used in all the analyses. The diffusion theory calculations were performed either in X-Y or R-Z geometry depending on the nature of the experiment. Both the reference and the corresponding high ^{240}Pu zoned assemblies were analyzed using identical techniques and the integral parameters were compared on a one-to-one basis.

The results of analysis including the k_{eff} bias, Δk (heterogeneity), fission ratios, reaction rate traverses, Na void reactivity effects, and the worth of fuel and B_4C drawer at the core center are shown in Table I. Of particular interest is the ^{238}U fission ratios. The C/E values in the high ^{240}Pu zoned assemblies are lower by 2 to 4%

compared to the reference assemblies. This suggests that the high-energy flux spectrum which causes the threshold fission reaction in ^{238}U does not have the desired degree of hardness. However, the $\text{B}_4\text{C}/\text{Na}$ -SS exchange worth of the central drawer suggests that the low-energy flux spectrum in the ^{240}Pu zone is predicted correctly on a comparative basis. All other calculated parameters shown in Table I suggest that the present calculational methods using the reference FTR set 300 cross sections are adequate with the one exception mentioned above.

1. S. RAMCHANDRAN et al., "Fuel Pin Enrichment Specifications for FTR Cores 1 and 2," WARD-2171-26, Westinghouse Advanced Reactors Division (January 1973).
2. A. TRAVELLI et al., "The ANL-FFTF Critical Experiments Program: A General Evaluation Based on ENDF/B-I Data," *Trans. Am. Nucl. Soc.*, 14, 811 (1971).
3. A. M. BROOMFIELD et al., "ZPR-3 Assemblies 48, 48A and 48B," ANL-7759, Argonne National Laboratory (December 1970).
4. Reactor Development Program Progress Report, ANL-7872, Argonne National Laboratory (October 1971).

TABLE I
Comparisons of Calculation with Experiment for Different Integral Parameters

PARAMETER	ZPR-3						ZPR-6					
	Ass'y 48			Ass'y 48 B			Reference			High ^{240}Pu Zoned		
	C	E	C/E or C-E	C	E	C/E or C-E	C	E	C/E or C-E	C	E	C/E or C-E
1. Eigenvalue bias ^a	0.9923	1.0000	-0.0077	0.9897	1.0000	-0.0103	0.9939	1.0000	-0.0061	0.9946	1.0000	-0.0054
2. Δk (heterogeneity)	0.0166	---	---	0.0171	---	---	0.0174	---	---	0.0177	---	---
3. Fission ratios (relative to ^{239}Pu)												
a. ^{240}Pu	0.238	0.244	0.957	0.233	0.238	0.977	0.183	0.185	0.991	0.190	0.193	0.984
b. ^{235}U	1.049	1.025	1.023	1.052	1.037	1.014	1.100	1.047	1.050	1.090	1.031	1.057
c. ^{238}U	0.032	0.032	1.025	0.031	0.031	1.008	0.022	0.022	1.000	0.023	0.024	0.958
4. Na-void reactivity ^b effect-core center	5.62	7.90	-2.28	7.63	8.40	-0.77	3.51	2.64	0.87	9.73	6.24	3.49
5. Central drawer worth ^c												
a. fuel/Na-SS	---	---	---	---	---	---	0.129	0.105	1.23	0.157	0.131	1.20
b. $\text{B}_4\text{C}/\text{Na}$ -SS	---	---	---	---	---	---	0.491	-0.408	1.20	-0.525	-0.435	1.21
6. ^{239}Pu (n,f) radial traverse, peak-to- average	---	---	---	---	---	---	1.691	1.697	0.966	1.490	1.459	1.021

Analysis of the SNEAK Assemblies 9B and 9A-O, P. Fehsenfeld, G. Jourdan, M. Pinter, A. M. Raberain, W. Scholtyssek (KFZ-Karlsruhe)

SNEAK-9B is the physics mockup of the German prototype fast reactor SNR. The outer zone is composed of uranium as we do not have enough plutonium. SNEAK-9A is the uranium equivalent in which extensive control rod studies will be performed.

The results of experiments and calculations are given in Table I. k_{eff} has been calculated with R-Z diffusion theory. A heterogeneity correction has been applied. Cell calculations with normal and 10^{-3} reduced thickness of the plates were performed with the ZERA code² and cell-averaged cross sections were generated. With these, two-dimensional diffusion calculations were made. The difference was then assumed to be the heterogeneity correction.

Reaction rate fine structure was measured by activation foils of 0.1-mm thickness for the metal foils and 0.25-mm thickness for the oxide foils. In Fig. 1, the results of 9B are compared with calculations with the cell code REAC. The effects are small, but the agreement between theory and experiment is poor. The cell average reaction ratios are calculated fairly well; an exception is the ^{238}U fission rate in SNEAK-9A-0, which is underpredicted by 7%.

Fig. 1. Reaction rate fine structure in SNEAK-9B.

Material worth measurements were performed in both assemblies with a vertical oscillator that minimized the lattice perturbation.³ The samples were inserted in 9A-0 between 20% ^{235}U plates and a Na_2CO_3 plate and in 9B between a $\text{PuO}_2\text{-UO}_2$ and a Na plate. Heterogeneity effects were calculated for ^{235}U and ^{239}Pu of 1%, for ^{238}U of 10%, and for B_4C of 5%.

The values for ^{240}Pu and ^{241}Pu were obtained from pure samples.

The results might indicate a bias in the normalization integral. The 20% discrepancy between the calculated over experimental ratios of the fissile isotopes and B_4C is too high to be explained by cross-section errors.

1. E. KIEFHABER, "The KFKINR set of Group Constants," KFK-1572, Karlsruhe (1972).
2. D. WINTZER, "Heterogeneity Calculations including Space Dependent Resonance Self-Shielding," *Fast Reactor Physics*, II, 237, IAEA, Vienna (1968).
3. W. J. OOSTERKAMP, "The Measurement and Calculation of the Reactivity Worth of Samples in a Fast Heterogeneous Zero Power Reactor," KFK-1036, Karlsruhe (1969).

TABLE I
Results of the calculations with the KFKINR set

	SNEAK-9B		SNEAK-9A-0	
	Exp.	Cal. Exp.	Exp.	Cal. Exp.
$k_{2D}\text{-Diff}$		1.002		0.999
k_{HET}		1.002		1.005
σ_{f8}/σ_{f9}	0.0310 ± 0.086	0.98		
σ_{c8}/σ_{f9}	0.1380 ± 0.002	0.99		
σ_{f8}/σ_{f5}			0.0278 ± 0.0006	0.93
σ_{c8}/σ_{f5}			0.1380 ± 0.002	0.97
$\rho^{235}\text{U}$	0.490 ± 0.01	1.13	0.2430 ± 0.004	1.11
$\rho^{238}\text{U}$	0.030 ± 0.0006	1.00	0.0193 ± 0.0002	1.05
$\rho^{239}\text{Pu}$	0.661 ± 0.015	1.14	0.340 ± 0.01	1.10
$\rho^{240}\text{Pu}$	0.132 ± 0.013	1.07	0.050 ± 0.01	1.20
$\rho^{241}\text{Pu}$	1.00 ± 0.05	1.01	0.460 ± 0.02	1.14
$\rho^{10}\text{B}$	9.90 ± 0.1	0.996	8.49 ± 0.15	0.94
$\rho \text{B}_4\text{C}$	1.54 ± 0.02	0.94	1.22 ± 0.02	0.94

Experimental reactivities in β/kg

SNEAK-9B $\beta = 0.00428$

SNEAK-9A-0 $\beta = 0.00698$

Multi-Cell Heterogeneity in Critical Assemblies, W. P. Barthold,* G. J. Calamai, T. F. Heenan (WARD)**

The analysis of critical assemblies is based on the unit cell concept which assumes that the cell under consideration is embedded in an infinite array of identical cells. Considering the Engineering Mockup Critical (EMC) experiment¹ with its two small core zones containing four types of fuel drawers (A, B, C, D) and 18 gross heterogeneities, we see that the assumption the unit cell concept is based on is not fulfilled at all. The longest chain of repetition in the inner EMC core is five cells. To take into account the effect of the environment as well as the flux gradient on the fuel cell under consideration, 14 basic cell sequences were analyzed² (Table I) which represent about 98% of the fueled drawers. S_0 calculations were performed in 26 groups.^{3,4} They showed that the flux distribution inside the fuel drawer depends very strongly on the environment. Flux advantage factors are shown in Table II (see next page). For the inner core zone, they vary for group 1 between 1.13 and 1.24 and for the outer core zone between 1.11 and 1.17.

To estimate the approximate magnitude of the different flux fine structure corrections, 1-D diffusion calculations were performed for a three-zone FTR (inner and outer core zone, reflector). The advantage factors obtained from each S_0 calculation were used to correct the 26-group microscopic cross sections which were then collapsed to 18 groups over the inner and outer core zone spectrum, respectively. The eigenvalues assigned to sequence 1, 2, ..., were obtained by using for the inner core zone the cross sections based on sequence 1, 2, ..., and for the outer core zone the cross sections based on sequence 7. The eigenvalues assigned to sequence 7, 8, ..., were obtained by using outer core cross sections based on sequences 7, 8, ..., and inner core cross sections based on sequence 1. Depending on the cell sequence the eigenvalue calculations are based on, they can differ by more than 2.5%. Combining the inner and outer core cross sections by volume-flux weighting, the eigenvalues differ by <0.1% when compared with the standard unit cell treatment (sequences 1 and 7).

This represents a remarkable cancelation of diverse effects. Whenever local effects are being studied (off-center Doppler reactivities, power mappings, reaction rate traverses) one has to expect that the multicell heterogeneity and the errors introduced by using the standard unit cell treatment play a very important role.

1. S. RAMCHANDRAN and G. H. MADDEN, "Analysis of ZPR-9/Engineering Mockup Critical (EMC) Experiment in Support of FTR Fuel Enrichment Specifications," *Trans. Am. Nucl. Soc.*, 15, 509 (1972).
2. W. P. BARTHOLD, G. J. CALAMAI, T. F. HEENAN, and S. A. SCHELLIN, "Preanalysis of the FTR Engineering Mockup Critical," FFTF Project FRP-411 (April 5, 1971).
3. S. RAMCHANDRAN, A. J. KOUELKA, and G. H. MADDEN, "Comparison of Modified ENDF/B-2 and ABN Cross Sections in the Analysis of FTR Critical Experiment," *Trans. Am. Nucl. Soc.*, 14, 812 (1971).
4. S. RAMCHANDRAN and G. H. MADDEN, "Analysis of FTR-3 Critical Experiments in Support of FTR Nuclear Design," WARD-2171-19, Westinghouse Advanced Reactor Division (April 1971).

TABLE I.
EMC Cell Sequences

Sequence

1	A	B				
2	IC	A	B	IC	OC	REF
3	IC	A	B	Na	OC	REF
4	IC	Na	A	B	OC	REF
5	IC	B	A	Na	OC	REF
6	IC	Na	B	A	OC	REF
7	C	D				
8		IC		C	D	OC REF
9		IC		OC	C	D REF
10		IC		OC	C	D B ₄ C REF
11		IC		OC	D	C REF
12		IC		OC	Na	D C REF
13	C	D	Na	OC		
14	OC	Na	C	D		

IC - inner core homogenized
OC - outer core homogenized
REF - reflector
B₄C - B₄C drawer homogenized

Critical Experiments on Cluster-Type Fuel Lattices of ATR in Japan, H. Sakata, Y. Hachiya, K. Shiba, N. Fukumura, A. Nishi (PRNFDC-Japan)

The advanced thermal reactor ATR now under development in Japan is highly heterogeneous in core structure: a cluster-type fuel is contained in a pressure tube with coolant, which is separated from D₂O moderator by a calandria tube. In such a system, neutron behavior is complicated and then very sensitive to the slight change of structure or material of the core. Therefore, a large amount of experimental data on both microscopic¹ and macroscopic point of view must be accumulated especially for the lattices close to that of the ATR. Then, accuracies of calculational codes which are used for the core design should be confirmed by using those data in reference to the lattices of the ATR.

In the present experiments, critical D₂O moderator levels (B_m^2) and lattice parameters (δ^{25} , ρ^{28} , and δ^{28}) have been measured on both sides (22.5 and 25.0 cm) of the 24.0-cm pitch of the ATR, and their dependences of coolant void ratio and fuel enrichment were made clear.

TABLE II (Paper 7)

Diffusion Calculation Eigenvalues and ^{239}Pu Advantage Factors

Sequence	1	2	3	4	5	6	7	8	9	10	11	12	13	14
1	1.1734	1.1602	1.2352	1.1827	1.1339	1.2062	1.1072	1.1130	1.1320	1.1258	1.140	1.1744	1.1300	1.1253
2	1.1608	1.1488	1.2220	1.1692	1.1212	1.1901	1.0984	1.1039	1.1236	1.1173	1.0942	1.1660	1.1201	1.1151
3	1.1191	1.1104	1.1758	1.1265	1.0800	1.1433	1.0681	1.0742	1.0918	1.0861	1.0639	1.1263	1.0870	1.0815
4	1.0846	1.0792	1.1352	1.0928	1.0480	1.0990	1.0337	1.0510	1.0642	1.0610	1.0381	1.0818	1.0591	1.0546
5	1.0650	1.0622	1.1142	1.0715	1.0256	1.0747	1.0447	1.0420	1.0506	1.0525	1.0268	1.0436	1.0487	1.0436
6	1.0257	1.0297	1.0668	1.0359	0.9878	1.0225	1.0142	1.0193	1.0252	1.0295	1.0028	1.0079	1.0215	1.0171
7	1.0080	1.0160	1.0443	1.0246	0.9733	0.9947	1.0036	1.0080	1.0123	1.0166	0.99193	0.99260	1.0073	1.0044
8	1.0016	1.0111	1.0366	1.0194	0.9661	0.9843	0.99795	1.0023	1.0060	1.0096	0.98652	0.98545	1.0006	0.99801
9	0.99808	1.0097	1.0311	1.0192	0.9650	0.9776	0.99620	1.0006	1.0039	1.0071	0.98477	0.98638	0.99808	0.99622
10	0.98969	1.0027	1.0186	1.0163	0.9624	0.9656	0.98983	0.99429	0.99631	1.0008	0.98085	0.08163	0.99091	0.99036
11	0.98507	0.99848	1.0099	1.0137	0.9626	0.9596	0.98593	0.99016	0.99070	0.99707	0.98030	0.97362	0.98648	0.98704
12	0.99011	1.0002	1.0156	1.0063	0.9579	0.9675	0.98337	0.98770	0.98580	0.99495	0.98285	0.96836	0.98460	0.98450
13	0.97416	0.9859	0.99945	0.99277	0.9471	0.9540	0.97130	0.97574	0.97085	0.98298	0.97259	0.94483	0.97303	0.97287
k	1.1464	1.1476	1.1569	1.1507	1.1356	1.1433	1.1629	1.1640	1.1653	1.1658	1.1604	1.1624	1.1644	1.1640
weighting factor	0.2529	0.0656	0.2098	0.1038	0.2641	0.1038	0.3681	0.2061	0.0873	0.0425	0.1092	0.0317	0.1175	0.0376
k combined			1.1460								1.1634

Fig. 1. One-quarter cross-section of a cluster lattice.

A cross-sectional view of a fuel cluster and the surrounding pressure and calandria tubes is shown in Fig. 1. Fuel meat length of the cluster is 200 cm, and the D₂O moderator level can be changed from 80 to 300 cm in the Al tank which is 300 cm in diameter and 350 cm in height.

The coolants used in the experiment were H₂O (0 %), air (100 %), and two kinds of mixture solution: D₂O(36.82)-H₂O(63.17)-H₃BO₃(0.0092 wt%) as 30%, and D₂O(81.91)-H₂O(18.07)-H₃BO₃(0.0215 wt%) as 70% void coolant. Small amount of boric acid additionally dissolved in the mixture solution is to simulate Σ_a of coolant with actual void in thermal-energy region.

The material buckling, B_m^2 , was obtained from axial and radial neutron flux distributions by Cu wire activation measurements in the case of 1.2% enriched UO₂ fuel lattices. For the 0.7 and 1.5% UO₂ fuel lattices which were made critical by being surrounded with the 1.2% fuel lattice region, B_m^2 was measured by the progressive substitution method.² Fine distributions of thermal-neutron flux were also measured by Dy foil activation.

Cell values of lattice parameters are obtained from the following definitions:

$$\delta_{\text{cell}}^{25} = \frac{\Sigma_{\text{cell epi-Cd}}^{235}\text{U fission rate}}{\Sigma_{\text{cell sub-Cd}}^{235}\text{U fission rate}}$$

$$\rho_{\text{cell}}^{28} = \frac{\Sigma_{\text{cell epi-Cd}}^{238}\text{U capture rate}}{\Sigma_{\text{cell sub-Cd}}^{238}\text{U capture rate}}$$

$$\delta_{\text{cell}}^{28} = \frac{\Sigma_{\text{cell}}^{238}\text{U fission rate}}{\Sigma_{\text{cell}}^{235}\text{U fission rate}}$$

TABLE I
Results of the Material Buckling and Lattice Parameters

Lattice Pitch 22.5 cm									
Fuel Enrichment	Coolant Void Ratio	B_m^2 ($\times 10^{-4} \text{ cm}^{-2}$)		$\delta_{\text{cell}}^{25}$		ρ_{cell}^{28}		$\delta_{\text{cell}}^{28}$	
		Exp.	Cal.	Exp.	Cal. ^a	Exp.	Cal. ^a	Exp.	Cal.
1.2% UO ₂	0%	11.6	11.4	0.0585	0.0537	0.823	0.803	0.0475	0.0493
	30%	11.2	11.2	0.0594	0.0617	0.914	0.902	0.0459	0.0502
	70%	10.3	10.2	0.0798	0.0778	1.17	1.08	0.0458	0.0515
	100%	8.8	9.60	0.0957	0.0924	1.24	1.12	0.0559	0.0591
1.5% UO ₂	0%	16.1	16.1	0.0713	0.0639	1.09	0.967	0.0462	0.0485
	100%	11.2	12.3	0.104	0.103	1.45	1.36	0.0533	0.0583
Lattice Pitch 25.0 cm									
0.7% (natural) UO ₂	0%	0.70	-0.10	0.0366	0.0310	0.419	0.454	---	---
	100%	3.94	3.60	0.0518	0.0477	0.560	0.567	---	---
1.2% UO ₂	0%	10.3	10.4	0.0430	0.0470	0.727	0.699	0.0429	0.0474
	30%	11.1	10.6	0.0535	0.0526	0.852	0.770	0.0465	0.0481
	70%	10.9	10.5	0.0627	0.0633	0.926	0.886	0.0475	0.0493
	100%	9.84	10.7	0.0664	0.0703	0.936	0.867	0.0568	0.0556
1.5% UO ₂	0%	14.6	14.5	0.0520	0.0557	0.925	0.845	---	---
	100%	13.1	13.5	0.0826	0.0834	1.11	1.05	---	---

^aCorrected for the difference of the thermal cutoff energy.

Experimental error: $B_m^2 \sim \pm 2\%$ except 0.7% UO₂ ($\pm 10\%$), $\delta^{25} \sim \pm 4\%$, $\rho^{28} \sim \pm 5\%$ in 22.5 cm P.L. and $\pm 3\%$ in 25.0 cm P.L., $\delta^{28} \sim \pm 5\%$.

To obtain ρ^{28} and δ^{25} , Cd-ratio of Np γ -ray activity and that of fission product γ -ray activity were measured, respectively, by irradiating depleted (378 ppm ^{235}U , 0.05 mm) and enriched (90% ^{235}U , 0.1 mm) uranium foils with and without a Cd filter and UO_2 buttons (2.0 mm). For the δ^{28} , ratio of ^{238}U -to- ^{235}U fission γ -ray activities was measured by irradiating enriched and depleted uranium foils.

The B_m^2 , δ^{25} , ρ^{28} , and δ^{28} measured are shown in Table I as a function of lattice pitch, coolant void ratio, and fuel enrichment, as well as the calculational results of METHUSELAH-II code.³

Increase of void ratio means decrease of H_2O coolant in the cluster fuel lattice. The slowing down effect by hydrogen decreases values of δ^{25} , ρ^{28} , and critical mass. This tendency is strongly observed in 22.5-cm pitch lattice but weak in the 25.0 cm one. The maximum value of B_m^2 is seen at 30% void ratio. It is concluded that in the 25.0-cm pitch lattice, contribution of slowing down effect by H_2O coolant is surpassed by its thermal-neutron absorption effect.

Values of B_m^2 , δ^{25} , and ρ^{28} increase with fuel enrichment. Large reduction of thermal-neutron flux in fuel, then the increase of thermal-neutron utilization factor, is

conceivable. Since fast-neutron fission event is tightly connected with thermal fission event, it is considered that the dependence of δ^{28} on fuel enrichment is small. It is seen in the values of δ^{28} measured for the 22.5-cm pitch lattice of 0 and 100% void ratio.

The calculated values of B_m^2 , δ^{25} , ρ^{28} , and δ^{28} are in good agreement with experimental values as seen in Table I. However, some discrepancy is also seen for the values of B_m^2 of 100% void ratio lattices.

The above experiments made in the deuterium critical assembly DCA of the PNC have provided useful design information for the ATR in Japan and also made it possible to confirm the calculational accuracy of the design code METHUSELAH-II.

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2. K. SHIBA et al., "Measurements of Material Buckling by the Progressive Substitution Method," *Trans. J. At. Energy Soc.*, A-16, Fall Meeting (1971).
3. M. J. BRINKWORTH et al., "METHUSELAH II—A Fortran Programme and Nuclear Data Library for the Physics Assessment of Liquid Moderated Reactors," AEEW-R480 (1966).

Calculation of Reactor Parameters

Criticality Analyses for the FFTF Interim Decay Storage Facility, W. L. Bunch, D. R. Marr (West-Hanford)

The Interim Decay Storage (IDS) facility provides storage for fresh or irradiated fuel for the Fast Test Reactor (FTR). A series of one- and two-dimensional transport and diffusion theory calculations was made to establish the effective multiplication factor of the IDS facility when fully loaded with initial driver fuel for the FTR, and also to establish the upper plutonium content of fuel that can be stored in the facility. No consideration was given in these analyses to accidents in which sodium coolant would be lost or fuel spacing changed.

The IDS facility consists of a sodium-filled vessel ~12.5 ft in diameter and 17 ft deep. A ball-bearing-supported, gear-driven, stainless-steel rack rotates within the sodium and contains 112 storage positions arranged in four concentric annular rings, giving an effective fuel spacing of about 10 in. (The FTR contains about 75 fuel assemblies in a 4.715-in. hexagonal lattice arrangement.) The rack rotates beneath ports in the structural cover to provide access to each storage position. Fresh fuel is introduced into sodium in the IDS vessel prior to entering the reactor vessel. Irradiated fuel from storage positions within the reactor vessel is transferred to the IDS vessel to decay until subsequent examination or processing takes place.

The effective multiplication factor for a full loading of 112 initial FTR driver fuel assemblies (26.4 wt% Pu) was calculated to be 0.73 ± 0.04 . The isotopic composition of the plutonium in the driver fuel is 0.864, 0.117, 0.017, and 0.002 for the isotopes 239, 240, 241, and 242, respectively. Based on this isotopic composition, it was calculated that mixed-oxide fuel containing up to 35 wt% Pu could be stored in the IDS facility with the effective multiplication factor remaining below 0.9 at the 95% confidence level. In all cases, the diluent was natural uranium oxide.

To establish the applicability of the nuclear data used in this study, two-dimensional diffusion theory calculations were made for the experimental critical configuration employed in the ZPPR/FTR shield experiment.¹ In addition, diffusion theory and discrete ordinates calculations were made to establish the applicability of diffusion theory for this composition. Although the IDS array is best described in three dimensions, a series of calculations was made to demonstrate that homogenization of regions could be employed to reduce the analysis to two dimensions. An R- θ model was employed to provide relatively accurate detail on the geometric relationship of individual fuel assemblies. This case was subsequently collapsed into annular fuel rings, and then into a single annular fuel zone. The homogenized condition was calculated to be slightly more reactive than the discrete assembly model. The effect of homogenization on axial leakage was determined by comparing an R-Z annular ring model with a homogeneous model. These calculations indicated that homogenization reduced the effective multiplication of the system. Results of the two-dimensional calculations were adjusted on the basis of these results to remove the bias introduced by the limitations in the model. Calculations were also made to verify the adequacy of the mesh structure and the angular quadrature employed in the calculations.

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Data and Analysis for Nuclear Criticality Safety

Critical Experiments to Measure the Neutron Poisoning Effects of Cu and Cu-Cd Plates, S. R. Bierman, R. C. Lloyd, E. D. Clayton (PNL)

Shipping casks, storage facilities, and process vessels involved in handling fissile materials contain neutron-absorbing materials—either for their neutron absorbing properties or for structural reasons. These types of systems can be fully utilized, safely, only if the neutron absorbers are properly accounted for in the criticality analyses. The reactivity worth of two such materials, copper and copper containing 1 wt% cadmium, was measured recently at the Battelle-Northwest-operated Critical Mass Laboratory. The results of these measurements are presented in this paper and provide a set of clean, well-defined, poisoned critical assemblies that can be used to check calculational techniques and cross-section data.

Two different fuels were utilized in these measurements to provide data on the relative worth of the Cu and the Cu-Cd in systems of differing neutron energy spectrum. Both sets of fuel were homogeneous mixtures of PuO_2 - UO_2 -polystyrene having Pu concentrations typical of the LMFBR fuels. One set had an atomic H:Pu+U ratio of 30 and contained 15 wt% PuO_2 in the oxide. The second set had an atomic H:Pu+U ratio of 3 and contained 30 wt% PuO_2 in the oxide. (A complete description of each fuel can be found in Refs. 1 and 2.) The critical assemblies consisted of rectangular parallelepipeds, fully reflected with 6 in. of Plexiglas, and containing Cu or Cu-Cd plates up to 1 in. in thickness. Each plate was positioned horizontally in its respective assembly and had essentially the same cross-section dimensions as the assembly.

The relative effect for single plates of either Cu, Cu-Cd, or Al, in thicknesses up to 1 in., on the critical height of the 30 H:Pu+U fuel, is shown in Fig. 1. In each

Fig. 1.

Fig. 2.

case shown in the figure, the Cu, Cu-Cd, or Al plates were located in the fuel parallel to and 4.01 in. from the bottom reflector.

Data for the 3 H:Pu+U fuel are shown in Fig. 2. In these assemblies the poison plates were located in the fuel parallel to and 6.01 in. from the bottom reflector. As would be expected, since copper and cadmium are basically thermal-neutron absorbers, their effect on the critical size of the 30 H:Pu+U fuel is much greater than on the faster 3 H:Pu+U fuel. In the more thermal fuel, a $\frac{1}{8}$ -in.-thick plate of Cu-Cd increased the critical height by about 20%; this same plate caused only a 4% increase in the critical height of the 3 H:Pu+U fueled assembly having the same cross-section dimensions.

Additional measurements were made to determine the relative effects that plate position and multiple plates have on the critical size. These will be reported together with calculational checks against the integral critical experiments.

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2. S. R. BIERMAN and E. D. CLAYTON, "Critical Experiments With Mixed Oxides Containing 8 and 30 wt% Plutonium," *Trans. Am. Nucl. Soc.*, 15, 307 (1972).

Array Critical Experiments with Materials Used in Shipping Containers, D. W. Magnuson (UCC-OR)

The Oak Ridge Critical Experiments Facility has had a continuing program in support of nuclear safety in the handling, storage, and shipping of subcritical units of fissile material. The present experiments are extensions of earlier air-spaced, uranium-metal array studies and consist of 8-unit, $2 \times 2 \times 2$ near-cubic arrays. These data

provide reference criticality information for materials used in shipping containers.

Cubic units of plywood, Celotex, Foamglas, and borated plastic foam were fabricated in two pieces, each having a cylindrical void region so 21-kg cylinders of U(93) metal could be placed at the center of each unit. Cuboids of plastic-bonded vermiculite having a thickness varying from 1 to 2 in. were cut to form boxes to enclose each U-metal cylinder resulting in about 10% voids for the experimental cell. Adjustments in the outer box dimensions were made to achieve criticality for $2 \times 2 \times 2$ arrays closely surrounded by a 6-in.-thick polyethylene reflector. Concrete-reflected, air-spaced arrays were also assembled to delayed criticality. The thickness of part of the reflector was varied from 8 to 16 in.

Multiplication factors for 28 critical experiments were calculated for the detailed geometry of each experiment using the KENO Monte Carlo code¹ with Hansen-Roach 16-group cross sections.² Although some experiments were small perturbations of other experiments, in no case was the disagreement, $\Delta k = k(\text{exp}) - k(\text{calc})$, greater than 0.02. We consider the calculational method used for comparison to these experiments to have been validated within this limit on Δk . Since these validations were done on near-cubic geometries, calculated spacings for cubic arrays are used for comparison of materials and are given in Table I. In these calculations, the material fills all of the volume between the units and the effectively infinite reflector was close fitting.

The Foamglas, borated plastic foam, and vermiculite have negative reactivity effects; the others are positive when compared to the air-spaced paraffin-reflected array. The neutron absorption in the boron contained in Foamglas and borated plastic foam was expected to have a negative effect. Moderation effects of hydrogen in wood and Celotex were expected to be positive. The reactivity effect of the vermiculite was negative. If the hydrogen content would be increased in the vermiculite, the reactivity effect would become positive. The concrete reflector, when compared to paraffin, increased the reactivity.

TABLE I

Calculated Critical Spacings for 8-Unit Cubic Arrays

Separating Material	Reflector	Cubic Cell Size or Critical Spacing (cm)	Δk^a
Air	Paraffin	23.2 ^b	
Air	Concrete	27.7	0.066
Wood	Polyethylene	26.5	0.052
Celotex	Polyethylene	24.5	0.032
Foamglas	Polyethylene	18.3	-0.114
Borated plastic foam	Polyethylene	17.2	-0.148
Vermiculite	Polyethylene	22.8	-0.007

^aThe k is estimated from the change in spacing of the array from the air-spaced array

$$k = \left[\left(\frac{a_2}{a_1} \right)^2 \left(1 - \frac{m}{m_0} \right) + \frac{m}{m_0} \right]^{1/3}$$

where $m = 20.96$ kg, $m_0 = 56.39$ kg, and $2a_2 = 23.2$ cm and the change in k for the reflector or material change is $\Delta k = -(k - 1)$. See Ref. 3.

^bSee Ref. 4.

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3. J. T. THOMAS, "Generic Array Criticality," Y-CDC-13, Union Carbide Corporation, Oak Ridge Y-12 Plant (1973).
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Light-Water-Moderated Reactivity of Tubular Slug Pairs Containing 1.1 wt% Enriched Uranium, James R. Smotrel (duPont-SRL)

Experiments in the exponential facility at the Savannah River Laboratory have provided a basis for safe storage and handling of slightly enriched tubular uranium slugs. The reactivities of light-water-moderated and -reflected arrays of clustered fuel slugs enriched to 1.1 wt% ²³⁵U were measured as a function of pitch, both square and triangular. This work represents an expansion of nuclear safety data presently available by providing new data on larger and concentric tubular assemblies. These data can be used to extend experimental correlations of calculational codes.

The test assemblies consisted of a stack of five uranium slug concentric pairs, each 11.114 in. long (28.2 cm) and supported by a central aluminum rod (see Table I for dimensions). Spacers of solid acrylic rods (three per annulus, equally spaced) were used to keep the slugs aligned and yet preserve the properties of the light-water environment.

The measurements were made in the Subcritical Experiment,¹ a 5-ft-diam cylindrical tank. The assemblies were pinned to a solid steel plate at the bottom of the tank and laced together with rigid spacers at the top to provide good alignment and spacing. The open tank was filled with water to the top of the array to provide reflection on all sides, but not to the top surface, which was exposed to air.

The light-water reflector necessitated the determination of an extrapolation distance for each of the arrays. This distance was taken to be that value of λ which, in conjunction with the measured κ^2 , gave the same material buckling for each array size of the same pitch (Table II). The measured k_{eff} 's were then obtained by using this derived λ , the measured κ^2 , and a calculated M^2 . A 3.0-cm extrapolation distance was assumed for the air boundary at the top of the array.

TABLE I

Nuclear Safety Assembly

Component	i.d.-o.d.-in.	Uranium Conc., wt%		
		²³⁵ U	²³⁶ U	²³⁸ U
Aluminum rod	0 - 0.735			
Fuel slug	2.018 - 2.605	1.10	0.024	98.876
Fuel slug	3.165 - 3.640	1.10	0.024	98.876

TABLE II
Measured Reactivity of Arrays of Tubular Slug Pairs
Containing 1.1 wt% Enriched U

Pitch, in.	V_{H_2O}/V_U	K_{calc}^2, cm^2	λ, cm	$B_m^2, \mu B$	Array ^a	K_{eff}^b μB	k_{eff}
4.25 ^c	2.777	29.9	6.3	2407 ± 16	4 x 5	3013 ± 14	0.913
					5 x 5	2055 ± 8	0.935
					6 x 5	1476 ± 7	0.950
					6 x 6	901 ± 4	0.964
4.00 ^c	2.335	29.7	6.5	3146 ± 48	4 x 5	2734 ± 30	0.921
					5 x 5	1702 ± 17	0.945
					6 x 5	1087 ± 7	0.960
					6 x 6	473 ± 5	0.976
3.80 ^c	2.001	29.7	7.1	3325 ± 110	4 x 5	2761 ± 5	0.921
					5 x 5	1723 ± 11	0.945
					6 x 5	1127 ± 4	0.960
					6 x 6	467 ± 3	0.976
3.64 ^c (contact)	1.746	29.8	7.7	3199 ± 63	4 x 5	3015 ± 29	0.914
					5 x 5	1990 ± 27	0.938
					6 x 5	1374 ± 24	0.953
					6 x 6	736 ± 3	0.969
4.00 ^d	1.881	29.7	7.1	3598 ± 109	5 x 5	1630 ± 5	0.947
					6 x 5	1078 ± 14	0.961
					19 assay hex	2660 ± 31	0.923
					31 assay ^e	685 ± 9	0.971
					37 assay hex	52 ± 3	0.986

a. The array height was 55.57 inches (5 slugs high).

b. From exponential theory $K^2 = B_m^2 - (B_x^2 + B_y^2)$

c. Square Pitch

d. Triangular Pitch

6 x 5 Array

6 x 5 Array

37 Assay Hex

e. 37 assay, hex with 6 corners removed.

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Shortcomings of the Albedo Approximation in KENO Calculations,* J. K. Long (ANL-Idaho)

An albedo option is incorporated in the KENO Monte Carlo code¹ providing considerable savings in computer time compared to the full stochastic calculation, especially for hydrogenous reflectors. The spatial distribution of returning neutrons is ignored in this option, making it applicable only to rectangular reflectors with "large" faces. The effect of this approximation is to attribute too great an importance to neutrons that enter the reflector from the edges of corners of an array. The albedo option will overestimate reactivity because of this

approximation. The extent of the overestimate in criticality introduced by this option, for array dimensions down to 50 cm on a side, was reported previously.² In the present study, a series of additional calculations has extended the comparison between albedo and stochastic calculations to small sizes, and has indicated that the errors introduced by this short-cut are larger than would have been anticipated, though still in the direction of overestimates of criticality.

Calculations were performed in which cubic cores as small as 25 cm on a side, of low-density 93%-enriched uranium, were surrounded by a water reflector. Core density was adjusted to achieve multiplication of about unity. The difference between albedo and stochastic calculations is shown by the triangles in Fig. 1. Similar calculations for a 2 x 2 x 2 array of dense uranium cylinders in the same volumes are shown by the squares in Fig. 1. Earlier reported data from Whitesides and Thomas for 2 x 2 x 2 arrays² are also shown.

Although the present array calculations agree with those of Whitesides and Thomas in the region of overlap, they show much larger errors for smaller arrays. The

*Sponsor: J. I. Sackett

Fig. 1. Difference between rigorous solution and albedo vs edge size.

present calculations also show substantially larger errors in the albedo approximation when the fuel is distributed through the core volume than when it is clustered in an array, even for a 50-cm core size.

Calculations for still more compact cores, though not shown on the figure, have indicated errors as high as 30% for the albedo approximation.

It is concluded that discretion must be used in applying the present albedo approximation to arrays <50 cm on a side, and even to larger cores if the fuel is adjacent to the reflector, since the albedo approximation can significantly overestimate reactivity.

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2. G. E. WHITESIDES and J. T. THOMAS, "The Use of Differential Current Albedos in Monte Carlo Criticality Calculations," *Trans. Am. Nucl. Soc.*, 12, 889 (1969).

Monte Carlo Analysis of Experimentally Critical Pipe Intersections, N. F. Cross, G. E. Whitesides, R. J. Hinton (ORNL)

Nuclear safety analysts often encounter situations in which criticality evaluations must be made for systems involving large tubing or piping arranged in various configurations. Those involving intersections cannot be calculated using the popular 1-D codes and no standards have been developed for the calculational reliability of other codes for this type of system. In this endeavor, calculations were made to determine the adequacy of the KENO¹ and CHERI² codes to treat critical systems of pipe intersections. Both the 16-group Hansen-Roach³ cross-section set and the 123-group XSDRN⁴ cross sections were used so cross-section effects could be evaluated. Ten of the critical experiments evaluated were performed at Rocky Flats⁵ and the remaining 21 were done at Oak Ridge.^{6,7}

The Rocky Flats experiments used a square central column and cylindrical arms filled with aqueous solutions of $U(93)O_2(NO_3)_2$. All the experiments were unreflected and the configurations included 2 square arms at 45 deg, 6 and 8 cylindrical arms at 45 deg to the central column, and 6, 8, 12, and 16 cylindrical arms at 90 deg to the central column. The average computed k_{eff} 's for the complete set of Rocky Flats experiments was 0.9994 ± 0.0027 when using the 123-group cross sections, indicating that these cross sections accurately reflect the

criticality of the experiments. However, the 16-group Hansen-Roach cross sections yield an average k_{eff} of 0.9776 ± 0.0027 for this set of experiments.

The critical experiments done at Oak Ridge involved a 30-deg lateral (2 cylindrical aluminum pipes intersecting at 30 deg, with one pipe extending below the intersection) and a 90-deg Plexiglas cross. Both sets of experiments were reflected by water. Six different solution concentrations of $U(5)O_2F_2$ were used in the 30-deg lateral experiments. The average calculated k_{eff} , using 16-group cross sections, was 0.9957 ± 0.0051 , and the average calculated k_{eff} for the 123-group set was 0.9960 ± 0.0034 . Several different critical experiments were made utilizing the 90-deg Plexiglas cross filled with $U(5)O_2F_2$ solution. Four inside diameters and several solution concentrations were used for a total of 15 critical experiments. The average calculated k_{eff} for the 16-group cross sections was 0.9967 ± 0.0022 and for the 123-group cross sections was 0.9999 ± 0.0022 , indicating that both cross-section sets accurately predict the criticality of these 5% uranyl fluoride pipe intersection experiments.

It was concluded that when using the 123-group cross sections, both KENO and CHERI accurately predict criticality for pipe intersections filled with highly enriched uranyl nitrate and 5% uranyl fluoride, regardless of the number of arms involved in the intersection. The 16-group Hansen-Roach cross-section set accurately predicts criticality (less than 2 standard deviations from critical) for the fully reflected low-enriched uranyl fluoride systems, but gives nonconservative results for the bare optimally moderated highly enriched uranyl nitrate systems.

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KENO Calculations Pertaining to Interaction Between Low-Enriched Fissile Arrays, Fred G. Welfare (B&W)

The purpose of this work is to evaluate the degree of isolation obtained by placing various media between fissile arrays.

Using the Monte Carlo code KENO¹ and the Hansen-Roach² 16-group cross-section set, the 4%-enriched, near-optimum moderation systems shown in Figs. 1 and 2 were analyzed. A quantity called the "interaction frac-

Fig. 1. Nuclear isolation of slabs by various materials.

Fig. 2. Nuclear isolation of arrays of cylinders by various materials.

tion" was defined as

$$\rho = (K_S - K_I) / (K_0 - K_I)$$

where K_S is the effective multiplication factor at separation S , K_I is the effective multiplication factor for a single array reflected by the isolating material, and K_0 is the effective multiplication factor when the arrays are in contact. The uncertainty in ρ was evaluated³ as

$$\Delta\rho = \left[\left(\frac{\partial\rho}{\partial K_S} \right)^2 \Delta K_S^2 + \left(\frac{\partial\rho}{\partial K_I} \right)^2 \Delta K_I^2 + \left(\frac{\partial\rho}{\partial K_0} \right)^2 \Delta K_0^2 \right]^{1/2}$$

where ΔK_S , ΔK_I , and ΔK_0 represent the uncertainty in the calculation of the multiplication factors.

The quantity ρ is equal to the fraction of the maximum interaction which remains at a spacing S . Therefore, ρ is equal to one when the systems are in contact and equal to zero when there is no interaction. For purposes of this work, systems were assumed to be isolated when ρ was reduced to within one standard deviation of zero. The isolation conditions calculated are for the specific situations described. They will not necessarily apply to other systems.

The results obtained are shown in Figs. 1 and 2. The uncertainties shown in these figures are one standard deviation. The following conclusions may be reached by examination of these results:

1. The large slabs are isolated to within one standard deviation by 8 in. of water, 30 in. of concrete, or 7 ft of void combined with 4 in. of concrete. These slabs are not isolated by 17 ft (i.e., the largest dimension of the orthogonal projection) of void, though the interaction is small at that point.

2. The arrays of cylinders are isolated to within one standard deviation by 10 ft of void, 6 in. of water, 1 ft of concrete, and 2 ft of void combined with 4 in. of concrete.

3. The amount of any of the isolating media required is a sharp function of the geometry of the arrays to be isolated.

4. Many systems of interest may indeed be isolated by the "traditional" 1 ft of concrete, but many other systems may not.

5. Separation by void can be combined with smaller thicknesses of concrete to isolate fissile units.

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Criticality of Plutonium-Uranium Nitrate Solutions Containing 30 wt% Pu, R. C. Lloyd, E. D. Clayton (PNL)

Data are required on the criticality of Pu-U nitrate solutions¹ to accurately establish nuclear safety limits for use in processing and handling of LMFBR-type fuels. Since the fuel must be processed both safely and economically, it is necessary that criticality considerations be based on accurate experimental data. Little experimental data on Pu-U nitrate solutions with 30 wt% Pu (typical of LMFBR fuels) have heretofore been reported.² The current research emphasis is on the procurement of criticality data for plutonium-uranium mixtures that will serve as the basis for handling criticality problems subsequently encountered in the development of technology for the LMFBR community. The experiments also provide useful benchmarks for data testing and analysis on

Fig. 1. Criticality of plutonium-uranium solution in cylindrical geometry.

TABLE I
Criticality of (U + Pu) Solution in Cylindrical Geometry*

Pu Conc. ^a g/liter	U Conc. g/liter	H ⁺	Specific Gravity	Critical Height (cm)	Critical Volume (liter)	Critical Mass (kg Pu+U)
97.3	200.4	4.8	1.562	19.96	58.37	17.37
87.9	184.3	4.4	1.512	19.48	57.00	15.52
75.4	168.5	3.7	1.450	18.82	55.04	13.42
63.1	147.7	2.9	1.389	18.72	54.76	11.54
49.6	122.3	2.4	1.320	19.35	56.59	9.73
39.9	96.7	2.1	1.257	20.32	59.44	8.12
30.2	72.5	2.2	1.213	22.89	66.97	6.88
25.3	60.9	2.0	1.182	25.12	73.48	6.33
20.3	48.9	1.8	1.154	29.92	87.53	6.06
17.2	41.5	1.7	1.132	36.04	105.44	6.19
14.1	34.4	1.5	1.117	52.60	153.91	7.46
13.3	32.2	1.4	1.110	64.01	187.24	8.52
12.4	29.9	1.3	1.103	95.20	278.50	11.78
^a Isotopic Composition: ²³⁹ Pu - 93.964 wt% ²⁴⁰ Pu - 5.585 wt% ²⁴¹ Pu - 0.352 wt% ²⁴² Pu - 0.072 wt% ²³⁸ Pu - 0.027 wt% ²³⁸ U - 99.324 wt% ²³⁶ U - 0.012 wt% ²³⁵ U - 0.659 wt% ²³⁴ U - 0.005 wt%						

*Preliminary chemical analysis data

integral criticality experiments for verification of the analytical techniques used in support of criticality control.

The experiments were performed with Pu-U nitrate solutions in a water-reflected stainless-steel cylindrical vessel. The wall thickness of the stainless-steel cylinder was 0.079 cm. The Pu-U nitrate solutions had ~30 wt% Pu in Pu+U. The plutonium contained 5.6 wt% ²⁴⁰Pu; the uranium 0.7 wt% ²³⁵U. In these experiments, the Pu+U concentration was varied throughout a range from 42.3 to 297.7 g/liter. The critical height and volume of solution were measured as a function of Pu+U concentration.

The data are presented in Table I. The variation in critical mass and volume as a function of fuel concentration is shown in Fig. 1. For this large cylinder, a minimum mass is obtained with these solutions at a Pu+U concentration of about 70 g/liter; the minimum volume occurs for a concentration of ~200 g/liter. The criticality factors were computed using ENDF/B cross-section data in the KENO³ Monte Carlo code. Epithermal energy groups were obtained through application of the ETOG and GAM cross-section processing codes, and the thermal-

group data via the FLANGE code followed by the TEMPEST portion of the GAMTEC⁴ code. The results of these calculations, illustrating the accuracy of these methods to the Pu-U nitrate solutions, will be discussed.

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Safety Analysis of Shipping Packages and Casks—I

Criticality Safety Evaluation of Wooden Pallets for Transporting N-Reactor Fuel, Hans Toffer, J. P. McNeece, K. L. Fowler (United Nucl Indus)

As operator of the Hanford N-Reactor, United Nuclear Industries is faced with a variety of fissile material shipping problems. Transportation of both irradiated and unirradiated fuel takes place between Hanford contractors¹ and off-site locations.² The largest amount of fissile material, in the form of finished fuel assemblies, is shipped by an exclusive-use motor carrier from the UNI Fuel Fabrication Facility to the N-Reactor plant 30 miles away.

The fuel assemblies are transported by truck in the shipping and storage pallet, shown in Fig. 1. A maximum of 36 uranium metal fuel assemblies with an enrichment of either 0.95 or 1.25 wt% ²³⁵U can be loaded in one container. The pallet's design fixes the fuel elements at a lattice pitch of 4.75 in., a distance, which on water flooding, would guarantee subcriticality for the fuel shipment.

Criticality of fuel-containing pallets under normal and accident conditions was analyzed with both the KENO Monte Carlo³ code and the HAMMER-SYSTEM.⁴ The KENO code was used to evaluate the effect on reactivity of the wooden container structures such as walls and fuel spacers, and to calculate k_{eff} for arrays of pallets. Moderation by water, hydrocarbons, and sand-water mixtures was considered. All are possible materials to which fuel assemblies may be subjected during an accident.

The HAMMER calculations provided criticality parameters for infinite systems of fuel assemblies latticed in these moderators. The results of the calculations are contained in Table I. Under normal transport conditions and for accidents involving rearrangement of pallets and moderator addition, the fuel shipments are safe, $k_{eff} < 0.90$. These calculations were carried out for close-spaced pallets. Increasing the separation between the containers reduced the k_{eff} values.

The criticality of the infinite systems of fuel assemblies latticed at 4.75 in. and at optimum arrangements in the different moderators are expressed in terms of k_{∞} .

Fig. 1. Fuel shipping pallet with fuel elements.

TABLE I
CRITICALITY STATUS OF FUEL SHIPMENTS ASSOCIATED WITH VARIOUS ACCIDENT STATES*

	Shipment Condition		Finite System				Infinite System		
	Pallet Arrangement	Pallet Structure	Fuel Lattice, in.	Moderator Density g/cm ³	k _{eff}		Fuel Lattice, in.	Moderator Density g/cm ³	k _∞
Normal Transport:									
Long row of pallets, two high	2 x 1 x 12	Undamaged	4.75	Wood; 0.5	0.59				
Accident Cases:									
Pallets rearranged No moderator addition	2 x 3 x 4	Undamaged	4.75	Wood; 0.5	0.66	4.75	Hom. Wood; 0.25	1.10	
	Disarrayed	Destroyed				4.75	Hom. Wood; 0.25	1.10	
Pallets rearranged partial water addition	2 x 3 x 4	Undamaged	4.75	Water; 0.2	0.75	4.75	Water; 0.20	1.13	
	Disarrayed	Destroyed				3.00	Water; 1.00	1.14	
Pallets rearranged full water addition	2 x 3 x 4	Undamaged	4.75	Water; 1.0	0.62	4.75	Water; 1.0	0.72	
	Disarrayed	Destroyed				3.00	Water; 1.0	1.14	
Pallets rearranged hydrocarbon addition	2 x 3 x 4	Undamaged	4.75	Hydro-Carbon; 0.9	0.55	4.75	Hydro-Carbon; 0.9	0.64	
	Disarrayed	Destroyed				2.50	Hydro-Carbon; 0.9	1.13	
Pallets rearranged sand-water addition	2 x 3 x 4	Undamaged	4.75	Sand-Water; 1.91	0.70	4.75	80 vol% sand 20 vol% water	1.91	1.01
	Disarrayed	Destroyed				3.0	0 vol% sand 100 vol% water	1.0	1.14

*Fuel for which k_{eff} and k_∞ values are quoted are tube-in-tube mixed enrichment, 1.25 and 0.95 wt% ²³⁵U uranium metal fuel assemblies.²

A value of k_∞ < 1.0 indicates subcriticality for a finite system. If, however, the fuel-constraining pallet geometry is destroyed, the fuel assemblies are rearranged, and moderating material is introduced, the k_∞ can be greater than 1.0 and finite size critical configurations are possible. Accidents of such severity are incredible in the transport environment and under the meteorological conditions encountered on the Hanford reservation.

Based on the criticality calculations in Table I and credible accident situations, safe limits for the transport of fuel assemblies in wooden pallets were established.

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3. G. E. WHITESIDES and N. F. GROSS, "KENO—A Multi-group Monte Carlo Criticality Program," CTC-5 UCC, Computing Technology Center (1969).
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Packaging of LWR Fuel Assemblies for Transport, L. E. Hansen, R. Nilson (Exxon Nucl)

To meet the requirements set forth in Title 10, Part 71 of the Code of Federal Regulations, and indeed to protect the health and safety of the general public, an exten-

sive series of tests and analyses has been performed to demonstrate the structural integrity and safety of the Model 51032-1 shipping container when loaded with a variety of LWR fuel assemblies. A cutaway view of that shipping container is shown in Fig. 1. The container consists of an interior strongback capable of supporting two LWR fuel assemblies with an active fuel length of at least 12 ft. (Separator blocks are placed between the assemblies to provide a minimum separation of 6 in.) This strongback is attached to the external containment vessel by shock mounts and mounting brackets, and reinforcing ribs (stiffener rings) are used to provide rigidity to the sheet-metal containment vessel.

Preliminary analytical evaluations of this package indicated it was structurally capable of surviving a hypothetical transportation accident. To demonstrate this capability, and to ascertain information regarding the most reactive credible configuration of an array of damaged packages, three "free drop" tests were conducted in orientations most likely to produce extensive damage. For these tests, the shipping container was loaded to rated capacity and was fully instrumented with accelerometers placed along the principal acceleration axis. Interestingly, the measured peak load on the containment vessel was considerably greater than was assumed in previous calculations. As expected, these tests produced major deformations of the containment vessel. In all cases, however, the strongback remained within the container and the payload remained affixed to the strongback.

Post-test examination of the shipping packages permitted the relaxation of previously assumed minimum spacings between fuel assemblies in an array of damaged containers. Subsequent criticality safety evaluations using this larger minimum spacing between assemblies in

Fig. 1. Model 51032-1 shipping container.

an array of damaged containers permitted the acquisition of a Fissile Class I shipping license for a variety of fuel assemblies. Those assemblies include PWR and BWR fuel assemblies containing UO_2 with average enrichments up to 4.0 and 2.9 wt% ^{235}U , respectively. Additionally, the shipments licensed include fuel assemblies containing mixed $(\text{Pu}, \text{U})\text{O}_2$ fuel rods with up to 5.5 wt% plutonium admixed with natural uranium. For each of the assemblies licensed for shipment, the reactivity of a single package and of infinite arrays of damaged and undamaged containers was conservatively evaluated using the KENO Monte Carlo code. A maximum reactivity of 0.938 ± 0.005 was obtained for an infinite array of undamaged containers with optimum interspersed moderation between containers. Also, for infinite arrays of containers subjected to the hypothetical accident conditions, the maximum computed reactivity was 0.908 ± 0.011 . As can be seen, in all cases the computed reactivities were < 0.95 at the 95% confidence level.

Packaging for FFTF Driver Fuel Pins and Assemblies, J. B. Edgar (West-Hanford)

The Westinghouse Hanford Company, operator of the Atomic Energy Commission's Hanford Engineering Development Laboratory (HEDL), has the responsibility for design, construction, and operation of the Fast Flux Test Facility (FFTF). This responsibility includes the development of safe packaging for transporting FFTF fuel. Two packages have been developed and are now in use: the Model 60 and the F-144-1. This paper describes the design, testing, and safety analysis of these packages.

The Model 60 is a Fissile Class II, Type B container designed for the shipment of FFTF driver fuel pins under controlled shipping conditions. It consists of a 6-in. schedule 40 $\times 101\frac{1}{2}$ -in. inside-length stainless-steel containment vessel rigidly mounted centrally in an 18- $\times$ 18- $\times$ 112 $\frac{1}{2}$ -in. steel birdcage.

The F-144-1 is also a Fissile Class II, Type B container designed for shipment of either FFTF driver fuel pins or FFTF driver fuel assemblies under uncontrolled shipping conditions. The shipping container consists of a 6-in. schedule 40 stainless-steel containment vessel centrally suspended in a 42- $\times$ 42- $\times$ 118-in. steel birdcage by a system of 16 damped spring-cable-pully devices.

The DOT Special Permits for these containers allow the following contents:

	Model 60	Model F-144-1
Fuel Form	≤ 31 wt% $\text{PuO}_2\text{-U(nat)O}_2$	SS-clad $\text{PuO}_2\text{-U(nat)O}_2$ fuel pins
Amount	7.0-kg Pu	8.79-kg Pu
Special Requirements	Material must be clad or canned	Must be 217, 120, or 25 pin bundles

Model 60: Engineering analysis of structural integrity plus extrapolations of normal transport and hypothetical

TABLE I
Calculated Fraction of Critical Number Represented by
120 Pins in Water in the Model 60 Containment Vessel
(Full Water Reflection)

Number of Pins in Model 60	Effective Lattice Pitch, in. (Hexagonal)	Water-to-Fuel Volume Ratio (Vw/Vf)	Calculated Critical Number of Pins at Vw/Vf	Fraction of Critical Number in Model 60
120 (max)	0.475	4.91	300	0.40
110	0.497	5.48	272	0.40
100	0.521	6.17	246	0.41
90	0.550	7.02	217	0.41
80	0.584	8.08	193	0.41
70	0.625	9.45	172	0.41
60	0.675	11.25	153	0.39
50	0.740	13.79	137	0.36
40	0.828	17.59	127	0.31
30	0.958	13.95	125	0.24

accident tests performed on the Model 10 (a smaller container of similar configuration) showed that the Model 60 was sufficient to qualify as a Type B package.¹

A criticality safety analysis² was carried out using a combination of experimental approach-to-critical data for PuO₂-UO₂ fuel pins similar to those to be transported in the Model 60 and computer code calculations (see Table I). This analysis showed that a Model 60 may be loaded with up to 120 FFTF driver fuel pins and that a single Fissile Class II shipment may consist of up to 12 loaded containers.

F-144-1: A safety analysis³ of the shipment of FFTF driver fuel pins and assemblies in the F-144-1 has been performed. The structural analysis, consisting of engineering calculations and computer modeling, showed the F-144-1 sufficient to meet the normal transport and hypothetical accident conditions of 10 CFR (Appendixes A and B).

The criticality safety analysis of the F-144-1 showed that the critical number of undamaged containers is 51 and that, under hypothetical accident conditions, the maximum k_{eff} of four damaged containers is 0.73 [0.697 ± 4 (0.007)]. The addition of a fifth container would not significantly increase k_{eff} under hypothetical accident conditions; consequently, the transport index was conservatively taken to be 50/5 = 10.

TABLE II

Values of k_{eff} for Four-Containment-Vessels-Array
Under Hypothetical Accident Conditions
(KENO II Monte Carlo Code)

	$k_{eff} + \text{One Sigma}$
No water between containment vessels	0.697 ± 0.007
Full density water between containment vessels	0.630 ± 0.014
10% density water between containment vessels	0.688 ± 0.009

Road testing programs⁴ have been carried out with both containers to assure that the driver fuel pins and assemblies transported in them will arrive undamaged under conditions of transport to which they are subjected.

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2. C. L. BROWN, "Criticality Safety Analysis—Model 60 Shipping Package for FFTF Driver Fuel Pins," BNWL-1512, Battelle-Northwest (Nov. 1970).
3. "Safety Analysis Report for the Shipment of FFTF Fuel in the F-144-1 Shipping Containers," Battelle Columbus Laboratories (May 1972).
4. J. B. EDGAR, unpublished data, Westinghouse Hanford Company (March 1972). Internal Report: Results of the Evaluation of the Model 60 Shipping Container for FFTF Pin Shipments.

Criticality Control for Multiple Array of Light-Water Reactor Fuel, J. L. Ridihalgh (BCL)

A study was made to characterize the use of physical separation and thermal poisons for criticality control in large (7 to 10 assemblies) spent-fuel shipping casks. Only PWR¹ fuel was considered because of the generally higher enrichments and greater resultant reactivity in shipping cask configurations. k_{eff} calculations were performed using the KENO² code for 3 × 3 arrays of the fuel assemblies. Each assembly was surrounded by a channel box containing a thermal-neutron poison and the entire 3 × 3 array was surrounded by a reflector. For the cases considered here, the channel box material is $\frac{1}{2}$ -in. Boral clad on both sides with $\frac{1}{2}$ -in. stainless steel. The general geometry used in the calculations is shown in more detail in Fig. 1. The fuel regions are homogenized, but all other regions are modeled discretely.

A summary of the KENO calculations is shown in Table I. Several significant trends can be inferred from examination of these results.

1. As expected, physical separation decreases k_{eff} for fuel arrays surrounded by poison. The magnitude of this effect is seen by comparing cases 7, 8, and 9. In these cases, aluminum was placed between the fuel/channel box arrays. The aluminum provides little moderating effect, thus the change in k_{eff} can be attributed mainly to changes in physical separation.

Fig. 1. Geometry description for KENO calculations.

2. As expected, a lead reflector around the array increases k_{eff} when compared to a water reflector. The magnitude of this effect is seen by comparing cases 1 and 4, 2 and 5, 3 and 6. Since lead is a typical gamma shielding material for spent-fuel shipping casks, this effect will have to be considered for most casks.

3. The increased reactivity effect of the lead reflector can be negated by introducing at least 1 in. of water between the fuel array and the lead reflector. Cases 10 through 12 show this trend. With the introduction of 1 in. of water between the array and the lead (case 11), k_{eff} is reduced to the same value as the array with only a water reflector (case 1).

4. A significant reduction in the k_{eff} of these fuel/channel box arrays is made by placing water between the channels. This effect can be seen by comparing cases 4 through 6, which have water between the channels, with cases 7 through 9, which have aluminum between the channels.

5. The effect of varying the ^{235}U content of the fuel from 3.0 to 4.0% is shown in cases 13 through 18. The array configuration for these cases included 0.5 in. of water between the channels holding the fuel and 1 in. of water between the 3×3 array of fuel/channel and the lead reflection.

The validity of the modified Hansen and Roach³ cross sections and the selection of the ^{238}U potential scattering cross-section (σ_p) value were tested by calculating k_{eff} for several critical assemblies.^{4,5} A σ_p value of 40 was selected for all cases in this study. Using this σ_p value, the average calculated k_{eff} for the critical experiments was 1.038 ± 0.011 . This indicates the 3×3 PWR fuel ar-

TABLE I
Calculated k_{eff} Values for 3×3 Arrays of PWR Fuel*

Case No.	Spacing Between Channels ^a	Material Between Channels	Reflector	Enrichment ^b wt% ^{235}U	k_{eff}
1	0.5	H_2O	8-in. H_2O	3.4	0.94 ± 0.01
2	1.0	H_2O	8-in. H_2O	3.4	0.86 ± 0.01
3	1.5	H_2O	8-in. H_2O	3.4	0.83 ± 0.01
4	0.5	H_2O	8-in. Pb	3.4	0.99 ± 0.01
5	1.0	H_2O	8-in. Pb	3.4	0.92 ± 0.01
6	1.5	H_2O	8-in. Pb	3.4	0.87 ± 0.01
7	0.5	Al	8-in. Pb	3.4	1.03 ± 0.01
8	1.0	Al	8-in. Pb	3.4	0.99 ± 0.01
9	1.5	Al	8-in. Pb	3.4	0.97 ± 0.01
10	0.5	H_2O	1 in. H_2O then 8-in. Pb	3.4	0.97 ± 0.01
11	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.4	0.94 ± 0.01
12	0.5	H_2O	3-in. H_2O then 8-in. Pb	3.4	0.94 ± 0.01
13	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.0	0.906 ± 0.010
14	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.2	0.927 ± 0.010
15	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.4	0.937 ± 0.008
16	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.6	0.945 ± 0.009
17	0.5	H_2O	1-in. H_2O then 8-in. Pb	3.8	0.965 ± 0.010
18	0.5	H_2O	1-in. H_2O then 8-in. Pb	4.0	0.969 ± 0.010

*PWR fuel is 15×15 pin array, 8.426×8.426 -in. overall dimension, 0.422-in. pin o.d., 0.563 pin pitch. See WASH 1082, pp. 5-6 for complete fuel description.

^aSee Fig. 1 for detailed geometry description.

^bPotential scattering values: ^{235}U , $\sigma_p = 2000$; ^{238}U , $\sigma_p = 40$.

ray calculations are conservative and probably have a 2 to 3% conservative bias.

The results of the calculations indicate that criticality control in large spent-fuel shipping casks can be enhanced by (a) surrounding each fuel assembly with a channel containing thermal poison material such as Boral; (b) separating individual channels by 0.5 in. or more and ensuring water or other hydrogenous material remains between the channels; and (c) assuring that the fuel in the inner cavity of the cask is separated from the lead (or uranium) gamma shield by at least 1 in. of water.

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An Engineering Approach to Criticality Safety Analysis in an Irradiated Fuel Cask, J. D. Rollins (Nucl Assurance Corp), R. E. Best (AGNS)

There are different ways to approach the nuclear criticality safety analysis of an irradiated fuel cask. This is true in terms of the analytical tools and nuclear data used as well as in the case of fuel configurations considered. In the former case, approaches range from that of a pure "physics" approach in which k_{eff} is calculated to a high degree of accuracy with specialized codes and a multigroup cross-section set to that of an "engineering" approach in which k_{eff} is determined with a generalized code and a few-group cross-section set to a less accurate but more conservative value.

In the case of fuel configurations considered, approaches to criticality analysis range from that of analyzing fixed configurations of intact fuel assemblies to that of analyzing variable configurations of deformed fuel assemblies; i.e., those simulating geometries under hypothetical accident conditions.

In the criticality safety analysis of the NFS-4 and NFS-5 casks, the aforementioned engineering approach was followed. The KENO¹ computer program and Hansen and Roach² sixteen-group cross data (resonance absorber cross sections) were used and assumptions made to determine conservative values of k_{eff} . Two levels of sophistication were used in determining the resonance absorber parameter, σ_p . These included (a) the use of absorber lump parameters only (NFS-4 cask analysis), and (b) the use of a more sophisticated method involving benchmarking and normalization of calculated k_{∞} values (NFS-5 cask analysis). In the latter case, the LASER computer program was used to determine accurate values

TABLE I

Results of Benchmarking Calculations for 15 Selected Reactor Critical Experiments

Experiment	k_{eff} (LASER Corrected)	k_{eff} (KENO-H&R)
(Ref. 4)	(Ref. 3)	
TRX 1	0.978	1.012
2	0.984	1.023
3	0.983	1.032
4	0.991	1.028
5	0.980	1.017
6	0.972	1.008
7	0.981	1.021
8	0.980	1.022
(Ref. 5)		± 0.006 Max
BAW 1	0.991	1.006
2	0.993	1.056
13	0.982	1.000
20	0.982	1.035
(Ref. 6)		
YANKEE 1	0.991	1.020
2	0.996	1.024
3	0.986	1.003

of k_{∞} for infinite pin cell systems. These values provided a basis for normalizing KENO results using the Hansen and Roach cross sections. Table I indicates the conservatism of the simplified approach.

One of the more difficult tasks in the criticality safety evaluation of irradiated fuel casks is that of determining the most reactive fuel assembly type for both normal and accident conditions. In the criticality analysis of the NFS-4 and NFS-5 casks, the conservative approach used was that of analyzing the fuel under the worst possible conditions. These conditions included fuel pin bowing (increase in fuel-to-moderator ratio and concomitant increase in fuel assembly envelope dimensions) and fuel pin drop out (increase in fuel-to-moderator ratio).

For the case of BWR-fuel-type assemblies in the NFS-5 cask, k_{eff} increased more than 9% before dropping off at the bowing limit (boundary restriction). A similar analysis of PWR-type assemblies in the NFS-5 cask indicated a k_{eff} increase of over 4% before boundary limits were reached. In the latter case, the reactivity was still increasing as the boundary was reached. The reactivity increase due to fuel pin drop out was less than that due to fuel assembly bowing in both of the above cases.

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The Minimum Critical Concentration of Plutonium-239 in Water

Minimum Critical Concentration of ^{239}Pu ,
Hugh K. Clark (duPont-SRL)

"The American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" gives a subcritical concentration limit of 7.3 g/liter for ^{239}Pu in a uniform aqueous solution.¹ According to the standard, this limit was chosen to be far enough below the minimum critical concentration to compensate for uncertainties in the latter and hence to ensure subcriticality regardless of the volume of solution. However, there appears to be a strong possibility that this limit actually exceeds the minimum critical concentration, and that it should be replaced by a smaller value.

The minimum critical concentration as measured in a Physical Constants Testing Reactor (PCTR) experiment has been reported to be 8.0 ± 0.3 g ^{239}Pu /liter (Ref. 2). Although by the same techniques the experimenters obtained a limiting concentration of 12.1 ± 0.1 g ^{235}U /liter, in excellent agreement with critical experiments,³ there is ample reason to doubt that their estimate of the uncertainty in the plutonium concentration is sufficiently large. Large corrections were required for neutron absorption by the steel tanks holding the plutonium solution, and there is considerable room for opinion as to how extrapolation to zero thickness should be made. The report contains several discrepancies. Among these is one between concentration and $H/^{239}\text{Pu}$ ratio, which was later noted by Richey, who corrected the concentration to agree with the ratio, obtaining 7.8 g/liter (Ref. 4). The precision of the experiment appears insufficient to warrant the choice of a subcritical limit as high as 7.3 g/liter, and solely on the basis of this experiment, the limit should perhaps be chosen to be as low as 6 g/liter.

There are no critical experiments with plutonium at concentrations near 8 g/liter, but there are experiments at concentrations ranging upward from about 25 g/liter (Refs. 5 and 6). Bias in calculational techniques may be established by correlations with these experiments, and if the bias is a slowly varying function of concentration (or alternatively of $H/^{239}\text{Pu}$), extrapolation of the bias to a concentration of 7 to 8 g/liter should be fairly reliable.

With S_4 theory, Richey calculated an average k_{eff} of 1.015 ± 0.005 for eight critical experiments performed with aqueous solutions of plutonium⁴ containing 4.6% ^{240}Pu . Six of the experiments were with water-reflected spheres; two, with bare spheres. On the basis of this correlation, he corrected the minimum critical concentration of 7.2 (that he calculated for a k of unity) to 7.5 ± 0.1 g ^{239}Pu /liter to correspond to a k of 1.015. Plotting his correlations against $H/^{239}\text{Pu}$ ratio, however, reveals a slight downward trend in k_{eff} with increasing ratio, particularly if attention is focused on the experiments with the smallest amounts of excess nitrate. Extrapolation of this trend indicates there may be very little bias near 7 g/liter, and that the minimum critical concentration may be only slightly greater than 7.2 g ^{239}Pu /liter.

Two other techniques were correlated with the same critical experiments with water-reflected spheres. The first of these involved a 12-group B_0 calculation to obtain the buckling and two-group parameters for a subsequent diffusion theory calculation of extrapolation distance.⁷

A fairly strong trend was found in the bias, indicating a minimum critical concentration of 7.2 compared with an uncorrected value of 7.66 g/liter. The second of these consisted of S_4 calculations with Hansen-Roach cross sections.⁸ Again, a trend appears to exist, this time toward higher values of k_{eff} as $H/^{239}\text{Pu}$ increases; and the minimum critical concentration appears to be about 7.3, compared with an uncorrected value of 7.0 g/liter.

Finally, direct calculations were made by HAMMER⁹ and the Savannah River Laboratory's successor to HAMMER, RAHAB,¹⁰ which has the same group structure but uses ENDF/B-III cross sections. The minimum critical concentrations obtained were, respectively, 7.39 and 7.19 g ^{239}Pu /liter.

In conclusion, the subcritical concentration limit should make allowance for the strong possibility that the minimum critical concentration may be as low as 7.2 g ^{239}Pu /liter. A limit of 7.0 g/liter is suggested.

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Reexamination of the Value for the Minimum Critical Concentration of ^{239}Pu in Water, C. R. Richey (PNL)

Masterson et al.¹ reported a value of $8.0 \pm 0.3 \text{ g } ^{239}\text{Pu/liter}$ for the limiting critical concentration of ^{239}Pu in water. This value was determined by the well-known null reactivity method in the PCTR facility at Hanford.² As recently pointed out by Clark,³ large corrections were necessary in this experiment for spectral perturbations caused by the stainless-steel tanks containing the plutonium solutions in the reactor core. The correction for the stainless steel was determined in the original experiment by making measurements with varying thicknesses of steel containers and extrapolating the results to a zero wall thickness.

To check the validity of the correction for stainless steel, a theoretical model of the original experiment was developed with an 18-group diffusion theory calculation using the HFN code.⁴ Seventeen fast-group constants $E > 0.683 \text{ eV}$ were obtained with the EGGNIT code⁵ and spectra-dependent thermal constants were computed with THERMOS.⁶ From this model, the calculated value for the Pu concentration that gives a null reactivity ($K^\infty = 1.0$) is 12.08 g Pu/liter (uncorrected for stainless-steel vessel, NO_3 , and ^{240}Pu). This compares with the 12.2-g Pu/liter value measured by Masterson et al. The computed null reactivity for $\text{Pu}(\text{NO}_3)_3$ aqueous solution, corrected for stainless steel, is 8.58 g Pu/liter and compares with the 8.7-g Pu/liter value obtained by Masterson et al. Applying the computed corrections for stainless steel, NO_3 , and ^{240}Pu to the experimental null reactivity value (12.2 g Pu/liter) results in a net value of $8.04 \text{ g } ^{239}\text{Pu/liter}$ for the limiting critical concentration of ^{239}Pu in water. This compares well with the 8.0 ± 0.3 value reported by Masterson et al.

Clark suggested that theoretical methods might be utilized to determine the limiting critical concentration for ^{239}Pu by validating a calculational method at higher concentrations where these are critical data and then extending the area of applicability. Clark also suggested that bias (eigenvalues differing from unity and variations in eigenvalues with $\text{H}/^{239}\text{Pu}$ atom ratio) exhibited by the calculational model be accounted for in extending the range of applicability. A summary of both S_n and diffusion theory calculations for six fully water-reflected critical experiments with $\text{Pu}(\text{NO}_3)_3$ solutions computed by both Clark³ and Richey⁷ is given in Table I along with corresponding bias, projected limiting critical concentrations, and eigenvalues at a $\text{H}/^{239}\text{Pu}$ of 4000. The diffusion theory calculations by Richey are in much better agreement with the critical experiments and exhibit the least bias in $\Delta k_{\text{eff}}/\Delta(\text{H}/^{239}\text{Pu})$. However, it is known that diffusion theory overestimates the leakage with decreasing vessel size; therefore, the slope $\Delta k_{\text{eff}}/\Delta(\text{H}/^{239}\text{Pu})$ should become positive as the vessel size approaches infinite where the eigenvalues for the S_n and diffusion theory calculations become equal. This fact points out that to extend the calculational method from a high leakage range to one of low leakage is also encumbered by pitfalls that could be equally as dangerous and questionable as the stainless-steel extrapolation made in the original experiment by Masterson et al.

1. R. H. MASTERSON, J. D. WHITE, and T. J. POWELL, "The Limiting Critical Concentration for ^{239}Pu and ^{235}U in Aqueous Solution," HW-77089, General Electric Co., Hanford Atomic Products Operation (1963).
2. D. J. DONAHUE, D. D. LANNING, R. A. BENNETT, and R. E. HEINEMAN, *Nucl. Sci. Eng.*, 4, 297 (1958).

TABLE I
Summary of S_n and Diffusion Theory Calculations for
Water-Reflected Spheres Containing $\text{Pu}(\text{NO}_3)_3$
(^{240}Pu ; 4.6 wt%; Thickness of Water: 15 cm)

Diam of Sphere (in.)	g Pu/liter	g NO ₃ /liter	H/ ^{239}Pu	k_{eff}			
				Clark S_n	Clark Diffusion Theory	Richey S_n	Richey Diffusion Theory
11.47	73	86	370	0.9854	1.017	1.021	1.001
	268.7	346	91	0.9978	1.015	1.018	1.001
13.94	33	162	795	0.9989	1.000	1.015	0.999
	47.9	465	485	0.9924	1.006	1.019	1.005
15.21	24.4	58.1	1119	1.0031	0.993	1.014	1.000
	38.7	517	579	1.0011	1.003	1.013	1.007
BIAS [$\Delta k_{\text{eff}}/\Delta(\text{H}/^{239}\text{Pu})$]				$+9.78 \times 10^{-6}$	-2.39×10^{-5}	-5.69×10^{-6}	-2.08×10^{-6}
k_{eff} (at $\text{H}/^{239}\text{Pu} = 4000$)				1.030	0.924	0.997	0.995
Approx Limiting Critical Conc (g ^{239}Pu /liter)				>8.0	~6.5	7.14	7.1

3. H. K. CLARK, "Establishing Subcritical Limits," A paper presented at the Nuclear Criticality Safety Short Course at Taos, New Mexico (May 6-11, 1973).
4. J. R. LILLEY, "Computer Code HFN-Multigroup, Multiregion Neutron Diffusion Theory in One Space Dimension," HW-71545, Hanford Laboratories (Nov. 1961).
5. C. R. RICHEY, "EGGNIT: A Multigroup Cross Section Code," BNWL-1203, Pacific Northwest Laboratory (1969).
6. H. C. HONECK, "THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5836, Brookhaven National Laboratory (Sept. 1961).
7. C. R. RICHEY, "Theoretical Analyses of Homogeneous Plutonium Critical Experiments," *Nucl. Sci. Eng.*, 31, 32 (1968).

Safety Analysis of Shipping Packages and Casks—II

DOT-6M Packages for the Shipment of PuO_2 , Frank E. Adcock (Dow)

The DOT specification 6M container (Fig. 1) is approved for shipping up to 4.5 kg of plutonium in the form of oxide or metal. Thermal decay energy is limited to 10 W; however, DOT special permits have been issued allowing up to 65 W (see Table I). Numerous AEC contractors and licensees have been using this container since early 1967, first under Special Permit 5000 and later as a specification package.

The results of accident condition tests are well documented for virtually all possible variations of 6M-type containers. Sandia Corporation's report of extra-severe test conditions¹ is of particular interest. Typically, a 6M package weighing under 150 lb will show no loss of spacing from the 30-ft drop test. The contain-

ment vessel temperature will rise about 180°F during the half-hour thermal test.

Under normal operating conditions, assuming exposure to direct sunlight and 130°F air, and providing for an internal heating load of 10 W, the containment vessel would be expected to remain under 250°F. This is below the design limit of 285°F which is recommended for Celotex insulation.² Sealed food cans make an ideal inner container for dry oxide or metal at temperatures up to about 300°F. For higher temperatures, a heavier steel vessel must be provided.

The original criticality calculations were made at Rocky Flats by John McCarthy. Later, during the 6M utility review, the AIF appointed a working group which included Dave Smith of LASL and Joe Thomas of Oak Ridge. This group was responsible for establishing the material limits as they are now published.

TABLE I
Approved Plutonium Oxide Containers

	CONTAINER	PERMIT	PuO ₂ (kg)	WATTS	INSIDE DIM (in.)	OUTSIDE DIM (in.)	SAFETY ANALYSIS REPORT
6M TYPE	DOT-6M	--	4.5	10	5.25 max diam.	150 x 18 (minimum)	RFP-1042, April 1968
	Mound AL-M1	SP5790	4.5	29	40 x 6	150 x 18	o
	Mound AL-M3	SP5791	5.0	37	80 x 7	200 x 30	o
	Mound AL-M6	SP5791	5.3	65	6.50 x 16	240 x 35	o
OTHERS	KKD/LLD	SP4960	4.5	21	50 x 12	16 sq. x 25	DPSPU 71-124-1, Feb. 1971
	L-3	SP5330	4.0	10	4.80 x 18	240 x 35	RFP-437, Oct. 1964
	ORNL Foenglas	SP5795	3.9	--	4.80 x 18	240 x 35	ORNL-TH-3468, Dec. 1972
	LLL Foenglas	SP5848	4.5	--	120 x 12	300 x 28	UCRL-14903, Mar. 1972

¹Nuclear Criticality Safety Evaluation of Four Different Size Shipping Containers, Monsanto Research Corporation, July 24, 1968, and Heat Transfer Evaluation of Nine Different Size Shipping Containers, Monsanto Research Corporation, January 1968.

Fig. 1. Disassembled DOT-6M container showing Celotex industrial board insulation, steel containment vessel with pipe plug closure, and sealed food cans containing Pu oxide.

With the present emphasis on shipping plutonium in solid form, it appears that DOT-6M-type containers will continue to fulfill an important role in the nuclear industry.

1. L. F. STRAVASNIC, "Special Tests for Plutonium Shipping Containers 6M, SP5795, and L-10," SC-DR-72 0597, Sandia Corporation (Sept. 1972).
2. E. E. LEWALLEN, "Celotex Insulated Shipping Containers," BNWL-SA-3906, *Proc. Third Intern. Symp. Packaging and Transportation of Radioactive Materials* (Aug. 16, 1971).

Criticality Calculations for Two Plutonium Shipping Containers,* B. B. Ernst (NUMEC)

It is possible to do simple hand calculations using published constants and methods to show the criticality safety of both individual units and arrays of shipping containers.

NUMEC, a subsidiary of the Babcock & Wilcox Company, has evaluated and licensed two shipping containers using hand calculations. One is the DOT SP-4824 which is used for shipment of high-enriched uranium compounds, plutonium compounds, and plutonium alloy fuel plates. The other is the DOT SP-4877 which is a nitrate shipping container for both plutonium and uranium.

The DOT SP-4824 is a 65-gal 17-H container with an inner container centered by 0.25-in.-diam spring steel rod spacers. A 5-in. steel tube insert with a flanged gasketed closure is placed inside this inner container. For fuel plate shipments, a fuel plate holder fits inside this insert.

The criticality calculations for the plutonium fuel plate holder are given as one example and the calculations for the nitrate container as another.

The density analog method¹ is used to obtain a conservative estimate of the critical mass of plutonium when diluted with natural uranium. Both reflected and bare critical values are calculated.

Then using one-group constants from ANL-5800, a value of K_{∞} is calculated. From ANL-5800 a value of the extrapolation distance is obtained. This is used with the values of K_{∞} and sphere radius obtained above to calculate an effective migration area. These constants are then used to calculate a value of K_{eff} for the fuel plate holder.

The solid angle of interaction² is calculated for 37 containers for both normal and accident conditions. The total solid angle subtended in both cases is less than that allowed for the calculated K_{eff} . Therefore, a Class III shipment of 19 containers is allowed.

At the present time no plans exist to modify the container. The container will continue to be used to ship high-enriched powders and plutonium powders if the need arises. There are no plans to use the container for shipment of plutonium metal fuel plates.

As a second example, the DOT SP-4877 nitrate shipping container has also been evaluated using hand calculations. This container is basically a steel outer drum filled with vermiculite. The fissile material is contained in a polyethylene bottle which in turn is doubly bagged and placed in a cadmium-wrapped stainless-steel pressure vessel. A 2-in.-thick polyethylene-polyester resin composite surrounds the cadmium-wrapped pressure vessel.

Data from ARH-600 show that the minimum infinite critical cylinder diameter for uranyl nitrate is smaller than the minimum infinite critical cylinder diameter of plutonium nitrate. Therefore, the analysis is done for uranyl nitrate. Using B^2 , M^2 , and K_{∞} from ARH-600, and an experimentally determined λ for cadmium-wrapped cylinders, K_{eff} is calculated for the cylinder.

Then, using neutron penetration data,³ it is shown that an infinite array of containers, when separated by greater than the 3.4 in. of polyethylene, is safe.

This container will continue to be used to ship both plutonium and uranium nitrate solutions. It is planned to substitute a dual-purpose stainless-steel bottle in place of the plastic bottle. This new bottle will also be used as an in-plant storage vessel.

It is also intended to keep the plastic bottle and to ship uranium chloride solutions in it.

These examples show that it is possible to do criticality evaluations of shipping containers using hand calculations and readily available published data to show the safety of both the individual units and arrays of units. Since these hand calculations cannot describe the system exactly, certain simplifying conservative assumptions are made. Assumptions such as ignoring the steel in the fuel plate holder, the steel in the pressure vessel, and the metal other than plutonium in the fuels, give an inherent conservatism to the hand calculations.

1. W. R. STRATTON, "Criticality Data and Factors Affecting Criticality of Single Homogeneous Units," LA-3612, Los Alamos Scientific Lab.
2. Subcommittee 8 of the American Standards Association Sectional Committee N6 and Project 8 of the American Nuclear Society Standards Committee, *Nuclear Safety Guide*, TID-7016, rev. 1, USAEC.
3. H. F. HENRY, C. E. NEWLON, and J. R. KNIGHT, "Extensions of Neutron Interaction Criteria," K-1478, Oak Ridge Gaseous Diffusion Plant.
4. H. F. HENRY et al., "Studies in Nuclear Safety," K-1380, Oak Ridge Gaseous Diffusion Plant.
5. R. D. CARTER et al., *Criticality Handbook, Volume II*, ARH-600, Atlantic-Richfield Hanford Company.
6. *Reactor Physics Constants*, ANL-5800, 2nd ed., USAEC, Division of Technical Information.

Evaluation of ORNL Foamglas Shipping Container, J. R. Parrott, L. B. Shappert (ORNL)

ORNL has designed and constructed 65 foamglas shipping containers which are approved for the shipping of fissile materials in the form of oxide, metal, or solutions. The container, approved by the Department of Transportation Special Permit No. 5795, consists of a section of 5-in. Schedule-80 pipe suspended in foamglas contained in a 55-gal carbon steel drum. A cross section of the container is shown in Fig. 1.

The container has undergone thermal, drop, and puncture tests¹ as well as a criticality evaluation.² The mass limits imposed as a result of the criticality calculations are shown in Table I. The Keno³ Monte Carlo Code and the Hansen-Roach⁴ neutron cross-section sets were used to determine the neutron multiplication factor of cubic

Fig. 1. Foamglas shipping containers.

arrays of the containers when closely reflected on all faces by a 20-cm-thick water reflector. Mass limits were established to satisfy the condition that K_{eff} would not exceed a value of 0.97 ± 0.01 . The constant cell volume associated with the container in an array permits a linear interpretation to be made of the number of units as a function of the mass loading for arrays with constant K_{eff} . Since package tests verified the integrity of the inner container, the allowable mass limits under the damaged package criteria exceed those of normal transport.

The container maintained its integrity after drop tests from a height of 30 ft onto a flat unyielding surface, puncture tests by dropping a distance of 40 in. onto a 6-in.-diam vertical bar, and exposure in a fire at 1475°F for 50 min. Since the inner container remained intact after the tests, the fourth condition of the hypothetical accident—immersion in water—was met.

1. B. B. KLIMA and L. B. SHAPPERT, "Drop, Puncture, and Thermal Tests Performed on the Foamglas Shipping Container, DOT Special Permit No. 5795," ORNL-TM-3713, Oak Ridge National Lab. (April 1972).
2. J. T. THOMAS, Union Carbide Corporation, Personal Communication (Aug. 1973).
3. G. E. WHITESIDES and N. F. CROSS, "KENO-A Multigroup Monte Carlo Criticality Program," CTC-5, Oak Ridge Computing Technology Center (1969).
4. GORDON E. HANSEN and WILLIAM H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos Scientific Lab. (1961).

TABLE I
ORNL FOAMGLAS SHIPPING CONTAINER
MAXIMUM MASS (kg x)

TRANSPORT INDEX	METAL	DIOXIDE $H/X \approx 0.4$	DIOXIDE $H/X \approx 3$	AQUEOUS SOL'N (ANY CONC'N)
URANIUM-235				
	$\rho \approx 18.76$	$\rho \approx 8.09$	$\rho \approx 4.48$	
0.1	17.2	18.8	15.4	2.35
0.2	19.5	24.1	18.4	—
0.3	20.8	28.2	20.4	—
0.5	23.0	33.8	23.0	—
1.0	25.6	43.7	27.4	—
2.0	28.8	56.1	32.8	—
PLUTONIUM-239				
	$\rho \approx 19.7$	$\rho \approx 8.73$	$\rho \approx 4.71$	
0.1	5.3	10.5	9.2	2.35
0.2	5.6	11.6	10.7	—
0.3	5.8	12.4	11.8	—
0.5	6.2	13.5	13.3	—
1.0	6.5	14.9	15.6	—
2.0	7.0	16.5	18.3	—
URANIUM-233				
	$\rho \approx 18.4$	$\rho \approx 8.08$	$\rho \approx 4.46$	
0.1	7.5	11.1	8.3	1.35
0.2	8.0	12.6	10.1	1.76
0.3	8.3	13.7	11.5	2.00
0.5	8.8	15.0	13.3	2.35
1.0	9.2	17.1	16.2	—
2.0	9.9	19.3	19.8	—

Design Improvements in the L-10 Shipping Container, E. F. Curren (ARHCO)

In view of considered action by the Atomic Energy Commission in banning the shipment of plutonium nitrate in solution form, a review of the history of the L-10 shipping container, improvement in its design, shipping experience, and other factors, is appropriate. This container is used, almost exclusively, for the shipment of plutonium nitrate solutions in type B quantities.

The L-10 shipping container is licensed under Department of Transportation Special Permits 5061 and 6615 and meets all DOT and AEC requirements.

The container was originally developed and tested by Dow Chemical U.S.A. at Rocky Flats, Colorado, at the request of the AEC Division of Nuclear Materials Management for the shipment of plutonium nitrate solution.

The L-10 shipping container, in the Class II configuration, consists of two 55-gal DOT 17H steel drums welded end to end. A tubular steel "bird cage" supports and centers a Specification 2R pressure vessel in which a 10-liter polyethylene bottle, carrying the plutonium nitrate solution, is carried. The annular space between the pressure vessel and drum wall is filled with loose vermiculite.

A family of containers has evolved from the basic Dow design. The first variation of this design was originated by the New York State Atomic and Space Development Authority with redesign of the pressure vessel utilizing Schedule-40 pipe and a secondary containment for the polyethylene bottle consisting of a light-gauge stainless-steel "capsule." This version of the L-10 container was used for the shipment of plutonium nitrate solution from the West Valley, NY, reprocessing plant to Hanford.

Plutonium nitrate solutions are shipped, normally, at an authorized concentration of 250 g/liter in nitric acid at a maximum concentration of 6.0 M. Alpha radiolysis of the solution results in pressure building within the polyethylene bottle which must be vented. Modification of the standard 83-mm closure was necessary to provide venting capability. Two types of vents were developed: a slit vent arrangement, and a vent incorporating a patented valve known as the Duo-Vent.

Migration of solution through the vents presented a problem which was solved by providing a restraining device between bottle cap and the closing flange which ensures that the bottle will vibrate at the same frequency as its container during normal transport.

A Class I L-10 container, designated FL-10-1, was developed by Nuclear Containers, Inc., Elizabethton, Tennessee, which incorporates a double O-ring seal for the pressure vessel and a solid fire-retardant foam in place of the conventional bird cage and loose vermiculite.

Operating experience with the L-10 family of shipping containers, involving literally many hundreds of shipments, has been excellent. There have been no incidents involving the release of solution from the 2R container. As recently as 1972, the container, in the original design configuration, was subjected to extra severe testing without impairment of pressure vessel integrity. Earlier, a safety analysis of the L-10 pressure vessel revealed that the vessel would be undamaged by an internal hydrogen-oxygen explosion.

Nuclear Criticality Safety Especially for Mixed Oxides

Progress Toward Criticality Safety Standards for Mixed Oxides, H. K. Clark (SRL), invited

Increasing interest in mixtures of plutonium and uranium dioxides as reactor fuels has led to an effort within the American Nuclear Society's Subcommittee ANS-8 to formulate criticality safety standards for mixed-oxide operations. The work group comprises ten members representing AEC licensees and contractors, the UK, and Japan. The work group hopes to have a mixed-oxide standard by mid-1975. Although the basic principles in the general criticality safety standard¹ are sufficient, additional subcritical limits are needed. By placing reliance on, and hence controlling, the isotopic concentration of ²⁴⁰Pu and the ratio of UO₂-to-PuO₂, the single-parameter limits for ²³⁹Pu may be increased.

As in the general standard, the subcritical limits will contain margins designed to be sufficiently large to allow for uncertainties in calculations and experimental data used in their derivation, but simultaneously, sufficiently small to discourage attempts to justify larger limits. Similarly, the margins will contain no allowance for contingencies (e.g., double-batching or inaccuracy of analytical techniques). Operations may be performed safely by complying with any limit together with the conditions under which it is valid. Compliance means simultaneously maintaining all parameters (e.g., mass, PuO₂ concentration, % ²⁴⁰Pu, reflector composition, etc.) within their respective limits. Maintaining a parameter within its limit implies making allowance for credible contingencies.

Homogeneous systems are being considered first, but limits for lattices are planned. The following compositions were selected as having the greatest utility. The oxide mixtures contain 30, 15, 8, and 3% PuO₂ by weight. The uranium is natural. Isotopic compositions of plutonium are: 100% ²³⁹Pu; 79% ²³⁹Pu, 15% ²⁴⁰Pu, 6% ²⁴¹Pu; 60% ²³⁹Pu, 25% ²⁴⁰Pu, and 15% ²⁴¹Pu (²³⁸Pu and ²⁴²Pu are ignored).

Limits will be provided for both dry and water-moderated systems. A completely dry oxide system may be difficult to guarantee; hence, limits will also be provided for damp [H/(U + Pu) < 0.45] oxide.² Solutions and slurries are assumed to be uniform, homogeneous mixtures of UO₂ (10.96 g/cm³) and PuO₂ (11.46 g/cm³ for ²³⁹PuO₂) in water. Critical dimensions of such systems have minima as a function of concentration, provided the oxide mixture does not contain much more than 30% PuO₂ by weight.

Although Pacific-Northwest Laboratories (PNL) have a continuing program of critical experiments with mixed-oxide systems, data are sparse and not directly translatable into limits. Rather, these data are chiefly of value for normalizing the many calculations required to generate limits. These calculations are being made by several members of the work group with a wide variety of methods (including cross-section sets). Agreement, although no guarantee, should be a good indication of the correctness of the results. Calculations made so far for moderated systems agree fairly well. Most of the differences

in calculated critical conditions amount to <1% in *k*_{eff}. The maximum difference of about 2% occurs mainly for slabs. Critical masses for unmoderated systems calculated at PNL by DTF-IV with ENDF/B-III cross sections processed by ETOG and FLANGE, however, are larger than those calculated at the Savannah River Laboratory (SRL) by ANISN with Hansen-Roach cross sections by a margin in *k*_{eff} of 3 to 4%. For dry 30% oxide at theoretical density, PNL, SRL, and Chalmers (England) obtained 31.6, 28.1, and 28.3 kg Pu, respectively, and for damp oxide at $\frac{1}{2}$ theoretical density, 91.7, 75.6, and 89.6 kg Pu, respectively.

Publication of proposed limits now would be premature. Differences need to be resolved. No limit should exceed the lowest critical value obtained by a reasonably valid method, but margins for uncertainties remain to be selected. Eventually, limits representing the consensus of the work group as being subcritical, but not excessively so, together with their derivation, should be published in the open literature by the work group to serve as a basis for the standard.

1. American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI N16.1-1969, American National Standards Institute.
2. RICARDO ARTIGAS, "Density Effects on the *k*_∞ of Shipping Packages," *Trans. Am. Nucl. Soc.*, 15, 311 (1972).

Status of Criticality Data and Experiments on Homogeneous Systems of Plutonium-Uranium, S. R. Bierman (Battelle-Northwest), invited

The primary emphasis to date in obtaining experimental data on mixed plutonium-uranium systems has been to obtain information on the critical sizes of homogeneous systems having well-defined spherical, cylindrical, or rectangular geometries and Pu enrichments between 8 and 30 wt%. Emphasis has also been placed on obtaining the data over a wide range of concentrations, selected to give the more useful information in the shortest period of time. One such concentration region of interest is that at which the minimum mass and the minimum critical volume occurs for different Pu enrichments. At present, experimental data are available in the H: Pu + U regions at which these minima occur for 8, 15, and 30 wt% Pu-enriched mixtures of aqueous Pu-U nitrate and hydrogenous PuO₂-UO₂, except for that region in which the minimum critical mass occurs for 15 wt% Pu enrichment. Also, data do not exist in the concentration regions at which the minimum critical volume occurs for the 8 and 15 wt% Pu-enriched nitrate solutions. However, calculations indicate that these minima occur around 450 and 600 g (Pu + U)/liter, which are outside the stable concentration range for the nitrate solutions.

Although the experiments performed to date have been primarily with fuel mixtures having a low ²⁴⁰Pu content, there is an economic incentive, because of the high exposures planned for the LMFBR fuels, to eventually obtain

TABLE I
Available Experimental Data on Homogeneous Pu-U Mixtures

Concentration g (Pu + U)/liter	²⁴⁰ Pu wt% in Pu	Geometry and Reflection	Source
~ 30 wt% Plutonium Enrichment			
42 to 298	5.58	Water-reflected 61-cm-diam cylinder ^a	Lloyd, <i>Trans. Am. Nucl. Soc.</i> , 17, 269 (1973)
57, 61, 103, 330	5.63	Water-reflected 50.7-cm-diam cylinder	Lane, AWRE 0 58173 (1973)
61, 103, 330	5.63	Water-reflected 38-cm-diam cylinder	Lane, AWRE 0 58173 (1973)
103, 330	5.63	Water-reflected 30.6-cm-diam cylinder	Lane, AWRE 0 58173 (1973)
330	5.63	Water-reflected 25.4-cm-diam cylinder	Lane, AWRE 0 58173 (1973)
111	4.66	Water-reflected 19.3-cm-radius sphere	Lloyd, <i>Trans. Am. Nucl. Soc.</i> , 15, 803 (1972).
228	4.66	Water-reflected 17.9-cm-radius sphere	Lloyd, <i>Trans. Am. Nucl. Soc.</i> , 15, 307 (1972).
529	8.0	Bare and Plexiglas-reflected parallelepipeds	Bierman, <i>Nucl. Sci. Eng.</i> , 50, (1973).
3973	5.9	Polythene-reflected parallelepipeds	Lane, AWRE 0 32/68 (1968)
4807	11.5	Bare and Plexiglas-reflected parallelepipeds	Bierman, <i>Trans. Am. Nucl. Soc.</i> , 15, 307 (1972).
~ 15 wt% Plutonium Enrichment			
311	4.66	Water-reflected 19.3-cm-radius sphere	Lloyd, <i>Trans. Am. Nucl. Soc.</i> , 15, 803 (1972).
783	8.0	Bare and Plexiglas-reflected parallelepipeds	Bierman, <i>Nucl. Sci. Eng.</i> , 50 (1973).
5000	11.5	Bare and Plexiglas-reflected parallelepipeds	Bierman (in progress) (1974)
~ 8 wt% Plutonium Enrichment			
304 to 421	22.8	Water-reflected 61-cm-diam cylinder ^b	Lloyd, <i>Trans. Am. Nucl. Soc.</i> , 15, 307 (1972).
484	8.0	Plexiglas-reflected parallelepipeds	Bierman, <i>Nucl. Sci. Eng.</i> , 50 (1973).
1198	23.0	Plexiglas-reflected parallelepipeds	Bierman, <i>Trans. Am. Nucl. Soc.</i> , 15, 805 (1972).

^aExperiments in progress with this vessel loaded with $\frac{1}{2}$ wt% borated Raschig rings. Experiments planned with gadolinium nitrate solution.

^bExperiments were performed, also, by diluting this solution with 477 g U/liter $\text{UO}_2(\text{NO}_3)_2$ solution. Wt% Pu varied from 7.3 to 5.5.

experimental data on mixtures containing 30 to 40 wt% ²⁴⁰Pu. Critical experiments with fuel mixtures in this exposure range are currently being planned. Also, data should soon be available on Pu-U nitrate solutions containing fixed and soluble neutron poisons.

The experimental criticality data currently, or soon to become, available for homogeneous plutonium-uranium mixtures are summarized in Table I along with a brief reference to the originating source. Although the results from some of the experiments listed in Table I have not been published in sufficient detail to permit their use in validating calculational techniques or cross-section data, the referenced source can be contacted for whatever additional details are needed. It is anticipated that a complete detailed compilation of all the data will be available soon.

With the advent of the Monte Carlo codes, such as KENO, GEM, and MONK, it has not been as important that the experimental data be obtained in, or be capable of being reduced to, one-dimensional geometries. Consequently, considerably more of the actual critical assemblies are useful for checking calculations. Also, since the Monte Carlo technique can accommodate geometric irregularities, some geometric perturbations can be tolerated in the experimental assemblies. However, this wider degree-of-freedom, in the design and performance of the experiments, places a greater burden on reporting the experimental results to assure that the experimental assemblies are completely and unambiguously defined. Also, whenever possible, the experimental data should still be reported in such a manner that the widest range of calculational techniques can utilize the data. A proposed format for reporting such experimental data will be presented for constructive suggestions on how it can be better compiled and presented.

Status of Experimental Criticality Data for (Pu,U)O₂ Fuel Rods, C. L. Brown (Batelle-Northwest), invited

A complete summary of experimental criticality data for (Pu,U)O₂ fuel rods in water is presented in Tables I and II.¹⁻⁴ Fuel rod dimensions and isotopic compositions are presented in Tables III and IV. The information is sufficiently complete for use in computer-code validation.

The experimental data applicable to LWR fuel criticality safety (Table I) are reasonably complete. Measurements cover rod sizes from 0.34 to 0.50 in.; ²⁴⁰Pu content in the range of 8 to 24 wt%; and fuel enrichments from 1.5 to 6.6 wt% PuO₂. This span provides a good basis for calculations directed toward the criticality safety of PWR and BWR plutonium recycle fuels.

A series of calculated minimum critical parameters for plutonium recycle fuel rods was reported in Ref. 5. These calculations show that, based on minimum critical number of rods, 5 wt% ²³⁵U-enriched UO₂ is equivalent to ~UO₂-3.7 wt% PuO₂, where the plutonium contains 5 wt% ²⁴⁰Pu and the uranium is natural.

The experimental data applicable to LMFBR fuels (Table II) are much less extensive: only one fuel has been measured to date. To provide an adequate data base for criticality safety in the long term, four additional lattice critical experiments are needed to extend the data as follows:

1. a (Pu, U)O₂ fuel enrichment near 15 wt% PuO₂
2. a (Pu, U)O₂ rod size near 0.3-in. diam
3. a ²⁴⁰Pu content in (Pu, U)O₂ fuel near 25 wt%
4. similar measurements for (Pu, U) carbide and (Pu, U) nitride fuels.

TABLE I

Summary of Experimental Criticality Data For (Pu,U)O₂
Rod Lattices Applicable To LWR Fuels

Lattice Spacing ^a (in.)	H ₂ O Rod Volume Ratio	Approach-to-Critical Critical Size (No. rods)	Critical Mass (kg Pu)	Reflector Savings (cm)	Material Buckling (m ⁻²)
UO ₂ - 1.5 wt% PuO ₂ Rods in H ₂ O ⁽¹⁾ [7.83 wt% ²⁴⁰ Pu]					
0.55	0.8382	1487 ± 3	16.29	8.40 ± 0.06	48.0 ± 1.2
0.60	1.187	829 ± 7	9.08	7.99 ± 0.10	65.1 ± 1.8
0.71	2.063	484 ± 1	5.30	7.27 ± 0.03	78.5 ± 0.3
0.80	2.889	420 ± 1	4.60	6.93 ± 0.03	74.9 ± 0.3
0.90	3.922	452 ± 2	4.95	6.75 ± 0.03	60.9 ± 0.2
0.93	4.255	488 ± 2	5.35	6.64 ± 0.05	55.2 ± 0.2
UO ₂ - 2.0 wt% PuO ₂ Rods in H ₂ O ⁽²⁾ [7.65 wt% ²⁴⁰ Pu]					
0.80	1.211	319.7 ± 0.1	6.35	7.04 ± 0.04	93.7 ± 0.3
0.93	1.987	192.4 ± 0.1	3.83	7.52 ± 0.03	103.3 ± 0.3
1.05	2.808	152.1 ± 0.2	3.03	7.70 ± 0.06	101.3 ± 0.5
1.143	3.513	147.5 ± 0.1	2.93	7.11 ± 0.03	97.0 ± 0.2
1.32	5.019	163.1 ± 0.1	3.24	6.96 ± 0.06	75.6 ± 0.3
1.386	5.635	179.5 ± 0.2	3.57	6.35 ± 0.06	68.9 ± 0.3
[16.54 wt% ²⁴⁰ Pu]					
0.93	1.987	245.6 ± 0.1	4.89	7.85 ± 0.03	86.3 ± 0.2
1.05	2.808	194.3 ± 0.1	3.87	7.91 ± 0.06	85.4 ± 0.4
1.143	3.513	187.5 ± 0.1	3.73	7.32 ± 0.04	81.5 ± 0.2
1.32	5.019	221.1 ± 0.1	4.40	6.95 ± 0.06	61.6 ± 0.2
1.386	5.635	254.6 ± 0.1	5.07	5.85 ± 0.04	55.6 ± 0.2
[23.50 wt% ²⁴⁰ Pu]					
0.80	1.211	519.5 ± 0.1	10.34	8.21 ± 0.04	63.1 ± 0.2
0.93	1.987	286.1 ± 0.1	5.69	7.62 ± 0.04	79.4 ± 0.2
1.05	2.808	233.2 ± 0.1	4.64	7.50 ± 0.05	77.6 ± 0.3
1.143	3.513	232.1 ± 0.1	4.62	7.00 ± 0.03	72.2 ± 0.2
1.32	5.019	296.2 ± 0.1	5.89	5.79 ± 0.03	53.7 ± 0.1
1.386	5.635	365.3 ± 0.1	7.27	5.42 ± 0.04	44.3 ± 0.1
UO ₂ - 4.0 wt% PuO ₂ Rods in H ₂ O [18.10 wt% ²⁴⁰ Pu]					
0.85	1.500	252.6 ± 0.5	9.61	7.89 ± 0.04	94.7 ± 0.3
0.93	1.993	178.9 ± 0.2	6.81	7.55 ± 0.03	107.9 ± 0.3
1.05	2.815	138.9 ± 0.4	5.29	7.54 ± 0.04	108.7 ± 0.5
1.143	3.521	122.4 ± 0.1	4.66	7.29 ± 0.01	107.9 ± 0.1
1.386	5.647	123.6 ± 0.7	4.70	6.46 ± 0.06	88.4 ± 0.8
1.60	7.859	180.9 ± 2.1	6.89	5.30 ± 0.17	59.5 ± 1.1
1.70	9.000	271.9 ± 0.5	10.35	5.36 ± 0.09	41.1 ± 0.2
UO ₂ - 6.6 wt% PuO ₂ Rods in H ₂ O [8.57 wt% ²⁴⁰ Pu]					
0.52 ^b	-	472	15.02	8.56	108.8 ± 1.3
0.56	-	337	10.72	7.97	122.1 ± 1.1
0.735	-	151	4.80	6.83	159.6 ± 1.9
0.792	-	133	4.23	6.47	159.3 ± 2.2
1.04	-	117	3.72	6.23	128.4 ± 1.4

^aTriangular lattices except where noted^bSquare lattices

TABLE II
Summary of Experimental Criticality Data For (Pu,U)O₂
Lattices Applicable to LMFBR Fuels

Lattice Spacing ^a (in.)	H ₂ O/Fuel Volume Ratio	Approach-to-Critical Critical Size (No. Rods)	Critical Mass (kg Pu)	Extrapolation Distance (cm)	Material Buckling (m ⁻²)
UO ₂ - 25.2 wt% PuO ₂ Rods in H ₂ O [11.88 wt% ²⁴⁰ Pu]					
0.450	4.26	417.0 ± 5	12.8	8.01	153.8 ± 2.7
0.600	8.60	219.2 ± 0.6	6.74	7.25	172.7 ± 2.2
0.779	15.42	156.2 ± 0.8	4.80	6.89	160.9 ± 1.0
0.900	21.01	149.8 ± 0.6	4.61	6.90	138.5 ± 1.0
1.039	28.45	165.0 ± 0.2	5.07	6.61	111.6 ± 0.3
1.200	38.37	232.6 ± 0.1	7.15	6.11	77.0 ± 1.0
1.350	48.92	410.0 ± 10	12.6	----	~ 46.8
UO ₂ - 25.2 wt% PuO ₂ Rods in Boron Solution [0.542 g B/liter]					
0.600	8.60	333.4	10.25	6.70	141.8
[1.098 g B/liter] ^b					
0.600	8.60	~ 430	-	6.70	113
[0.410 g B/liter]					
0.900	21.01	322.8	9.90	5.9	91.7
[1.103 g B/liter]					
0.900	21.01	-	-	8.2	7.0
UO ₂ - 25.2 wt% PuO ₂ Rods in Cadmium Solution [0.879 g Cd/liter]					
0.900	21.01	328.4	10.10	5.99	90.1

^aTriangular lattice.

^bData obtained using two exponentials rather than approach-to-critical method.

TABLE III
Description of (Pu, U)O₂ Fuel Rods Used In Experiments

Fuel i.d.	Fuel			Cladding		
	Diameter (in.)	Length (in.)	Average Loading (g PuO ₂)	Material	o.d. (in.)	Wall (in.)
UO ₂ - 1.5 wt% PuO ₂	0.372	48.5	12.42	Zirc-2	0.462	0.027
UO ₂ - 2.0 wt% PuO ₂	0.505	36.0	22.56	Zirc-2	0.565	0.03
UO ₂ - 3.98 wt% PuO ₂	0.4975	36.0	43.16	Zirc-2	0.565	0.034
UO ₂ - 6.6 wt% PuO ₂	0.3374	36.6	36.074	Zirc-4	0.391	0.023
UO ₂ - 25.2 wt% PuO ₂	0.195	27.25	34.88	316 SS	0.230	0.015

Nonetheless, the experimental criticality data currently available for LMFBR fuels do provide a secure validation point for (Pu, U)O₂ fuels to be used in the fast flux test facility (FFTF) program. Indeed, the data in Table II were obtained for this program. As a consequence, the criticality safety parameters for FFTF fuel pin handling are currently based *directly* on the experimental data, since the fuel rods used in the experiments are nearly the same sizes as those to be used in the FFTF reactor. These criticality parameters are as follows:

minimum critical number: 140 pins
minimum critical cylinder diameter: 8.98 in.
minimum critical slab thickness: 3.64 in.

1. V. O. UOTINEN, J. H. LAUBY, L. C. SCHMID, and W. P. STINSTON, "Lattices of Plutonium-Enriched Rods in Light Water—Part I: Experimental Results," *Nucl. Technol.*, 15, 257 (1972).
2. G. G. TAYLOR, "Saxton Plutonium Program—Critical

TABLE IV
Isotopic Compositions of (Pu, U)O₂ Fuel Rods
Used in Experiments

Fuel I.d.	Isotopic Composition (wt%)					
	²³⁵ U	²³⁸ Pu	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu
UO ₂ - 1.5 wt% PuO ₂	0.16	---	91.41	7.83	0.73	0.03
UO ₂ - 2.0 wt% PuO ₂	0.711	---	91.62	7.65	0.70	0.03
	0.711	---	81.11	16.54	2.15	0.20
	0.711	---	71.76	23.50	4.08	0.66
UO ₂ - 3.98 wt% PuO ₂	0.711	0.28	75.38	18.10	5.08	1.15
UO ₂ - 6.6 wt% PuO ₂	0.711	---	90.49	8.57	0.89	0.04
UO ₂ - 25.2 wt% PuO ₂	0.711	0.04	86.19	11.88	1.73	0.16

Experiments for the Saxton Partial Plutonium Core,"
WCAP-3385, Westinghouse Electric Corp. (1965).

3. C. L. BROWN and C. A. ROGERS, "Critical Approach and Exponential Experiments With Prototype FFTF Driver Fuel Pins In Water," *Trans. Am. Nucl. Soc.*, 13, 663 (1970).
4. C. L. BROWN, L. C. DAVENPORT, and E. D. CLAYTON, "Subcritical Experiments With (Pu, U)O₂ Fuel Rods in Boron and Cadmium Poisoned Water," *Trans. Am. Nucl. Soc.*, 14, 675 (1971).

5. C. L. BROWN and D. R. ODEN, "Critical Parameters for 0 to 7% PuO₂-U(Nat)O₂ Rods In Water," *Trans. Am. Nucl. Soc.*, 12, 873 (1969).

Calculation-Experiment Correlations on Homogeneous Pu-U Systems, R. J. Hall, S. R. Bierman (Battelle-Northwest)

Because of the great number of parameters involved in a multigroup nuclear criticality calculation, extensive experimental benchmarks are required to test calculational results. The calculation of just one experimental system yields information on the ability of a computational method to calculate that particular experimental system but gives little assurance as to accuracy on a system with different physical characteristics. (Good answers can be obtained fortuitously from chance combinations of physical characteristics which mask e.g., inaccurate basic data or poor cross-section averaging.) Calculations on experiments involving a wide range of fuel materials and neutron spectra increase the ability to identify cross-section errors according to the material and energy range in which they occur. For example, the properties of small, low-moderation bare assemblies essentially depend only on the high-energy data. Overmoderated assemblies, on the other hand, essentially depend only on the thermal data.

TABLE I
Some Correlations Possible with Current Experimental Data

Principle Parameter Varied	Parameters Held Constant	Experimental System Numbers ^a
Pu/(Pu + U)	H/X (~4) ²⁴⁰ Pu (~11.5 wt%)	5, 14
Pu/(Pu + U)	H/X (~17)	4, 12, 13
Pu/(Pu + U)	H/X (~50)	1, 3, 11
H/X	²⁴⁰ Pu (~8 wt%)	1, 2, 3, 13
H/X	Pu/(Pu + U) (~7.5%) ²⁴⁰ Pu (~23 wt%)	4, 10 (several data points)
H/X	Pu/(Pu + U) (~14.6%)	2, 8
H/X	Pu/(Pu + U) (~30%)	1, 5, 6, 7, 9 (several data points)

^aExperimental Systems (all constructed at the PNL Critical Mass Laboratory)

System Number	Fuel ^b	wt% Pu ^c	wt% ²⁴⁰ Pu ^d	H/X ^e	H/Pu
1	PuO ₂ -UO ₂ -Polystyrene	30.0	8.0	47.4	158
2	PuO ₂ -UO ₂ -Polystyrene	14.62	8.0	30.6	210
3	PuO ₂ -UO ₂ -Polystyrene	7.89	8.0	51.8	659
4	PuO ₂ -UO ₂ -Polystyrene	7.58	23.0	19.5	259
5	PuO ₂ -UO ₂ -Polystyrene	29.33	11.5	2.8	9.5
6	Pu-U Nitrate	31.11	4.66	100	321
7	Pu-U Nitrate	31.65	4.66	224	708
8	Pu-U Nitrate	14.68	4.66	76	518
9	Pu-U Nitrate	30	5.59	Varies	Varies
10	Pu-U Nitrate	~5-7	22.8	Varies	Varies
11	PuO ₂ -Polystyrene	100	18.35	49.6	49.6
12	PuO ₂ -Polystyrene	100	2.2	15	15
13	PuO ₂ -Polystyrene	100	8.0	15	15
14	PuO ₂ -Polystyrene	100	11.46	5.0	5.0

^bThe uranium is either natural or depleted

^cin Pu + U

^din Pu

^eH/X = atomic ratio of hydrogen-to-plutonium plus uranium in the fuel

Although additional experiments are required, the quantity and range of experimental data on Pu-U systems are now sufficient to initiate a systematic study of calculations covering a reasonably broad range of physical characteristics pertinent to fuel recycle operations. To facilitate this study, experimental data were grouped in various combinations to emphasize the variation of important criticality parameters. Pertinent groupings of a number of experiments, showing the principal parameters varied, are provided in Table I. The results of some of the correlations made to date, using both new and previously reported calculations, are discussed in this paper. These correlations will, when completed, permit better definition of shortcomings in calculational techniques and contribute to assessment of the accuracy of calculations as a function of the important criticality parameters.

A variety of codes have been used to perform criticality calculations. Use of the ENDF/B cross-section data in these calculations is emphasized. The FLANGE and ETOG codes are used to process the ENDF/B data for input to a spectrum-averaging group-reduction code (either EGGNIT¹ or GAMTEC-II). The 18-group (1 thermal, 17 fast) cross sections generated are used in either the HFN, DTF-IV, or KENO II code.

Figure 1 shows the results of criticality calculations on a number of $[Pu/(Pu + U) \sim 0.30]$ systems as a function of H/X . A trend or bias in the calculations is apparent in the figure. The calculations agree well with experiment at low moderation; however, at higher moderation ($H/X > 47$) there is a fairly constant, average $2\frac{1}{2}$ to 3%, overprediction in k_{eff} . This type behavior suggests a problem in either the thermal data or the thermal data processing and spectral averaging. The thermal spectral averaging in EGGNIT is performed as in TEMPEST. To study the effect of different spectral averaging, several calculations were repeated using thermal-group data obtained from the BRT-1² code.

The different thermal-group data had essentially no effect on the low-moderation system; however, it improved (typically by ~ 10 mk) the correlation for the more highly moderated systems. Use of ENDF/B-II, rather than ENDF/B-III, data also improved the correlation (7-mk average reduction in k_{eff}) for the higher moderation systems while having minimal effect (possibly slightly increasing k_{eff}) on the $H/X = 2.8$ results. Thus, for the systems in Fig. 1, ENDF/B-II data and BRT-1 spectral averaging reduce the overprediction of k_{eff} to $1\frac{1}{2}$ to 2% at higher moderation, while at low moderation essentially not differing from results using ENDF/B-III and TEMPEST. Further correlations as given in Table I are in progress to better define the source of experiment-calculation discrepancies.

Fig. 1. Calculations of Pu-U systems with $Pu/(Pu + U) \approx 0.30$.

1. C. R. RICHEY, "EGGNIT: A Multigroup Cross Section Code," BNWL-1203, Battelle Northwest Labs. (1969).
2. C. L. BENNETT and W. L. PURCELL, "BRT-1: Battelle-Revised-Thermos," BNWL-1434, Battelle Northwest Labs. (1970).

Criticality of UO_2 - PuO_2 Solids in Fissile Solution, R. C. Lloyd (Battelle-Northwest)

The dissolution of fuel elements involves conditions in which UO_2 - PuO_2 solids are immersed in fissile solutions. Critical masses, volumes, and dimensions are required for credible combinations of fuel, moderator, and reflector.

TABLE I

Variation of Critical Solution Height in Lattice Assembly

Plutonium Concentration (g/liter)	Uranium Concentration (g/liter)	Acid Molarity	Critical Solution Height (cm)
62.2	151.2	2.6	18.74
55.3	134.4	2.4	18.84
46.4	112.6	2.1	18.87
39.5	95.9	1.8	19.04
30.8	74.6	1.6	19.52
25.3	61.2	1.5	20.32
20.3	49.1	1.3	21.32
15.3	37.5	1.3	22.88
11.8	29.0	1.2	24.73
0.2	0.7	0.3	52.73

FUEL PIN DIMENSIONS

	o.d. (cm)	Length (cm)
Fuel column	0.495	69.22
Cladding (316 SS)	0.584	72.90

FUEL PER PIN

PuO_2-UO_2 : 138.4	1.3 g
Pu: 30.75 $\pm$ 0.03 g	
U: 91.16 $\pm$ 1.03 g	
O: 16.49 $\pm$ 0.17 g	

FUEL ENRICHMENT

25.2 wt% $PuO_2-U(nat)O_2$

FUEL DENSITY

10.35 $\pm$ 0.09 g/cm³
(93.34 $\pm$ 0.79% theoretical)

PLUTONIUM COMPOSITION OF PINS

²³⁸ Pu: 0.04 $\pm$ 0.01 at.%
²³⁹ Pu: 86.19 $\pm$ 0.06 at.%
²⁴⁰ Pu: 11.88 $\pm$ 0.06 at.%
²⁴¹ Pu: 1.73 $\pm$ 0.01 at.%
²⁴² Pu: 0.16 $\pm$ 0.01 at.%

SOLUTION COMPOSITION

²³⁸ Pu: 0.018 $\pm$ 0.0008 wt%	²³⁵ U: 0.659 wt%
²³⁹ Pu: 93.843 $\pm$ 0.015 wt%	²³⁶ U: 0.012 wt%
²⁴⁰ Pu: 5.700 $\pm$ 0.013 wt%	²³⁸ U: 99.329 wt%
²⁴¹ Pu: 0.360 $\pm$ 0.004 wt%	
²⁴² Pu: 0.079 $\pm$ 0.0004 wt%

Fig. 1. Lattice assembly.

tor. Until now there were no reported experimental data on Pu-U fuel elements in Pu-U nitrate solutions that might serve as computational checks. There have been experiments with pins in water¹ and with Pu-U nitrate solutions separately.^{2,3}

A series of criticality measurements was completed on a lattice of FFTF prototype fuel pins [25 wt% PuO₂-U (nat)O₂], immersed in Pu-U nitrate solutions of several different concentrations. The data from these integral critical experiments will be useful as a check on computational methods, and further, will provide the only data of this type directly applicable to the dissolver process.

The lattice assembly was composed of 301 fuel pins placed in a 55.5-cm-diam stainless-steel cylindrical vessel. A schematic diagram of the lattice assembly is shown in Fig. 1. A triangular pitch was used for the lattice with a 3.05-cm center-to-center spacing between pins. The composition and dimensions of the fuel pins used in the experiments are shown in Table I.

The Pu + U nitrate solution was made up of 30 wt% Pu in Pu + U. The ²⁴⁰Pu content of the Pu was 5.7 wt%, and ²³⁵U content of the uranium was 0.66 wt%. The Pu + U concentration was varied from over 200 to 40 g/liter, and in one case, a measurement was made on an essentially water-moderated lattice having no fissile material in the solution.

Data from these experiments are presented in the table. Data testing and analyses on these integral critical experiments, to verify analytical techniques with ENDF/B-III cross sections, are underway and will be discussed.

1. C. L. BROWN and C. A. ROGERS, "Critical Approach and Exponential Experiments With Prototype FFTF Driver Fuel Pins in Water," *Trans. Am. Nucl. Soc.*, 13, 663 (1970).
2. R. C. LLOYD, E. D. CLAYTON, and S. R. BIERMAN, "Criticality of Plutonium-Uranium Nitrate Solutions," *Trans. Am. Nucl. Soc.*, 15, 803 (1972).

3. R. C. LLOYD and E. D. CLAYTON, "Criticality of Plutonium-Uranium Nitrate Solutions Containing 30 wt% Pu," *Trans. Am. Nucl. Soc.*, 17, 269 (1973).

Predicting the Reactivity Effects of ²⁴¹Pu in Mixed-Oxide Lattices, Ricardo Artigas (GE-San Jose), invited

This paper presents a set of isoreactivity curves, applicable only to mixed-oxide rods of a certain design, which permit the determination by simple inspection, rather than laborious calculation, of whether a different plutonium isotopic composition in the same rod design is more or less reactive than a base design of known reactivity for criticality safety purposes. In effect, the curves incorporate reactivity equivalency factors for the three main isotopes of plutonium which are expected to vary as a function of recycling in reactors. An equation that predicts the percent reactivity increase of the mixed-oxide fuel rods where a certain percentage of the ²³⁹Pu isotope is replaced by ²⁴¹Pu is also given. Criticality safety limits for new plutonium isotopic composition of fuel rods of the same design, therefore, may be determined more rapidly.

For the computations, the design-basis mixed-oxide fuel rods have a pellet o.d. of 0.5 in. and a 0.03-in.-thick zirconium cladding. The uranium in the oxide is natural, and the rods are analyzed in an infinite triangular-pitched water lattice of various water-to-fuel ratios. The design-basis fuel rods will be simply called DBF rods in this paper. Modified versions of the THERMOS,¹ GAM,² and HRG³ codes are used to collapse multigroup sets of cross sections into three broad groups. Three-group two-dimensional diffusion theory cell calculations⁴ for different pitches determine the maximum or optimally moderated eigenvalue for each case.

Figure 1 shows for a given ²³⁹Pu wt% the minimum ²⁴¹Pu-to-²⁴⁰Pu weight ratio required to make the maximum reactivity of the DBF rods, with such plutonium composition, equal to the maximum reactivity of the DBF rods

with plutonium containing no ^{241}Pu , 85 wt% ^{239}Pu , and 15 wt% ^{240}Pu . The k_{∞} of this latter type of fuel is 1.501 for 8 wt% PuO_2 concentration and 1.442 for 4 wt% PuO_2 .

The closeness of the curves shown in Fig. 1 suggests that a single isoreactivity curve could be conservatively applied to the DBF rods. This single isoreactivity curve is applicable only to the DBF rods with PuO_2 concentrations not exceeding 8 wt%, and for maximum ^{239}Pu content of 85 wt%. Mixed-oxide rods, similar in design to the DBF rods, whose plutonium isotopic composition can be represented by a point that falls below the isoreactivity curve, are less reactive than the base system, and the criticality limits established for the base system are acceptable for the new rods.

Figure 2 shows the maximum increase in k_{∞} when a certain percent of the ^{239}Pu of the optimally moderated DBF rods is replaced by ^{241}Pu . That is, the (0, 0) point on the figure represents the maximum k_{∞} of the DBF rods with either 8 or 4 wt% PuO_2 concentrations, and plutonium with no ^{241}Pu and ^{240}Pu wt% between 10 and 30.

The near linearity of the curves suggests then that a single equation

$$(\Delta k)/k\% = 0.196 (\% \text{ } ^{239}\text{Pu} \text{ replaced by } ^{241}\text{Pu}) \quad (1)$$

could be used to conservatively predict increases in reactivity only for mixed-oxide rods such as the ones described here, and within the ranges indicated on Fig. 2, when a certain percent of the ^{239}Pu isotope is replaced by ^{241}Pu .

It should be emphasized that the results presented here are applicable only to rods similar in design to the design-basis fuel rods.

Fig. 1. Isoreactivity curves.

Fig. 2. Reactivity increase as a function of percent ^{239}Pu replaced by ^{241}Pu .

1. H. C. HONECK, "THERMOS—A Thermalization Transport Theory Code for Reactor Lattice Calculations," BNL-5826, Brookhaven National Lab. (1961).
2. P. P. WILCOX and S. T. PERKINS, "AGN-GAM; An IBM 7090 Code to Calculate Spectra and Multigroup Constants," AGN-TM-407 (Apr. 1965).
3. J. L. CARTER, "HRG-3—A Code for Calculating the Slowing Down Spectra in the B_1 or P_1 Approximation," BNWL-1432, Battelle-Northwest Labs. (June 1970).
4. T. F. THOMSON, "TWOD—A Program for the Solution of the Few Group Neutron Diffusion Equation in Two Space Dimensions," Users Manual, General Electric Company, NED (Jan. 1969).

Moderation Control for Processing Mixed-Oxide Fuels, Bruce Ernst (NUMEC), invited

The safe economical manufacturing of mixed-oxide fuel often requires large batch sizes which can only be achieved by controlling the amount of moderating material that is interspread within the fuel (moderation control); methods for achieving this must be both safe and economical. FFTF operations at NUMEC employ a unique method to do this. We specify the safe mass and amount of moderating material that is allowed. Table I shows some of the mass limits presently employed. In the table, the amount of moderating material is expressed as water equivalent. Water equivalent in this text means other moderating materials may be used if the amount of hydrogen in them does not exceed the amount of hydrogen in the allowable amount of water. These limits were derived

from calculations reported in Ref. 1; I have replotted these data, which are shown in Fig. 1.

Plotting data in this manner answers the following questions concerning the safety of an operation when H/X is one of the controlling parameters:

1. In a blending operation when the binder is added, is the system going to remain subcritical since the binder material causes a high localized H/X ratio even though the amount of fissile material with this ratio is small? The answer is "yes" if the amount of fissile material in the blender is less than the amount needed for criticality for the amount of binder being added. The safety of the blender could not be determined, however, if the amount of binder needed for production purposes exceeds the total safe amount that corresponds to the mass in the blender.

2. Will a system remain subcritical if fissile fuel is removed from a system but the total amount of moderating material remains constant? Again, the answer is "yes" if the total amount of moderating material is less than the amount required for criticality at a specified fissile material content.

This method of controlling moderation at NUMEC has the advantage of (a) being more easily understood by operators than does the terminology of H/X, and (b) giving operating groups greater flexibility when adding moderating material to fissile fuel.

Fig. 1.

TABLE I

Mass Limits for Oxide Fuels

Material Form	Mass Limit (kg Fissile Fuel)	Water Equivalent Limit (liters)
Pu Oxide Powder ($\rho \leq 5.6$ g oxide/cm ³)	11.0	0.500
Pu Oxide Powder ($\rho \leq 5.6$ g oxide/cm ³)	5.0	1.55
Mixed Oxide Powder ^a ($\rho \leq 5.7$ g oxide/cm ³)	25.0	2.3
Mixed Oxide Slurries, ^a pellets, or pressed powders ($\rho > 5.7$ g oxide/cm ³)	11.3	1.8

^aMixed oxides are mixtures of $\text{PuO}_2 + \text{U}(\text{nat})\text{O}_2 + \text{H}_2\text{O}$ or $\text{PuO}_2 + \text{U}(\text{dep})\text{O}_2 + \text{H}_2\text{O}$. They have a maximum of 31 wt% PuO_2 .

1. C. L. BROWN, "Impact of Criticality Safety on the Fabrication of LMFBR Fuel", BNWL-1150, pp. 5.21 through 5.33, Battelle-Northwest Labs.

IF-300 Spent Reactor Fuel Shipping Cask, Edward Walker (GE-Pleasanton)

The General Electric Company has designed, constructed, and had licensed by the AEC as a Fissile Class I, a shipping cask for the transport of either 7 pressurized-water-reactor (PWR) or 18 boiling-water-reactor (BWR) spent-fuel elements.¹ The cask is constructed using depleted uranium as the gamma shield and boron-carbide (B_4C)-filled stainless-steel tubes permanently affixed to the fuel baskets to achieve criticality control. A quarter-section view of the cask in the PWR loading configuration is shown in Fig. 1. This paper describes the criticality safety analysis of this cask.

The licensing strategy for this cask was to obtain approval for shipping a wide range of fuel element designs rather than obtaining limits based on specific fuel element designs. In addition to performing the calculations to support this approach, other factors considered in the criticality evaluations of the cask were the significance of the ^{235}U in the shield and effect of temperature coefficient of reactivity.

To analyze the criticality safety of the shipping cask, based on a generic PWR or BWR fuel bundle rather than specific designs, it was necessary to reduce the design variables of these types to as few as possible. The first parameter chosen was a maximum fuel enrichment that included existing designs. For this study, the maximum BWR enrichment was taken as 3.5% and the PWR as 4.0%. Cladding material was taken as typical for each fuel type as 0.030 Zircaloy for BWR design and 0.026 Zircaloy for PWR design. An average fuel pin diameter was selected for each fuel bundle type and then varied in dimension, increasing and decreasing, to encompass the current reactor fuel pin designs. Finally, for each fuel pin diameter, the pitch was varied within the confines of the fuel basket. The maximum pitch would bring the maximum fuel pin diameter array to within 0.050 in. of the sur-

Fig. 1. PWR-cask.

rounding basket and the minimum pitch would bring the same array of pins to within 0.050 in. of contact with one another. The results of this series of calculations are shown on Fig. 2.

The calculational method used was to analyze individual components of the cask system to obtain representative three-energy-group cross-section sets. The entire

cask was analyzed using a two-dimensional diffusion-theory computer code (TWOD).² The problem was modeled in X-Y geometry. The B₄C rods have a cladding i.d. of 0.460 in. This results in an X-Y dimension for the TWOD setup of 1.0275 cm, thus conserving total absorption rate.

The B₄C cross sections were obtained by modeling the pin positions relative to the fuel bundles as they are in the cask. It was important to reproduce the neutron spectrum as near as possible to the actual design in obtaining proper cross sections. It was found during this study that spectral differences could result in B₄C cross-section sets differing by 10 to 50%. The ability of the techniques used here to obtain cross-section sets, when properly employed, has been demonstrated by comparison to "poisoned" critical experiments to within 5%.

The sensitivity of the cask k_{eff} calculations to ²³⁵U content in the shield material was made deleting this isotope in a calculation of the PWR configuration. The effect of this was to reduce the reactivity by 0.4%. The effect of replacing the uranium shield region with water was to increase the system reactivity by 0.2%, which should be interpreted as no change at all. This apparent neutronic decoupling of the heavy-metal reflector and the core is apparently due to 1 to 6 in. of water between the bundles and the cask cavity wall.

For both the PWR and the BWR configurations, the overall temperature coefficient of reactivity was determined to be negative. For the PWR geometry this was $\Delta k/k = 3.7 \times 10^{-4}/^{\circ}\text{C}$, and for the BWR geometry $\Delta k/k = 4.0 \times 10^{-4}/^{\circ}\text{C}$.

The most reactive cases for both PWR and BWR fuel loads in the IF-300 cask have been analyzed assuming an infinite array. In both cases, the calculations show the difference in k_{eff} between one cask and an infinite array of casks is 0.005. Based on this, the cask has been assigned a Fissile Class I classification.

Fig. 2. Cask multiplication versus rod pitch and diameter.

1. "IF-300 Shipping Cask—Design & Analysis Report," NEDO-10084, General Electric Company (Feb. 1973).
2. T. F. THOMPSON "TWOD—A Program for the Solution of the Few-Group Neutron Diffusion Equations in Two Space Dimensions," Users Manual, General Electric Company, NED (Jan. 10, 1969).

Minimum Critical Masses of Water-Flooded and -Reflected Arrays, D. C. Hunt, C. L. Schuske (Dow-Colo)

Several fissile material handling situations commonly arise in which the possibility of water flooding must be considered in criticality evaluations. Examples are vault storage or shipment in storage containers where flooding could occur, autoclaving of elements in a water environment, or dissolution of elements such as machine turnings in batch operations. Previous critical measurements¹ and empirical analyses² of these data were done to determine the minimum critical masses of arrays of enriched uranium metal cubes in water. The present calculational study significantly extends these previous data to include arrays of metal or oxide rods and metal plates for both enriched (93.6% ²³⁵U) uranium and plutonium (96% ²³⁹Pu, 4% ²⁴⁰Pu) metal.

The Monte Carlo code KENO³ was used to calculate the required neutron multiplication factors (k_{eff}). The k_{eff} values typically had a standard error of ± 0.008 and involved the tracking of between 10,000 and 15,000 neutrons. Hansen-Roach 16-group cross-section sets⁴ were used with the exception of magnesium, silicon, manganese, and copper, which were obtained by XSDRN⁵ calculations, and calcium and zinc which were obtained by GAM-2⁶ calculations.

Several calculations were done to compare experimental data with the results of KENO calculations. The experimental data were of two types: (a) assemblies of plutonium metal disks interleaved with and reflected by Plexiglas,⁷ and (b) water-flooded and -reflected arrays of metal cubes or rods.¹ The agreement between computed and measured critical masses was satisfactory, ranging from a 1.6 to a 6.2% discrepancy with no detectable bias.

The minimum critical mass calculations involved arrays of plates, cubes, or square cross-section rods in water. A minimum critical array mass is found by changing the spacing of elements, d , in a particular array until a maximum value of k_{eff} is found when k_{eff} is plotted versus d . A second element size is then selected for the array and the maximum k_{eff} and the associated array element spacing found. The mass and spacing of the minimum critical mass ($k_{eff} = 1.0$) array are obtained by linear interpolation between the masses and spacings at the maximum k_{eff} values.

Some representative results appear in the figure where the minimum critical array mass is plotted vs the array element volume-to-surface ratio, v/s , for 2-ft-long rod arrays of either metal or oxide elements. An interesting

Fig. 1. Minimum critical masses of 2-ft-long rod arrays.

and important observation from these curves is that the minimum critical mass of oxide arrays always becomes less than that of metal arrays as v/s decreases. This may be explained in terms of the differences in self-shielding factors for metal and oxide rods of the same fissile mass.⁸

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2. C. L. SCHUSKE and C. L. BELL, "Empirical Analysis of Critical Mass Data Involving Enriched Uranium Metal in Water," RFP-258, Dow Chemical U.S.A., Rocky Flats Division (Mar. 21, 1962).
3. G. E. WHITESIDES and N. F. CROSS, "KENO, A Multi-group Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation, Nuclear Division (Sep. 10, 1969).
4. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos Scientific Lab. (Dec. 1960).
5. N. M. GREENE and C. W. CRAVEN, Jr., "XSDRN: A Discrete Ordinates Spectral Averaging Code," ORNL-TM-2500, Union Carbide Corporation, Oak Ridge National Lab. (July 1969).
6. G. D. JOANOU and J. S. DUDEK, "GAM-II, A B₅ Code for the Calculation of Fast-Neutron Spectra and Associated Multigroup Constants," GA-4265, General Atomic (Sep. 1963).
7. A. GOODWIN, Jr. and C. L. SCHUSKE, "Plexiglas- and Graphite-Moderated Plutonium Assemblies," *J. Nucl. Energy, A/B*, 15, 120 (1961).
8. *Reactor Handbook*, Vol. III, Part A—"Physics," Chap. 4, p. 181, 2nd ed., Interscience Publishers (1962).

Fast Reactor Safety: Mechanical Damage Safety

Energy Release from Molten-Fuel Recriticality Accidents, Thomas P. McLaughlin, Donald M. Peterson, William R. Stratton (LASL)

A postulated fast reactor accident that has been extant for many years is the collapse of the core following partial melting of the fuel. Independently of the cause or probability of such a situation, certain cases can be defined and some facets of the dynamic history of these cases can be described with more than enough accuracy for safety considerations. The situation considered is that during which a fair fraction of the fuel has become molten and has reassembled in a nearly critical state somewhere in or below the customary location of the core.

This report addresses itself to an extension of a previous study¹; the major addition being the consideration of systems with initial void fractions. Both herein and in the aforementioned work it is shown that for reactivity insertion rates postulated for such molten-fuel recriticality situations in fast reactors, the explosive energy release is very small and typically a few orders of magnitude less than the fission (thermal) energy release.

The analytical tool used in this investigation is the Los Alamos Pajarito Dynamics Code (PAD).² This code employs the coupled neutronic-hydrodynamics method in one dimension, with the neutronics provided by DTF-IV transport calculations. Without prior normalization, the PAD code has reproduced the fission energy releases of various reactor transient experiments to within a factor of 2 (Ref. 2). Furthermore, identifying the kinetic energy calculated by the PAD code with explosive energy, experimental results for this latter quantity have also been reproduced with about the same accuracy.³

The equation-of-state formulation used in this study and in the calculations of the above-mentioned experiments is of the form:

$$P = \text{maximum} \{P_S, P_V\},$$

where

$$P_S = C_1(T - T_0) + C_2(\rho - \rho_0) \quad (1)$$

and

$$P_V = C_3 F \rho T. \quad (2)$$

In these equations, $C_1 = \alpha B$, the volume coefficient of expansion multiplied by the bulk modulus; $C_2 = B/\rho$, the bulk modulus divided by the density; C_3 is evaluated at the critical point where the vapor fraction, F , equals 1.0. For molten UO_2 , the coefficient of expansion² was taken to be $1.05 \times 10^{-4} \text{ K}^{-1}$ and the bulk modulus³ was equal to $2.78 \times 10^{10} \text{ Pa}$.

Calculations were performed for unreflected spheres of molten UO_2 (~1000 kg), the void content accounted for in the equation-of-state [Eq. (1)], by setting the initial system density, $\rho = \rho_0(1 - f)$, where f is the void volume fraction of the sphere. This geometry is conservative for the calculation of kinetic energy relative to fission energy. Static DTF-IV calculations were performed to determine the critical radii of spheres with varying void fractions. The reference density, ρ_0 , was 8.7 mg/m^3 , the initial temperature, T_0 , 3123 K (assumed molten), and the initial power, 1 MW. The effective delayed-neutron frac-

TABLE I

Void Percent	Doppler Coefficient (T dk/dT)	Fission Energy (MJ)	Kinetic (Explosive) Energy (MJ)
1000 \$/sec			
0.0	0.0	121	0.5
1.0	0.0	427	6.9
2.0	0.0	625	14.0
5.0	0.0	1120	38.0
10.0	0.0	1960	95.0
10.0	-0.004	1510	43.0
300 \$/sec			
1.0	0.0	170	0.6
2.0	0.0	290	1.4
5.0	0.0	600	5.1
10.0	0.0	980	17.0
10.0	-0.004	790	6.6
100 \$/sec			
1.0	0.0	95	0.1
2.0	0.0	160	0.3
5.0	0.0	360	1.1
10.0	0.0	680	3.7
10.0	-0.004	520	1.0
30 \$/sec			
1.0	0.0	55	0.01
2.0	0.0	92	0.03
5.0	0.0	220	0.1
10.0	0.0	510	0.4
10.0	-0.004	320	0.1

tion and prompt-neutron lifetime were 0.004 and $6.7 \times 10^{-8} \text{ sec}$, respectively.

Energy releases are presented in Table I for various ramp rates, void fractions, and Doppler coefficients. For all cases, the dominant shutdown mechanism is system expansion due to condensed state pressures [Eq. (1)]. Results for the fully dense sphere (0.0 void percent, 1000 \$/sec) are given for comparison with the values listed in Ref. 1. Updated coefficients (C_1 and C_2) in Eq. (1) account for the ~30% higher energy releases in this report. It is apparent from the kinetic energy releases listed in Table I that very high reactivity insertion rates are required if an explosion is to be created in molten UO_2 systems having substantial void fractions.

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Application of Neutron Poisons In Nuclear Criticality Safety

Use of Borosilicate-Glass Raschig Rings in a Hostile Environment, *E. O. Nurmi (B&W)*

Borosilicate-glass Raschig rings purchased to ANSI N16.4-1971 standard are used as neutron absorbers in a hostile, corrosive chemical atmosphere in the NNFD uranium scrap recovery process. In the processing of highly enriched uranium, fixed neutron absorbers are required in all vessels greater than five-inch diameter to prevent the occurrence of possible critical reactions.

The chemical conditions found in the process vessels at the head end or feed preparation part of the recovery process exceed conditions outlined in ANSI-N16.4-1971. Although dissolver solutions are treated with aluminum nitrate to complex fluoride ions as well as to provide salting strength, a free acid concentration of 1 to 3 molar establishes the possibility of a highly corrosive atmosphere to borosilicate glass. Glass Raschig rings have been used for neutron absorption, or fixed poison, since 1968 in these process vessels. The vessels are filled to the overflow line with random-packed glass Raschig rings. Daily checks of the level of the rings are made and new rings are added as needed to maintain the proper level. A stringer of rings in a pipe is periodically checked to ascertain the integrity of the rings throughout the depth of the vessel. Semiannually, the tanks are completely emptied, the glass rings are cleaned, inspected, and returned to the vessel. Broken and/or thin, worn rings are discarded and replaced with new rings. The two tanks containing the most corrosive solutions have about 14,000 rings each. The monthly replacement rate is about 1% of the total rings per tank. The last cleanout required replacement of 20% of the glass rings in one tank and 10% of the glass in the other tank.

Those tanks or vessels containing solutions with little or no fluoride ions experience essentially no loss of glass rings. Any loss that occurs is the result of the semiannual cleanout and inspection which causes some Raschig ring breakage due to handling.

Samples of rings taken from the various vessels semi-annually are analyzed for boron content. No selective leaching of boron has been observed.

Borosilicate-glass Raschig rings have been successfully employed in a chemically hostile environment as neutron absorbers or fixed poisons. The hostile environment does require diligent inspection of the Raschig rings in the process units to ensure nuclear safety.

Array Poisoning, *Thomas Gutman (C-E)*

The use of 8- to 10-in.-thick concrete layers to isolate arrays of interacting subcrits of fissile material had, until recent years, been considered standard practice.¹ The use of KENO calculations, however, has shown that

concrete thickness of significantly <16 in. fails to provide isolation, and may even increase array reactivity in some cases.²

We have performed studies to identify the isolation provided by various materials for specific arrays. Although some general statements about isolating materials

ARRAY	a (INCHES)	a (COMPOSITION)	k_{eff}
1*	NA	NA	$0.9447 \pm .0073$
2a	6	CONCRETE	$0.9841 \pm .0073$
2b	8	CONCRETE	$0.9657 \pm .0067$
2c	10	CONCRETE	$0.9555 \pm .0076$
2d**	6	BORATED CONCRETE	$0.8941 \pm .0073$
2e**	3	BORATED CONCRETE	$0.9083 \pm .0073$

* ARRAY No. 1 CONSISTS OF A ONE HIGH PATTERN OF CYLINDERS WITH TOP AND BOTTOM CONCRETE REFLECTION

** BORATED CONCRETE SUPPLIED BY REACTOR EXPERIMENTS, INC.

Fig. 1. Layout and reactivities of tiered arrays of $U(3.5)O_2$ cylinders containing 2.05 g U/cm^3 .

can be made as a result of these studies, there are insufficient data to support the formulation of rules by which isolation is in all cases assured.

Two specific studies were made. In one case, the system consists of two tiers of air-spaced cylinders of low-enriched uranium, spaced on a square pattern, with isolating media between the tiers, as shown in Fig. 1.

In the second, a large array of buckets of low-enriched uranium oxide utilizes various isolating media to effect partial isolation, as shown in Fig. 2. These arrays were evaluated using the KENO code with 16-group Hansen-Roach cross sections.

The results of these calculations are presented in Figs. 1 and 2, and may be summarized as follows:

TABLE I CALCULATIONS OF k_{eff} USING KENO-II

ARRAY	SEPARATOR PARTITION	CONFIGURATION & FUEL	KENO-II $k_{eff} \pm o$
1	1/4" THICK BORAL PLATE (a)	∞ BY ∞ BY 5 ARRAY 72" e/e SPACING BETWEEN PAILS ACROSS AISLE 5.0 WT% ^{235}U ENRICHMENT (c)	1.0942 $\pm$.0072
2	2" THICK ORDINARY CONCRETE LINED WITH 1/4" THICK BORAL PLATE ON THREE SIDES OF U-SHAPED CUBICLES (SIDES CLOSEST TO STORAGE PAILS)	∞ BY ∞ BY 5 ARRAY 72" e/e SPACING BETWEEN PAILS ACROSS AISLE 5.0 WT% ^{235}U ENRICHMENT	1.0054 $\pm$.0090
3	10" THICK ORDINARY CONCRETE LINES WITH 1/4" THICK CARBON STEEL ON THREE SIDES OF U-SHAPED CUBICLES (b)	∞ BY ∞ BY 5 ARRAY 96" e/e SPACING BETWEEN PAILS ACROSS AISLE 5.0 WT% ^{235}U ENRICHMENT	0.9823 $\pm$.0078
4	10" THICK ORDINARY CONCRETE	∞ BY ∞ BY 5 ARRAY 96" e/e SPACING BETWEEN PAILS ACROSS AISLE 4.0 WT% ^{235}U ENRICHMENT	0.9930 $\pm$.0091
5	10" THICK ORDINARY CONCRETE LINES WITH 1/4" THICK CARBON STEEL ON THREE SIDES OF U-SHAPED CUBICLES	∞ BY ∞ BY 5 ARRAY 96" e/e SPACING BETWEEN PAILS ACROSS AISLE 4.0 WT% ^{235}U ENRICHMENT	0.9449 $\pm$.0080

(a) BORAL PLATE CONSISTS OF 0.1689" THICK B_4C IN ALUMINUM (35 WT% B_4C NATURAL C, 65 WT% A1; 0.0949 LB/IN³) CLAD ON BOTH SIDES BY 0.041" THICK ALUMINUM. TOTAL THICKNESS IS A 0.2509" SANDWICH.

(b) CARBON STEEL CONSISTS OF 99 1/4 WT% Fe, 3.4 WT% C; 7.81 g/cm³.

(c) 2.0 gm U/cm³ FOR ALL CASES.

Fig. 2. Representation and reactivities of concrete-block storage arrays.

1. Significant neutron transmission occurs through 10 in. of concrete.
2. Neutron transmission is significantly reduced by 3 in. of borated concrete.
3. Significant reduction of neutron transmission can be effected by lining 10-in.-thick concrete with $\frac{1}{4}$ -in.-thick steel plates.

As isolating media are generally utilized to effect a reduction in the reactivity of the system in question, rather than an arbitrary diminution of neutron flux through the media, specific calculations for various systems may demonstrate this goal using relatively economic combinations of materials.

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Criticality of Heterogeneous Systems— UO_2 - PuO_2 Solids in Fissile Solution Containing Gadolinium Nitrate, R. C. Lloyd, E. D. Clayton (BNW)

The application of neutron absorbers for criticality prevention in fuel cycle operations can permit the safe handling of large quantities of material with reduced probability of criticality. If, however, soluble poisons are to be considered as either a primary or secondary means of criticality control, their use must be based on a firm knowledge of the effects of the absorber.

To provide confirmatory data, a series of criticality experiments was performed to determine the effectiveness of soluble poison (gadolinium nitrate) on criticality of a heterogeneous assembly of FFTF-type fuel pins in Pu-U nitrate solution. Gadolinium was chosen as a neutron poison because of its high neutron cross section, high solubility, and compatibility in the separation process. The arrangement simulates the simplest approach to a possible dissolver configuration, wherein during the dissolution process, a heterogeneous system of solids in fissile solution (the partly dissolved fuel) would prevail.

The fuel assembly was composed of a lattice of 301 fuel pins positioned in a 55.5-cm-i.d. stainless-steel vessel, as shown schematically in Fig. 1. This vessel, in turn, was contained within a larger cylindrical vessel providing for water reflection as shown. The U-Pu nitrate solution, containing varying amounts of gadolinium, was added to the loaded lattice assembly to obtain criticality through variation on solution height. The gadolinium was added to the solution in the form of GdO_3 . The free acid of the solution readily formed GdNO_3 when mixed by circulation with a pump.

The experimental data, together with the fuel pin description and chemical makeup of the solutions, are included in Table I. The gadolinium proved to be very effective in this system. In the absence of the gadolinium poison the lattice-solution assembly became critical at a solution height of only 19.2 cm, whereas with 1.3 g Gd/liter the critical solution height was increased to 68.9 cm. The H/Pu ratio for the fuel-pin solution combination was ~ 86 .

The results of theory-experiment comparisons utilizing ENDF/B cross sections and the KENO Monte Carlo code will be presented and discussed.

Fig. 1. UO_2 - PuO_2 solids in fissile solution containing gadolinium.

TABLE I
Preliminary

Criticality of Pins in Gadolinium-Poisoned
Fissile Solution

Experiment Number	Critical Height (cm)	Gadolinium (g/liter)
115	19.205	0.0
116	23.066	0.25
117	28.227	0.5
119	45.753	1.0
120	64.506	1.25
123	68.862	1.3
Chemical Composition of Solution		
Pu ~77.6 g/liter	H ⁺ 3.25 g/liter	
U ~181.4 g/liter	NO ₃ 374.00 g/liter	
Sp gr 1.4685		
Isotopic Composition of Pu and U in Solution (wt%)		
Plutonium		Uranium
239 93.840	242 0.058	235 0.659
240 5.710	238 0.016	236 0.012
241 0.376		238 99.329

Fuel-Pin Description

Fuel-Pin Dimensions		
	o.d. (cm)	Length (cm)
Fuel column	0.495	69.22
Cladding (316-SS)	0.584	72.90
Fuel Per Pin		
PuO ₂ -U(nat)O ₂ 138.4 ± 1.3 g		
Pu 30.75 ± 0.03 g	U 91.16 ± 1.03 g	O 16.49 ± 0.17 g
Fuel Enrichment		
25.2 wt% Pu		
Fuel Density		
10.35 ± 0.09 g/cm ³ (93.34 ± 0.79% theoretical)		
Isotopic Composition of Pu in Pins		
²³⁸ Pu 0.04 ± 0.01 at.%	²⁴¹ Pu 1.73 ± 0.01 at.%	
²³⁹ Pu 86.19 ± 0.06 at.%	²⁴² Pu 0.16 ± 0.01 at.%	
²⁴⁰ Pu 11.88 ± 0.06 at.%		

Soluble Poisons and Chemical Processing
Equipment at ICPP, J. A. Eggert (Allied
Chem-Idaho)

Three headend processes that use soluble poison as a primary control to avoid a criticality have been developed at the Idaho Chemical Processing Plant. These processes are: zirconium-alloyed fuel dissolution, stainless-steel-clad fuel dissolution, and Rover fuel dissolution. The equipment was evaluated for nuclear criticality safety at flow-sheet concentrations of nuclear poison using the KENO-II code¹ and 16-group Hansen and Roach cross sections.² Validation of code and cross sections by comparison with critical experiments involving soluble

poisons³⁻⁴ has shown a negligible bias in the calculational models. The equipment is shown in Fig. 1 and a brief description follows.

Zirconium-Alloyed Fuel Dissolution

The fuel is typical of the early cores from the PWR at Shippingport, Pennsylvania—~3% U(93) and 97% zirconium. Kilogram quantities of uranium are charged to a nongeometrically safe dissolver⁵ containing borated charging water at 5.0 g B/liter. Dissolvent, consisting of 6.8 M HF and 4.1 g B/liter, is added to the dissolver charge to obtain a dissolution rate of up to 20 kg U/day.

The criticality safety of the dissolver was evaluated by establishing the most reactive metal volume fraction (MVf) of the fuel material as a function of boron concentration. UF₄- $\frac{3}{4}$ H₂O plateout, uranium precipitation, and undissolved heel were evaluated singly and in combination for effect on maximum k_{eff} . A maximum k_{eff} of 0.89 was obtained for the design basis conditions.

Stainless-Steel-Clad Fuel Dissolution

Stainless-steel-clad fuel from the Experimental Breeder Reactor-II at the Reactor Testing Station in Idaho, having material composition of 95% U(52.6) and 5% fission, is processed in an electrolytic dissolution process. The process equipment consists of a slab dissolver, a nongeometrically safe surge tank, and a heat exchanger,⁶ all of which are part of the recirculation loop for the dissolver. The dissolver consists of 9.0 M HNO₃ containing dissolved gadolinium oxide.

Safe concentrations of gadolinium were obtained by establishing a relationship between gadolinium concentration and an allowable mass limit for an arbitrarily selected $k_{eff} = 0.90$ in the nongeometrically safe surge tank. A concentration of 2.6 g Gd/liter and a mass limit of 36 kg ²³⁵U were determined for the process loop based on the surge tank evaluation.

A prolonged electrical power failure to the dissolver and loop pump could allow the concentration of uranium in the stagnant dissolver to build to 610 g U(52.6)/liter. Transfer of 610 g U(52.6) to the surge tank followed by dilution indicates a maximum k_{eff} of 0.90 would be obtained for the most reactive concentration in the surge tank, 240 g ²³⁵U/liter with 2.6 g Gd/liter present.

Rover Fuel Dissolution

Rover fuel is a highly enriched uranium-niobium-graphite material from the Nuclear Rocket Development Station in Nevada. This fuel will be recovered by a combustion-dissolution process. The combustion step will convert UC₂ and NbC to oxides and the dissolution step will dissolve the oxides in 8.0 M HF and 8.0 M HNO₃ with 3.5 g/liter boron present.⁷ The proposed annular dissolver vessel will have a center vessel and an external cooling/heating jacket filled with borated water at 5.0 g B/liter. The annulus has fuel and dissolvent with 3.5 g B/liter present.

General Considerations Common to Each Process

1. Qualitative selection of the specific process poisons was based on the solubility of the poison compound in acid dissolvent, the poison chemistry compatibility with the process steps, and the availability and cost of the poison compound.

2. All processes use a poison concentration equal to or greater than the concentrations established in the criticality safety evaluations.

Fig. 1. ICPP process equipment requiring use of soluble poisons for criticality safety control.

3. Redundant poison detection (and shutdown) instruments⁵ monitor the poison solutions entering the dissolution process.

4. A certified qualitative and quantitative chemical analysis is required before a poison solution is transferred to the dissolution system.

5. All lines that could dilute poison are physically disconnected from process equipment.

6. Flow-sheet poison concentrations are evaluated for abnormal operating occurrences.

Operating experience with soluble poisons at the ICPP demonstrates it is a safe and acceptable practice. Criticality safety evaluations are conservative and consider all credible variations in process conditions. However, use of soluble poisons requires additional controls to guarantee their presence.

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Application of Borosilicate Raschig Rings for Primary Criticality Control, C. L. Schuske, J. D. McCarthy (Dow-Colo)

This paper reviews the Rocky Flats Plant experience in the application of borosilicate-glass Raschig rings as a primary criticality control in fissile-solution handling processes. Included is a history of their introduction at Rocky Flats as well as a record of the problems encountered with their use.

The Oak Ridge critical facility made the first criticality measurements on 20-in.-diam vessels filled with borosilicate-glass rings or tubes. These measurements are reported in Ref. 1. Following these ORNL experiments, Rocky Flats personnel performed multiplication measurements on a 42-in.-diam, 250-gal tank filled with Raschig rings in April of 1959. Several uranyl nitrate solutions, ranging from 1000 to 360 g U/liter, were used in these tests. In the Spring of that year, this method of fissile-solution storage was adopted for the uranium production areas of Dow's Rocky Flats Plant.

In 1960, a similar experiment in a 30-in.-diam tank was done with 350 g/liter Pu in the form of plutonium

nitrate. Following this test, the method was also adopted for the plutonium recovery plant.

Since that date, much progress has been made and valuable lessons learned in the use of Raschig rings in the storage of both plutonium and uranium.

Some of the problem areas with Raschig rings over the past 15 yr are: (a) preferential leaching of boron from the rings by boiling nitric acid solutions; (b) chipping and cracking of rings leading to pump and pumping problems; (c) silicious deposits upon the rings in uranium solution; (d) plutonium oxide sludge deposits (see Fig. 1) on the rings; (e) difficulty in cleaning rings with plutonium deposits; (f) incomplete filling of domed tanks; and (g) ring disposal.

Solutions to these problems included the following. Reducing the B_2O_3 content from 19 wt% to 11.8 to 13.8 wt% eliminated the preferential leaching of boron from the rings. Fire polishing and tempering the rings, as well as increasing the wall thickness of the rings, almost completely eliminated the broken-glass problem. Silicious deposits observed for uranium solutions were easily cleaned from the rings by a combination of household detergents and low-pressure steam. Plutonium deposits containing PuO_2 required the *in situ* introduction of boiling nitric acid and calcium fluoride for removal of much of the sludge. High-fired oxides, in some cases, must be

removed from the rings in small batches by washing. Since the plutonium sludge problem originates in dissolution processes, these feed streams are prefiltered with 3- μ m filters. Gamma scanning of the tanks, as well as improved calibration techniques, are leading to earlier detection of the buildup. A double-check system has been instituted to minimize the possibility of incompletely filling the process tanks with Raschig rings. The supervisor in charge of the area involved is required to certify, by his signature and date, that the tank has been filled. Records on all tanks are maintained for future information and available for both AEC and Company audit teams. We feel that the problem of incomplete filling can be minimized by designing tanks specifically for Raschig ring use. In the past, most conventional vessels were filled through upper ports which did not provide good visual access to the filling process. This problem was solved by using removable flat tops on the tanks.² In addition, capacitance probes located some distance below the top of the tank can act as shutoff mechanisms of incoming feed solutions, to preclude overfilling. These features have been incorporated into the tanks, shown in Fig. 2.

Two documents have been developed at Rocky Flats to provide guidance in the use of Raschig rings. The first is SX-106D,³ "Standard for Borosilicate Glass Raschig Rings for Nuclear Poison," which was derived from ANSI

Fig. 1.

Fig. 2.

N16.4-1971 (Ref. 4). This document contains ring specifications as well as test specifications on these rings. Due to the corrosive nature of process solutions at Rocky Flats, we do not specify narrow chemical limits on the nature of the solution, but rely on frequent ring inspections to preclude a nuclear safety problem. The second document, "Standard Safety Practice #345" (SSP-345) (Ref. 5), delineates responsibilities for purchasing, sampling, labeling, quality control, and testing of rings by the Rocky Flats Analytical Laboratories. This document also spells out the responsibilities of the users of these rings.

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The Criticality of Periodically Boron-Poisoned Enriched-Uranium-Solution Systems, Robert E. Rothe (Dow-Colo)

Criticality measurements were made on a large, unreflected uranium-solution system poisoned with evenly spaced boron-loaded stainless-steel plates. Critical solution heights were measured as a function of the average spacing between plates within a large tank.^{1,2} Experimental results were compared with KENO-II³ calculations whose material and geometry descriptions simulated experimental conditions accurately. Sixteen-energy-group cross-section sets⁴⁻⁶ were used.

Three uranyl nitrate solutions (93.2% ^{235}U) covering a wide range of concentrations were studied in a 106.6-cm-i.d. tank. The tank was 150 cm tall with 0.63- and 0.95-cm-thick walls and bottom, respectively. The poison plates contained 1.02 wt% natural boron, had a density of 7.9 g/cm³, and measured 119.4 cm high $\times$ 0.259 cm thick. Additional details on both experimental and calculational aspects of this program are given in Refs. 1 and 2.

The theory/experiment comparison is presented in Fig. 1 at the three experimental concentrations. The calculational method consistently predicts neutron reproduction factors less than unity for experimentally critical systems. Based on this comparison, this calculational method may be used for nuclear safety evalua-

Fig. 1. KENO-II calculations (solid curve) are shown for accurately simulated experimental conditions, but the vertical asymptotes (dashed) were calculated for three-dimensional infinite systems using the DTF-IV⁷ code. The points near 100-cm spacing are the measured critical unpoisoned solution slab thicknesses. The upper arrow corresponds to the top of the plates; the lower arrow marks the equilateral cylinder case.

tions of appropriate plant problems provided a suitable allowance is made for the apparent bias between calculations and experiment. For such evaluations, Fig. 1 may be divided into two regions by a horizontal line at $H = 24$ cm. Above ($H > 24$ cm), calculated critical plate spacings should be reduced 15% and, below, the bias requires a reduction in calculated critical heights of 17%. These are conservative reductions and Fig. 1 can be used directly to permit a smaller reduction in many cases with equal safety.

For sufficiently small plate spacings, criticality did not occur as solution reached the top of the plates. The system was driven critical, however, by the formation of an unpoisoned solution slab immediately above the highly poisoned region. The critical thickness of this slab is shown in Fig. 2 for the three experimental concentrations. Again, calculational methods consistently predict critical heights in excess of measured values. All three calculated curves terminate at a plate spacing of 0.259 cm because that corresponds to a solid boron stainless-steel reflector.

Both borosilicate-glass Raschig rings and poison plates have industrial applications. Rings are frequently used to poison large tanks of fissile solution. Parallel plates have been used to poison large-diameter ion columns, allowing easy loading and removal of the ion-exchange resin. The effectiveness of each as a nuclear poison was compared, using the experimental ring measurements of Thomas⁷ and the present data. The conclusion is that both heterogeneous poisons are less "efficient" than a homogeneous distribution, but neither is clearly superior with respect to efficiency. Consequently, ring-poisoned systems may be evaluated for nuclear safety using Fig. 1, provided a 2% increase in the

Fig. 2. The identifying letters and legend are the same as Fig. 1. The last experimental point to the left ($P + S = 0.953$ cm) corresponds to all available plates in position. The arrows mark plate spacings below which the critical unpoisoned slab thickness is smaller with a thick water reflector than with a "reflector" composed of highly poisoned fissile solution (see Refs. 2 and 8).

overall boron density is made to account for uncertainties in the comparison.

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The Criticality Accident

Empirical Model to Estimate Energy Release from Accidental Criticality, A. R. Olsen, R. L. Hooper (BNW), V. O. Uotinen (B&W), C. L. Brown (BNW)

Each new plant that fabricates, processes, or otherwise handles fissionable materials undergoes a safety analysis prior to startup. One part of the safety analysis is to assess the potential consequences of a postulated worst credible criticality accident. To perform this assessment, the total energy release—expressed as “total number of fissions” from the criticality excursion—must be estimated or assumed. This paper presents the results of study for development of an empirical model to estimate energy release from a criticality accident.

In a plant where the assumed worst credible criticality accident is in a fissile solution system, the present industry practice is to assume a preestablished upper limit of 4×10^{19} fissions. This value is based somewhat on the past history of criticality accidents and the fact that the highest excursion to date resulted in an estimated 4×10^{19} fissions.

Recently, a summary of the results of a series of criticality excursion experiments with highly enriched uranium solutions conducted in France by the Commissariat à l'Énergie Atomique (referred to as the CRAC experiments) was published.¹ These experiments have provided the first firm basis for developing an empirical model for predicting the total energy release from a criticality excursion in a given solution system.

The criticality accident, as characterized by the model empirically derived from the experimental data provided by the CRAC experiments, is considered to be divided into (a) an initial fission burst followed by (b) a plateau period where the number of fission/sec, ignoring oscillations, decreases with increasing time in the plateau.

The model for the initial burst relates the total fissions in the burst to a function of the volume (liters) of solution at the time of the burst. The empirically derived equation is

$$F_B = 2.95(10)^{15} V_B^{0.82}$$

The estimated 95% upper confidence level is given by

$$\log(F_B^U) = 15.47 + 0.82 \log V_B^0 + 0.23 \left[1.04 + \frac{(\log V_B^0 - 1.73)^2}{4.07} \right]^{1/2}$$

Figure 1 presents the relationship. This is assumed to hold for the vessel diameter range (300 to 800 mm) and solution addition rates used in the CRAC experiments.

The model developed for predicting the number of fissions in the initial burst is mainly applicable to highly enriched uranium systems, since this was the material used in the CRAC experiments. However, this model is also applicable to plutonium and slightly enriched uranium systems, though the energy release predicted will be conservatively high because of the presence of ^{240}Pu and ^{238}U , respectively, and other isotopes that undergo

spontaneous fission. Considering the CRAC experiments with a neutron source added, we estimate that for plutonium systems, the fissions in the initial burst could be a factor of 2 or more lower than predicted by the model. A reduction factor has not been estimated for slightly enriched uranium.

The model for the number of fissions in the plateau is given by

$$F_p = 3.2(10)^{18} [1 - t^{-0.15}]$$

with the 95% upper confidence level given by

$$F_p^U = 4.6 \times 10^{18} [t_1^{0.02} - 1]$$

where t is the duration of the plateau in seconds and F_p is the number of fissions in the plateau. The model is developed from an empirically derived upper-envelope fission/sec plateau time plot and then integrated over the duration of the plateau (t). This relationship is presented in Fig. 2.

Thus, an estimate of the total number of fissions occurring during a criticality accident is obtained from $F_T = F_B + F_p$, the latter term generally being the main contributor.

Fig. 1. Least-squares estimated relationship between the number of fissions in initial burst and the volume at pulse peak. (Based on CRAC experiments without external neutron sources. Upper curve gives the 95% upper confidence bound.)

Fig. 2. Estimated total number of fissions occurring in plateau region of an excursion as a function of the duration of plateau. (The 95% upper confidence bound is presented as the upper curve.)

TABLE I
Comparison of Total Energy Release Predicted by Model and Actual
Energy Release of Past Criticality Accidents

Characteristics of Fissile Material System				Duration of Excursion (min)	Number of Fissions (x 10 ¹⁷)			
Location	Form	Mass, kg Fissile	Volume, liter		Actual Excursion		Predicted By BNW Model	
					Initial Burst	Total	Initial Burst	Total
Y12	UO ₂ (NO ₃) ₂ ^(a)	2.5	56	13	~ 0.1	13.0	0.8	21.0
LASL	Pu/Organic	3.3	168	< 1	1.5	1.5	2.0	16.7
IF	UO ₂ (NO ₃) ₂ ^(a)	34.5	800	20	1.0	400.	7.1	28.1
IF	UO ₂ (NO ₃) ₂ ^(a)	8.0	40	< 1	~ 0.6	6.0	0.6	15.3
Hanford	Pu Complex	1.5	~ 60	2220	~ 0.1	8.0	0.9	27.4
Wood River	UO ₂ (NO ₃) ₂ ^(a)	2.6	~ 70	< 1	1.1	1.3	1.0	15.7
Windscale	Pu/Organic	2.5	~ 100	< 1	0.01	0.01	1.3	16.0
ORNL	Pu(NO ₃) ₄	1.15	64	< 1	~ 0.8	0.8	0.9	15.6
ORNL	UO ₂ F ₂	18.3	55	< 1	0.5	0.5	0.8	15.5
ORNL	²³³ UO ₂ (NO ₃) ₂	~ 1.0	5.8	< 1	0.11	0.11	0.1	14.8

^aUranium enrichment ~ 93 wt% ^{235}U .

Application of the model to find the potential energy release from a criticality accident for use in a plant safety analysis could be accomplished by listing the vessels to be considered, estimating a potential duration

for the excursion based on past accidents, calculating for each vessel the total number of fissions using the model presented, and then selecting the highest number of fissions.

To test the model, the conditions of the seven past industrial criticality accidents² and three critical-mass laboratory accidents were used to calculate initial burst and total energy release. Results are presented in Table I. The model is conservative in all but one instance.

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Accidental Fissile Solution Excursions and Building Design Criteria, G. Tuck (Dow-Colo)

In the past few years, the AEC has required solution excursion analyses for new construction as well as plants already in operation. These hypothetical excursions should be chosen to be representative of the most severe excursion that could be reasonably expected in the particular systems involved since the results of the analysis will, in turn, determine the size and types of the hot exhaust filtering system and shielding thickness, etc. An example of the "most severe" excursion may be helpful. Consider a tank 3 ft in diameter by 4 ft high which is shown by a safety review to have solution. For the excursion parameter of maximum total fissions, the most severe condition is a tank that is critical when nearly full; hence, the full tank volume is used for the total fission estimate. The specifications of the process systems are usually such that the excursions chosen are limited by practical design considerations to tanks of 2- to 5-ft diam with a height of 1 to 2 diam and a solution transfer rate of $7\frac{1}{2}$ gpm or less.

All of the available solution excursion data are by no means directly applicable to excursion analysis. The data from accidental solution excursions have some limitations for our purposes. Of the five excursions in critical-mass laboratories, all involved volumes of <65 liter, three were in spherical geometry, and three involved the movement of control or scram rods. None of these characteristics is likely to be chosen for a reasonable most severe process plant excursion although the 65-liter volume is perhaps typical of the volumes involved in the smaller plant-type excursions.

All seven of the industrial excursions provide useful data. Two are representative of particular types: annular geometry and concentration buildup. Of the remaining, two involved volumes of <65 liter, two more for which the volumes are not given in the published excursion tables,¹ and one in an 800-liter volume. Summarizing the accidental solution excursion data, there appear to be two special cases, two typical smaller excursions at volumes of <65 liter but only one at the more reasonable process-area tank volume of 800 liter.

The French CRAC² data are directly applicable to the usual uranium solution process plant situations with respect to variables of the range of fill rates and solution concentrations. The data available are applicable for the first burst of a solution excursion for tanks of about 1-ft diam and for $2\frac{1}{2}$ -ft diam up to an 18-in. height. Unfortunately, there are no data above 18 in. for the $2\frac{1}{2}$ -ft-diam tank nor for the larger tanks which are more typical of the Rocky Flats areas. Of those experiments that lasted longer than 30 min, only one had much

reactivity added after critical. Also, to use these data for plutonium solutions, the difference in neutron lifetimes, reactivity addition rates, neutron sources, and shutdown coefficient must be accounted for. The CRAC program was an extremely worthwhile project and these comments are intended only to point out the limitations of the CRAC data for plant accident-analysis purposes.

To extend the experimental excursion data to the range of tank sizes and to plutonium solutions typical of the Rocky Flats process areas, the CRAC data together with some of the KEWB³ and industrial accident data were analyzed. This involved writing a simple excursion code utilizing the reactivity addition rate, the neutron lifetime, and a shutdown coefficient; deriving methods of obtaining these inputs; one series of calculations to validate the code and inputs with existing data; and another long series of calculations to evaluate the range of hypothetical plant excursions. It was found that for each tank diameter and size, there was a particular situation that produced the most severe case for each of the excursion parameters. These most severe cases were expressible in terms of simple equations and are given in Ref. 4. With these equations, one can start with the variables of tank diameter, height, and fill rate and easily estimate the excursion results needed for most building design criteria. It should be emphasized that the results obtained with these equations are not for a typical excursion but for a reasonable "most severe" excursion for the system involved.

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Analytical Studies of Criticality Accidents in Aqueous Solutions, H. M. Forehand, Jr. (ANL-Idaho)

A simple mathematical model has been constructed for simulating nuclear criticality accidents involving aqueous solutions.

The effect of radiolytic gas was demonstrated in the kinetics experiments on water boiler (KEWB)¹ program, a study designed to define the safety characteristics and dynamic behavior of the "water boiler" class of reactors. Radiolytic gas was observed to be the dominant shutdown mechanism for excursions with inverse periods above 100 sec^{-1} .

In late 1968, the Service d'Etudes de Criticité of the French Commissariat à l'Energie Atomique initiated a program of systematic experimental nuclear excursions initiated intentionally to obtain realistic criticality accident data. This program was designated Conséquence Radiologiques d'un Accident de Criticité (CRAC).²⁻⁵ Uranyl nitrate solutions of various concentrations were pumped at a steady flow rate into vertical cylinders (30- and 80-cm diam), simulating the accidental assembly

Fig. 1. Fission rate and integrated energy release in CRAC 08 as a function of time.

of a supercritical volume. A typical excursion obtained during the CRAC experiments is presented in Fig. 1.

The energy deposited in the fissile solution of an aqueous assembly is the sum of the energies contributed by the radiation environs. The majority of the energy is deposited by the fission fragments. The main products resulting from the presence of the radiations are H_2 and O_2 gases and peroxide. These gases can be removed from the liquid by diffusion through the air-liquid interface and by bubble formation. In addition, the gas bubbles can migrate to the air-liquid interface and leave the system.

Gas microbubbles are nucleated in the fissile solution by a 'localized thermal spike' generated by a fission fragment. Initially, the bubble contains a mixture of gas and water vapor. Below the boiling point the vapor condenses quickly, leaving a gas microbubble. Unless the solution is supersaturated, the gas bubble will collapse in $\sim 10 \mu\text{sec}$ (Ref. 6). In a supersaturated solution the bubble will grow and produce negative feedback by increasing the neutron leakage.

A simple mathematical model (derived from the KEWB models)⁷ in which the dissolved gas concentration is assumed to be proportional to the energy and the nucleation rate proportional to the power, is

$$\rho_f(t) = \phi V(t)$$

$$\frac{dV}{dt} = Cn(t) \left[\int_0^t n(\tau) d\tau - E_0 \right] \quad \text{for } \int_0^t n(\tau) d\tau \geq E_0$$

$$= 0 \quad \text{for } \int_0^t n(\tau) d\tau < E_0,$$

where

$\rho_f(t)$ = radiolytic gas feedback

ϕ = void volume reactivity coefficient

Fig. 2. Peak power vs maximum inverse period for ANCON and PAD calculations and CRAC 300 experimental data.

$V(t)$ = total void volume

E_0 = saturation threshold energy

$n(t)$ = fission power.

This model has been incorporated into the ANCON⁸ and PAD⁹ codes. The model contains no information about pressures or mechanical damage potential. It fails shortly after peak power in the first burst because it cannot predict reassembly. It is successful, however, in correlating peak power with maximum inverse period for the first burst, as shown in Fig. 2.

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Available Resources for the Design and Safety Analysis of Processing Plants, Robert L. Seale, David L. Hetrick (U of Ariz)

In the design of nuclear processing equipment, there should be an analytical basis on which to evaluate the detailed consequences of faults in plant design or operation which have a very low probability of occurrence. The need for criticality data to serve as the basis for the design of fuel processing equipment and systems has long been recognized. Thus, static critical measurements have been made and documented over the years for a wide range of fissile material, moderator, and reflector combinations.¹ Experience has shown the guidelines to be useful and reliable, if somewhat conservative, in the design of process equipment. The value of these data in system design can be appreciated by noting that a common feature of the criticality incidents to date is that their occurrence is invariably the result of the use of unapproved equipment or the processing of incompletely inventoried solutions.² Paxton³ has aptly observed that among the key ingredients in a safe processing operation is that design should be based on the concept that "the correct way should be the easy way" (and vice versa), and that plant management must be responsible, informed, and alert to the details of their operation.

There is, consistent with the rest of the industry, a growing need in processing fissile materials for the capability in design to evaluate the consequences of various potential accidents arising from plant faults or external causes and to assign quantitative numbers to the probability of their occurrence. Energy release, radiation exposure, pressures, temperatures, etc., are invaluable in the design of equipment and personnel protection systems. These data are also useful in establishing the need for containment and in evaluating potential threat to the health and safety of the public. To provide realistic data on such accident consequences, dynamic or transient experiments must be carried out with appropriate fissile process materials. The need was long recognized but only timidly considered in this country. In 1968, the French Commissariat à l'Energie Atomique started the now famous series of CRAC experiments.^{4,5} These experiments have provided a wealth of data on criticality transients and have been used as the basis for empirical correlations on total energy release, doses, etc.

More significantly, however, these experiments have provided the data needed to verify the applicability of various excursion codes to the analysis of criticality transients in processing situations. The basic physical properties needed for such calculations include the equation-of-state of the fissile material and an understanding of the relevant shutdown mechanisms dominant for the material in question.

It would seem that the time has come to identify other significant process forms for fissile material besides the aqueous solution and to do the limited number of transient experiments needed to characterize their accident potential. Certainly a check of the equation-of-state and the identification of shutdown mechanisms would be needed to make these other potential accidents grist for the excursion evaluation calculation mill.

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Recovery and Restoration Phases of a Criticality Accident, D. F. Cronin (UNC)

The recovery phase of a criticality (or any other) accident is defined as the period when control is established over the situation and all actions are planned, organized, and directed toward recovery from the accident. The restoration phase implements these actions, and is directed toward return to normal operation.

The accidental criticality occurred Friday, July 24, 1964 in a process vessel at the Fuels Recovery Plant of United Nuclear Corporation's Fuels Division, located at Wood River Junction, Rhode Island. Uranium enriched to ~93% in the ²³⁵U isotope, as a nitrate solution, was

manually transferred from a 5-in.-diam plastic storage cylinder to the 12-in.-diam, 30-gal capacity process vessel. The result was 1.3×10^{17} fissions, and sufficient radiation for a lethal exposure to the operator making the prohibited transfer. Five other persons also received exposures in excess of 1.25 rem during the accident and subsequent reentry procedures.

The existing plant emergency plan went into effect immediately with the sounding of the automatic siren alarm. Plant evacuation, head count, and confirmation that a criticality accident had occurred were accomplished within less than five minutes. Within the next few minutes, as provided in the emergency plan, an ambulance was summoned for the injured employee and notification was made to the Plant Superintendent, other Company officials, and the USAEC. Within the first hour, the Plant Superintendent had arrived on site. The ambulance had been redirected by radio to Rhode Island Hospital in Providence when the local hospital indicated it could not handle the case because of a lack of facilities for treating radiation injuries.

The Recovery phase began approximately at $T + 2\frac{1}{2}$ h after medical care had been provided, and after the uranium solution had been drained from the unsafe geometry into 1-gal bottles, which were then carefully spaced apart in line. The Divisional Manager of Chemical Operations was the responsible executive on site. The Division Vice President and technical staff from other divisions were en route to the plant. The AEC radiological emergency team and representatives of the AEC Division of Compliance from New York, and from the AEC Licensing and Regulatory Division in Washington, D.C. had informed UNC they would arrive on site as soon as possible.

Control had been established with no further hazard likely, although a state of emergency was still considered to exist. Office areas immediately adjacent (40 ft) to the process area were cleared ($T + 1$ day) by health physics for occupancy. Contamination was confined to the process area by uranium and fission products with maximum penetrating radiation levels of 50 R. There was no evidence that uranium or fission products had been deposited off site. The radioactive materials were contained primarily within the plant building and were concentrated in the immediate vicinity of the reacting vessel (the solvent extraction area).

Under the personal direction of the Division Vice President, all of the corporation's resources were placed at the disposal of a Company Investigation Task Force, and of the special "Go-No-Go Committee" established for final approval of any proposed action concerning plant startup. Three major areas were defined: Technical nuclear aspects were assigned to the Manager of Research, Research and Development Division; controls and written procedures assigned to the Quality Control Manager, Fuels Division, who was also Task Force Chairman; decontamination, physical inventory of total plant and material, and public information assigned to the Manager of Chemical Operations, Fuels Division. Control and continuity of information released to the press and the public by UNC was at all times channeled through a single person acting as the corporate representative; the first press release was issued approximately four hours after the incident; permission was later given the *Evening Bulletin*, Providence, Rhode Island, to visit the plant, gather information, and take photographs for a feature story. These strong positive actions taken by United Nuclear and the strict control of all phases of the restart were initiated by the company. The AEC was kept fully informed, and its concurrence requested in specific

areas. Amendments were issued by the AEC to possess the fission products generated. On November 25, 1964, an amended license for resumption of normal operations was issued. A letter of concurrence for an experimental test run had been issued earlier. The UNC "Go-No-Go Committee" did not release the plant to production until December 7, 1964. A final release to the normal organization and disbandment of the "Go-No-Go Committee" was issued by the committee on December 21, 1964 (approximately $T + 5$ months).

The investigation and report required by the emergency plan, and by Federal Regulations were performed by the Company Investigation Task Force, with the information developed being made available to the USAEC Ad Hoc Committee. The UNC Task Force issued the required report on August 24, 1964 (Ref. 1), with a supplemental report, "Nuclear Aspects of the Accidental Criticality at Wood River Junction" dated November 12, 1964 (Ref. 2). The report issued by the AEC Ad Hoc Technical Review Committee was dated November 6, 1964 (Ref. 3).

In preparation for return to normal operations, review meetings were held with representatives of Rhode Island state agencies, the local town council, and the USAEC. Both state and local officials concurred that the plant was ready to resume normal operations.

Strengthened safety controls, revised operating procedures, formal mechanisms for audit and revision, and an extensive retraining program were begun during the Recovery Phase and completed prior to startup. Through the years these programs have been reviewed and strengthened. Strict adherence has been and continues to be mandatory. The safety record compiled and the stature of the plant in the nuclear industry attest to the effectiveness of the Recovery and Restoration actions.

1. "Nuclear Incident at United Nuclear Corporation, Wood River Junction, Rhode Island, July 24, 1964" (Aug. 21, 1964).
2. "The Nuclear Aspects of the Accidental Criticality at Wood River Junction, Rhode Island" (Nov. 12, 1964).
3. "Report of the AEC Technical Review Committee" (Nov. 6, 1964).
4. J. A. AUXIER, "Nuclear Accident at Wood River Junction," *Nucl. Safety*, Spring issue (1965).
5. JOSEPH FOOTE, "Chain Reaction by Mistake," University of Connecticut Educational Services Civil Defense University extension program (1965).

The Oak Ridge Y-12 Plant Criticality-Accident Emergency Program, W. T. Mee, J. D. McLendon (UCC-OR)

Inadvertent criticality is a concern any time quantities of fissile materials in excess of the minimum critical mass for the material type and form are being processed. Although criticality accident experience has been remarkably good, it is unfortunately true that lethal radiation exposures have occurred in process plants as a result of such accidents. However small its probability, the potential for such an event dictates the need not only for precise safeguards, but also demands an effective emergency response program.

The Oak Ridge Y-12 Plant, operated by Union Carbide Corporation—Nuclear Division—for the U.S. Atomic Energy Commission, has developed and maintains a comprehensive nuclear emergency response organization to

contend with criticality emergencies. To provide consistent and continuing management of this program, responsibility has been assigned to designated personnel assisted by staff specialists who provide expertise in areas such as criticality, radiation protection, industrial relations, and medical surveillance and treatment.

The organization includes specialty teams that perform specified activities to assist in assuring personnel safety and provide information pertinent to production operational recovery and post-accident analyses. These teams include Site Survey, Area Monitoring, Personnel Survey, Radiation Dosimetry, and Medical Assistance.

To determine the need for a criticality-accident alarm system, all areas in which fissionable materials are handled, stored, or processed are surveyed and evaluated by staff specialists. In those cases where the activities are judged to be such that there is a risk of criticality, alarm systems are required.

In Y-12 Plant areas, gamma-ray rate meters utilizing commercially available gamma-detecting scintillators are employed. Two detectors comprise each monitoring station and are installed so as to require the simultaneous response of both instruments to activate the evacuation alarm signal. Fail-safe features are included to ensure that failure will not invalidate the system. To further ensure the continuing operability of the monitors, each instrument contains a built-in source to provide a positive meter response bias at all times. The alarm set point is 50 mR/h.

The signal for evacuation is that described in American Standard ANSI N2.3-1967. The electronically generated sound is initiated by a signal from an activated pair of instruments at any monitoring station. To ensure continuous operation during power failure, backup power is provided by motor generators, which come on-line automatically. In addition to alarming the immediate area, the operational status of each monitor is telemetered into the central control center for the plant. The telemetry system offers two benefits. First, visual display of the status of each monitor provides immediate identification of the alarmed area. Second, the meter values from selected monitors throughout the plant are continuously recorded. These values provide a grid of the plant and a graphic readout which will assist in evaluating the accident characteristics and provide greater confidence in emergency actions to be taken.

The number and placement of stations are determined in accordance with the "Criteria for System Design" of American Standard ANSI N16.2-1969, "Criticality Accident Alarm System." Since we have no massive shielding, the maximum coverage for our detectors is set at 400 ft. To improve reliability, all areas are monitored by two stations separated at least 100 ft to provide protection against failure from physical damage or any possible radiation effects.

In response to the sounding of the evacuation alarm, personnel have been instructed and trained to evacuate the area immediately, without running, through the nearest exit and proceed to the primary evacuation assembly stations located safe distances outside the building.

The assembly station will be manned by personnel assigned to perform specified duties. These personnel are trained to use radiation detecting instruments, to read personnel badges, and report the results through direct telephone hookup to the central control center. They will report pertinent information received from the evacuees about injured personnel and unusual circumstances observed while leaving the building.

The sound of the evacuation alarm also alerts the specialty teams to assemble in predetermined locations. The nucleus of each team is selected from the Radiation Safety Department. This department is responsible for establishing procedures, controls, and performing audits to ensure nuclear criticality safety, health physics, and environmental monitoring compliance for normal plant operations. Each team member thus provides to the team expertise that coincides with the need at times of such emergencies.

To demonstrate the team selection and type of training received, the Site Survey Team is described as an example. The responsibility of the Site Survey Team is to enter the area of the incident, under carefully prescribed safety conditions, obtain information pertinent to the location, physical conditions surrounding the incident, and collect Nuclear Accident Dosimeters (NADs) stationed in the area.

The information obtained will be used to evaluate safing activities, reentry for corrective action, and post-accident analyses. NADs collected from the field will be delivered to the Radiation Dosimetry Team for evaluation of personnel exposures.

Drills are held periodically as part of the training of emergency personnel. A typical practice drill is set up by the emergency director's office to simulate an accident in one of the plant facilities. Maps of the area depicting radiation readings are made for the team to use in entering the area. The team would enter the area and report the simulated readings to the director via radio and also how they are approaching the scene of the incident. The team will continue to search for the simulated incident until they find the site. Again by radio contact, the team describes the situation and makes recommendations for corrective actions. In entering and exiting during the drill, records are made of the NAD locations to indicate those that would have been recovered. After each such drill, critiques are held to review the practice and recommend changes for improving the team activities.

Nuclear-Accident Dosimetry in a Simulated Criticality Accident, R. T. Santoro, R. G. Affel, G. C. Cain, J. H. Greene (UCC-OR)

A criticality accident was simulated using the health physics research reactor (HPRR) to determine the response of three Union Carbide Nuclear Division health physics teams¹ in analyzing exposed personnel (film badges) and area dosimeters to determine the magnitude of the accident and the radiation dose received by personnel at the site of the accident. Since the recovery of exposed personnel may depend on early medical treatment, it is essential that reliable estimates of the dose received by each victim be obtained as rapidly as possible. Continuous reanalyses of the exposed dosimeters combined with clinical procedures must, of course, be made to determine the actual radiation dose accumulated by the exposed individuals.

In the simulated criticality accident reported here, personnel doses were defined as those received by four saline-solution-filled polyethylene manikins (phantoms). Three of the phantoms were placed in line of sight at distances of 1, 2, and 6 m from the reactor. These distances were chosen to assure a wide range of doses to the phantoms. Film badges were placed chest high on each phantom on both the anterior and posterior surfaces to simulate persons facing toward and away from the reactor. The fourth phantom was placed at a distance of

SUMMARY OF RESULTS FOR THE SIMULATED CRITICALITY ACCIDENT

Phantom	Distance from reactor to phantom midplane (m)	Dosimeter location ^b (m)	Initial total dose ^a			Final total dose ^a			Threshold detector (rad) ^{a,e}	Dose from Na activation ^f (rad)
			Team A ^c (rad)	Team B ^d (rad)	Team C ^d (rad)	Team A ^d (rad)	Team B ^d (rad)	Team C ^d (rad)		
1	1.02	A - 0.92	7900	4185	1700	2506	1769	1510	1906	1398
1		P - 1.12	1200	870	1500	2230	1634	1560		
2	2.11	A - 2.01	158	1256	530	1056	662	570	252	555
2		P - 2.21	250	331	610	767	575	590		
3	6.13	A - 6.03	190	213	90	156	145	95	77	98
3		P - 6.23	45	110	95	107	89	92		
4	2.09	A - 1.99	1000	960	200	205	129	195	165 ^f	164
Final area dosimeter total dose (at 10 m)						33	31	33		
Elapsed time (h) ^h			2.2	2.0	2.1	19.5	14.5	7.0		

^aTotal dose = neutron dose plus gamma-ray dose.

^bA = anterior, P = posterior; numeric value is the distance from reactor to film badge plane.

^cUncertainty in these data = $\pm$ factor of 3.33.

^dUncertainty in these data = $\pm$ 25%.

^eThese are doses at a distance corresponding to the midplane of the phantom.

^fNeutron dose only.

^gThe threshold detector adjacent to this phantom was not shielded from the reactor by phantom number 1. Correcting for the shielding of phantom 1, the value obtained is 165 rad.

^hTime from burst to report of results.

2 m from the reactor but in the shadow of the phantom nearest the reactor, and film badges were placed at waist level on the anterior surface.

Area dosimeters were placed 10 m from the reactor at a height of 1.5 m. Threshold detectors were positioned adjacent to each phantom and alongside the area dosimeters.

The criticality accident was simulated by a single burst of the HPRR that resulted in a total yield of 6×10^{16} neutrons. Following the burst, film badges and saline-solution samples taken from each phantom along with one of the area dosimeters were given to each health physics team. Each team was asked to provide in minimum time dose estimates and their uncertainties. The teams were also asked to report revised dose data as the data became available. No information other than the sodium concentration in the saline solution was provided to each team.

The results of the test are summarized in Table I. Immediately after receiving the badges, the three teams estimated the initial dose values from the activity induced in the badge components. While there are differences for a given badge location, the results reported by each team are sufficient for initiating appropriate medical treatment of the exposed personnel. The final dose results were obtained following a more detailed analysis of the activated badge components. Differences in the results reported by each team might arise from several sources: the foil selected as the reference, the use of thermoluminescent dosimeters in the badge, the time after exposure at which the final analysis was made, and, perhaps, human error. However, for a given location the data agree within $\pm 25\%$, consistent with the AEC requirements² for defining exposure doses. Based on this value, the spread in the results is felt to present no significant problem in initiating appropriate medical treatment and possible isolation of exposed victims. Additionally, the final dose values agree with those obtained from the reference threshold detectors within the specified uncertainties.

1. Health physics teams from the Oak Ridge National Laboratory, The Y-12 Plant, and the Oak Ridge Gaseous Diffusion Plant, all of Union Carbide Corporation Nuclear Division, participated in the test.

2. *United States Atomic Energy Commission Manual*, Chap. 0545.

The AEC Radiological Assistance Program, Wayne Smalley (USAEC)

It is the policy of the AEC to make available from its resources radiological advice and assistance, as appropriate, to the regularly constituted authorities at the scene of an incident. The objectives are to minimize injury to people, to minimize loss of property, to cope with radiological hazards, and to protect the public health whenever the AEC believes that such action is necessary, or on request from other agencies or individuals.

For implementation of the AEC Radiological Assistance Program, the United States is divided into eight geographical regions. An AEC office in each region is designated as the Regional Coordinating Office and is responsible for maintaining a Radiological Assistance Plan for that region. Each Regional Coordinating Office maintains a continuously manned emergency telephone.

Assistance may be provided in various ways. For some minor incidents, only advice by telephone may be necessary. In some cases the Regional Coordinating Office may arrange for assistance to be provided at the scene of the incident by knowledgeable persons who are located nearby. If the circumstances warrant, a radiological assistance team will be dispatched by the Regional Coordinating Office to the scene of the incident. For most incidents in which a team is dispatched, the response may be made by one or two individuals. However, for larger or more complex incidents the team may consist of several specialists. Teams may include: Team Leader, Radiation Monitors, Medical Officer, Public Information Officer, and other specialists as required.

The mission of the radiological assistance team is to advise the authorities at the scene of the incident concerning the radiological hazards, the means of reducing them, and the manner of decontaminating the area, and decontaminating or handling persons involved. The team will provide specialized assistance when needed to evaluate the hazard and to terminate the emergency. Medical advice can be given to local physicians regarding the treatment of injuries complicated by exposure to radioactive contamination and on questions of radiation exposure effects. The authorities will also be assisted in providing information to the public as quickly as possible to minimize public alarm and assure orderly conduct of emergency activities.

Specialized assistance for criticality accidents could include such things as: monitoring equipment of all types, extensive capability for analyzing radioactive samples, medical care or advice for radiation injuries, and specialized equipment. Some examples of specialized equipment are: robot for performing remote tasks, mobile whole-body counter, mobile changehouse and shower facility, and an airplane equipped for air sampling and aerial surveys.

Radiological assistance will be continued at the scene until the immediate hazard to public health and safety has been evaluated and the appropriate civil authority or private party responsible for the radioactive material or facility involved can assume responsibility for continuing the personnel monitoring, decontamination, material recovery, and other necessary post-incident operations.

Program statistics are available for the calendar years 1958 through 1973. During those 16 years the AEC responded to 1182 assistance requests. Advice only was given 484 times. The AEC sent one man 224 times and a team 318 times. In the remaining 156 instances, arrangements were made to send a person from outside the AEC, such as a representative from a state health department or a staff member from a university. There was actually radioactive material involved in 865 of these cases; in 317 cases there was not. Most of the 865 incidents involving radioactive material were transportation related and only a few of the incidents actually constituted a potential hazard to the health and safety of the public. (None of these incidents resulted in known radiation injury to a member of the public.)

Nuclear Criticality Safety Analyses of Data

Criticality Analysis of the ACPR Fuel Storage Container, Jeffrey S. Philbin, Berry F. Estes, F. Max Morris (Sandia)

Analysis of the 90-element fuel storage container in use at Sandia Laboratories' annular core pulse reactor (ACPR)^{1,2} was performed to characterize its subcritical multiplication properties under full and partial loading conditions. Neutron transport calculations were performed with the DTF-IV³ code (S_N transport theory) and 16 energy-group cross sections. Experimental verification exists for some of the calculated arrays but the limited number of fuel elements available (40) prevented a full loading of the storage container.

The ACPR fuel storage container is a water-moderated steel frame assembly with aluminum guide tubes and grid plates. Ten Boral plates, 0.635 cm thick, are placed between the grid plates creating 11 channels or rows. The center channel (10.16 cm wide) contains no fuel and the remaining 10 rows (6.35 cm wide) have a capacity of 9 elements each.

The storage container is analyzed by generating cell-disadvantaged cross sections for the fuel element rows so the rows can be treated as homogeneous regions in slab geometry.⁴ The container can then be described in one dimension as a series of parallel slab regions with buckling corrections for the uniform width and height of the rows.

The resultant flux profile for the fully loaded storage array is shown in Fig. 1 (solid line). The sharp depressions indicate the locations of the Boral plates. The flux peak occurs in the outside fuel row beyond the last Boral plate. The relatively low flux calculated for the central region is due to the very loose coupling between the two halves of the container. The separation distance of 17.78 cm between centerlines of the two innermost rows effectively isolates the two halves. Calculations predict a multiplication factor, k_{eff} , of 0.446 for a fully loaded container. Extrapolated experimental results for a fully loaded half-container (valid for both sides if coupling is negligible) yield $k_{eff} \sim 0.48 \pm 0.03$. The experimental results are slightly higher because neutron channeling through the Boral (Ref. 5) is not accounted for in the calculations.

Results from a coarse detail model (dotted line) are also included in Fig. 1. This model represents the Boral plates and the fuel channels as a single homogenized zone. The multiplication factor for this model is 0.438, representing 1.79% difference from the more detailed model.

Additional calculations and experiments assess subcriticality on the basis of geometry alone; i.e., without taking credit for any burnable or removable poisons such as the Boral plates. Although subcriticality by geometry control alone is not required, AEC documents⁶ and other sources⁷ indicate that it is desirable. For these calculations the Boral plates are replaced with water and the multiplication factor increased markedly to 0.945 for the fully loaded array (both halves). There is also a flux depression in the center with this geometry, but the depression is less severe. The relative flux at the center is ~ 0.54 (peak value ~ 1.17). Experimental and calculational results for a partially loaded half-container (35

Fig. 1. Relative flux profiles for the ACPR storage container.

elements without Boral plates) yield $k_{eff} = 0.92 \pm 0.02$ and 0.926, respectively, indicating that coupling between fuel regions is small even if the Boral plates are removed.

Recent criticality guides⁸ indicate that computed neutron multiplication factors for storage arrays should be ≤ 0.95 using validated computational techniques. To demonstrate conclusively that the 0.95 limit is satisfied on the basis of geometry alone, additional calculations (e.g., KENO II⁹) or experimental verification may be necessary since there has been no attempt to measure the fully loaded array without Boral plates or to estimate the error introduced by the one-dimensional model or the cross sections. However, a general discussion of the validity of cell calculations and the importance of the proper choice of boundary conditions is given in Glassstone and Bell.¹⁰

Additional information derived from these calculations is the reactivity worths of the Boral plates ($\Delta k = 0.499$) and the outside fuel rows ($\Delta k = -0.055$) on removal.

The ACPR fuel storage container is suitable for all Triga-type fuels. It has a large capacity (90 elements), is portable, and is inexpensive to build. The design and analysis are felt to be of interest to the research reactor community.

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Experimental Measurements with Arrays of U(97.7)F₆ Neutron-Coupled Through Concrete, J. T. Thomas (UCC-OR)

Critical experiments were performed at the Oak Ridge Critical Experiments Facility as part of a nuclear criticality safety evaluation program. The fissile material was solidified UF₆ in transport cylinders designated¹ as "UF₆ cylinder Model 8A." The isotopic composition of the uranium was 97.66 wt% ²³⁵U, 0.77 wt% ²³⁴U, 0.26 wt% ²³⁶U, and 1.34 wt% ²³⁸U. The hydrogen-to-uranium atomic ratio in the fissile material did not exceed 0.088. The cylinders were of Monel metal with an inside diameter of 20.3 cm, a wall thickness of 4.8 mm, a height of 1.25 m, and had an average tare weight of 45.4 ± 0.2 kg. The six cylinders used in these experiments had mass loadings between 108.0 and 111.5 kg UF₆ with an average of 110.4 kg UF₆. The height of UF₆ in the cylinders varied, resulting in estimated densities from 4.2 to 4.6 with an average of 4.4 g UF₆/cm³.

The reflector for the arrays was constructed of concrete blocks 10.2 × 20.3 × 40.6 cm having a density of 2.15 g/cm³. The chemical and spectrographic analysis of the concrete and the corresponding element number densities are given in Table I.

The experimental assembly consisted of two parallel linear 1 × 3 × 1 arrays of cylinders, one on each half of the tables comprising the Horizontal Displacement Criticality Testing Unit. The cylinders in an array were in a line perpendicular to the direction of table motion. Each array was closely reflected on five sides by 20.3-cm thickness of concrete. The remaining facing surfaces of the arrays, also reflected by close-fitting concrete, had different concrete thicknesses, 10.2 cm on the movable table and 40.6 cm on the stationary table. The total thickness of concrete between the two arrays was 50.8 cm. The cylinders in each array were in contact, center

TABLE I
Chemical and Spectrographical Analysis of Concrete*

Element	Atom Densities, cm ⁻³	Element	Atom Densities, cm ⁻³
H	4.24 × 10 ²¹	Cl	1.9 × 10 ¹⁹
C	1.13 × 10 ²²	K	3.11 × 10 ²⁰
O	4.02 × 10 ²²	Ca	7.27 × 10 ²¹
Na	7.9 × 10 ¹⁹	Ti	4.0 × 10 ¹⁹
Mg	4.99 × 10 ²¹	Mn	1.2 × 10 ¹⁹
Al	3.75 × 10 ²⁰	Fe	1.29 × 10 ²⁰
Si	1.93 × 10 ²¹	Sr	8.9 × 10 ¹⁸
S	1.0 × 10 ²⁰	Ba	3.9 × 10 ¹⁸

*Concrete shielding blocks, 10.2 × 20.3 × 40.6 cm at a density of 2.15 g/cm³.

separation of 22.53 cm, and the distance between centers of the parallel arrays was 73.3 cm at table closure.

The assembly, as described, was subcritical, but exhibited an apparent neutron source multiplication greater than 5. Criticality was achieved by increasing the outer-surface concrete reflector thickness from 20.3 to 30.5 cm of the array on the movable table; i.e., the surface perpendicular to the direction of table motion. Criticality occurred at a table separation of 0.39 cm, and at table closure the *k*_{eff} of the assembly was measured as 1.0007. A second similar addition to the outer reflector surface on the stationary table resulted in criticality at a table separation of 7.36 cm. A summary of these data appears in Table II along with a schematic diagram of the experimental arrangement.

TABLE II
Summary of UF₆ Critical Configurations

Concrete Reflector Thickness, cm	Table Separation, cm at Critical	Remarks
<u>N</u> 20.3	<u>S</u> 20.3	— Assembly subcritical.
30.5	20.3	0.39 <i>k</i> _{eff} = 1.0007 at table closure.
30.5	30.5	7.36 $\Delta k_{eff} > \$1.00$ at table closure.

These experiments demonstrate the misconception of regarding ordinary concrete as an isolating²⁻⁵ medium for neutrons from a neutron chain-reacting system. It is possible, however, for nuclear criticality safety purposes to define a level of reactivity due to neutron coupling among storage areas that may be neglected. Such a quantity would depend on the neutron multiplication factors of the storage areas, the number of areas, the thicknesses of concrete separating the areas,⁶ and an acceptable margin of safety in the operations.

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Criticality Safety Analysis of the Special BNW Nuclear Materials Storage Facility, L. C. Davenport (BNW)

A storage facility has been constructed to accommodate plutonium, enriched uranium, and other special nuclear materials at Pacific-Northwest Laboratories (Battelle-Northwest). The facility is designed to enhance plutonium containment, fire protection, safeguards, and criticality safety. The major storage capacity is provided by a 218-tube array consisting of steel tubes horizontally encased in concrete. To maximize the amount of material that may be stored, this array is designed with only 14- or 16-in. center-to-center spacing

TABLE I
MONTE CARLO ANALYSIS OF THE BNW TUBULAR STORAGE ARRAY

Tube Loading	Tube Size	Enrichment	Neutron Moderation	$k_{eff} \pm \sigma$		Applicable Limit
				Single Tube ^a	Storage Array ^b	
U-Metal	8"Sq.	5 wt% ²³⁵ U	Unmoderated	0.649 $\pm$.007	0.875 $\pm$.006 ^c	$k_{eff} < 0.95$
U-Metal	8"Circular	5 wt% ²³⁵ U	Unmoderated	0.602 $\pm$.005	0.785 $\pm$.006 ^c	$k_{eff} < 0.95$
U-Metal	8"Sq.	10 wt% ²³⁵ U	Unmoderated	-----	1.045 $\pm$.006	Mass Limit; Double Batch Safe
PuO ₂ -UO ₂ ^c	8"Sq.	4 wt% ²³⁹ PuO ₂	Unmoderated	0.568 $\pm$.005	0.832 $\pm$.007 ^c	$k_{eff} < 0.95$
PuO ₂ -UO ₂ ^c	8"Sq.	4 wt% ²³⁹ PuO ₂	H/Pu = 6.71	0.619 $\pm$.006	0.868 $\pm$.006 ^c	$k_{eff} < 0.95$
PuO ₂ -UO ₂ - Polystyrene ^d	8"Sq.	{H/Pu = 210} {H/U = 358}	Intermediate	0.669 $\pm$.008	0.840 $\pm$.008 ^c	$k_{eff} < 0.95$
U-Metal & H ₂ O	8"Sq.	2.25 wt% ²³⁵ U	H/ ²³⁵ U = 243.8 (3.8 g U/cc)	0.828 $\pm$.006	0.924 $\pm$.006 ^c	$k_{eff} < 0.95$
U-Metal & H ₂ O	8"Sq.	3 wt% ²³⁵ U	H/ ²³⁵ U = 264.6 (2.8 g U/cc)	-----	0.976 $\pm$.006	Mass Limit; Double Batch Safe
U-Metal & H ₂ O	8"Sq.	5 wt% ²³⁵ U	H/ ²³⁵ U = 189.8 (2.4 g U/cc)	-----	1.065 $\pm$.007	Mass Limit; Double Batch Safe
U-Metal & H ₂ O	8"Sq.	5 wt% ²³⁵ U	H/ ²³⁵ U = 152.3 (2.9 g U/cc)	-----	1.092 $\pm$.008	Mass Limit; Double Batch Safe
UO ₂	4"Circular	30 wt% ²³⁵ U	Unmoderated	0.604 $\pm$.005	0.804 $\pm$.008 ^c	$k_{eff} < 0.95$
U-Metal & H ₂ O	4"Circular	23 wt% ²³⁵ U	H/ ²³⁵ U = 5.00	0.728 $\pm$.007	0.903 $\pm$.008 ^c	$k_{eff} < 0.95$
U-Metal	4"Circular	25 wt% ²³⁵ U	Unmoderated	-----	0.933 $\pm$.009	Mass Limit; Double Batch Safe
U-Metal	4"Circular	30 wt% ²³⁵ U	Unmoderated	-----	0.973 $\pm$.007	Mass Limit; Double Batch Safe
U-Metal	4"Circular	94 wt% ²³⁵ U	Unmoderated	-----	1.401 $\pm$.008	Mass Limit; Double Batch Safe

^aEach single tube was fully reflected by concrete, 16" thickness minimum.

^bStorage array of 8" square tubes (7.625" inside, 0.1875" wall thickness) are 14 tubes wide by 5 tubes high by 80" deep; 16" center-to-center.

Storage array of 8" SCH 40 tubes are 14 tubes wide by 5 tubes high by 80" deep; 16" center-to-center.

Storage array of 4" SCH 40 tubes was taken to be 7 tubes wide by 8 tubes high by 80" deep; 14" center-to-center (78 tubes total in a L-shaped configuration in actual array).

Each storage array was fully reflected by concrete having a density of 2.281 g/cm³ and a composite aggregate of limestone and silicates. The composition was H - 0.008523, C - 0.020253, O - 0.034590, Na - 0.000020, Mg - 0.001860, Al - 0.000565, Si - 0.001700, K - 0.000040, Ca - 0.011126, Fe - 0.000190 atoms/barn-cm.

^cNatural uranium 0.72 wt% ²³⁵U.

^d580 g (Pu + U)/liter at 14.62 wt% Pu; H/(Pu + U) = 30.6;
240Pu content of Pu = 8.0 wt%; ²³⁵U content of U = 0.151 wt% ²³⁵U;
23.6 liters of solid fuel compact per tube.

^eMeets criticality safety criterion for storage.

which results in some neutron interaction between locations. This interaction is accounted for in the criticality safety analysis by using a Monte Carlo method.¹ Space is also provided in the facility for arrays of DOT-approved shipping containers loaded in accordance with DOT specifications. The facility has been in use since Feb. 1974.

Since the tubular storage array is comprised of storage tubes having fixed dimensions and geometry (see footnote b of Table I), much of the material in this array can be stored without mass limits as long as limits on ²³⁵U or effective Pu enrichment, moderation, and fissile density are met. The effect of cumulative neutron interaction between neighboring tubes has been shown to be significant so the effect of a particular loading on the whole array was considered in establishing criticality safety limits. Monte Carlo calculations were performed to obtain the effective neutron multiplication constant (k_{eff}) for a variety of materials hypothetically loaded into the tubular storage array with all tubes completely filled at theoretical density. Results are given in Table I. To be acceptable, k_{eff} of the array is limited to <0.95 on a 95% upper confidence level.

The presence of large quantities of fissionable material in the facility allows the possibility that a criticality accident could inadvertently occur if procedural controls are not followed. Consequently, strict limits are placed on the handling of fissionable materials within the facility.

The results of the criticality safety analysis show that most of the fissionable material placed in the Battelle-Northwest storage facility is safe by geometry in the tubular storage array provided the ²³⁵U or effective Pu enrichment is limited, and that either moderation or fissile density is limited. In a few cases it is necessary to limit the fissile mass placed in storage tubes to provide criticality safety. The tubular storage array was constructed with a minimum amount of concrete between adjacent locations to maximize the amount of fuel that may be stored. Consequently, a Monte Carlo analysis of the entire array was performed which determines the

interaction within the array and provides the basis for limits on which safe storage can be based. This facility provides Battelle-Northwest with a storage facility that greatly enhances fire protection and safeguards for fissionable material storage.

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Evaluation of Very Low Moderation Low-Enrichment UO_2 Systems, Fred G. Welfare (B&W)

The purpose of this work is to develop the data necessary to allow limited moderation to be used as a nuclear criticality safety parameter either alone or in connection with mass limits. The systems considered are of 4.0% or less in enrichment and are moderated by water.

All calculations were performed using the ANISN¹ one-dimensional transport code and the Hansen-Roach² sixteen-group cross-section set.

Homogeneous systems consist of a uniform mixture of UO_2 and water in which the individual particles are so small that flux variations within the particles may be neglected. Figure 1 shows infinite multiplication factor as a function of H/U ratio for a homogeneous mixture of water and $U(4.0)O_2$. This figure also shows a $U(3.0)O$ curve for purposes of comparison with PCTR experiments.³ The PCTR data exist only down to an H/U of about 3, but this comparison shows acceptable agreement between the calculations and the experimental data down to this point. Figure 1 shows that for $U(4.0)O_2$ -water mixtures, the system cannot be made critical below an H/U ratio of about 0.45.

The value of H/U ratio as obtained above can be used as a nuclear safety parameter, but for many applications it is difficult to limit the ratio to this low value. Figure 2

Fig. 1. Infinite multiplication factor vs moderation.

Fig. 2. Effective multiplication vs density, moderation, and mass; 4% enriched UO_2 water-reflected spheres.

shows the effective multiplication as a function of mass and density for fully water-reflected spheres of homogeneously mixed UO_2 and water. These data show that for systems where the moderation ratio is low, the critical masses are very large. Even including appropriate safety factors, some quite large systems can be allowed if homogeneity can be guaranteed.

The results presented herein forecast that systems of U(4.0)O_2 will have $k_\infty \leq 1.0$ if the H/U ratio is < 0.45 . These results also show the effective multiplication factors resulting from various combinations of limited mass and limited moderation.

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Risk and Reliability Assessment

Systems Analysis of Criticality Safety, R. J. Hall, E. D. Clayton (BNW)

As the nuclear fuel recycle industry grows, the probability of a criticality accident will tend to grow proportionately unless current criticality safety control is improved. (Accident Probability = Probability per Plant times Number of Plants.) Optimum control of criticality safety requires an overview of the entire system. The relationships between the criticality safety philosophy, the criticality data, the use of the data, and various human factors must be considered. Application of the fault tree analysis (FTA) methodology will permit identification of the important control factors and their relationships, indication of what is required to reduce the accident probability per plant, and systematic evaluation of our current criticality safety position. FTA is not new but the application to criticality safety presented in this

paper is. The fault tree is a model that graphically and logically represents the various combinations of possible conditions, both normal and abnormal, that could occur in a system and lead to a criticality accident. Fault trees can be developed in whatever detail is necessary to suit your purposes. The fault tree given in Fig. 1 was specifically constructed to illustrate the above applications. For clarity, some branches have been left out and some have only cursory development.

In fuel recycle, criticality safety is controlled by a set of criticality control requirements (CCR). Current programs, in general, are set up with the goal that at least two independent, concurrent violations of CCR are required before criticality can occur. However, one must always guard against the possibility of a criticality occurring with one, or no, violation of CCR. The partially developed left-hand branch of the tree illustrates a way

Fig. 1. Criticality accident fault tree.

this could occur. The construction of this branch was chosen to illustrate its use as a program monitoring or control tool. The probabilities for the basic events can be determined by the subjective judgment of experts in the field. The probability of occurrence of a criticality through this branch could easily be reduced by increasing the safety margin, which would decrease the probability of "SAFETY MARGIN INADEQUATE TO RENDER SYSTEM SUBCRITICAL." However, this would penalize, perhaps unnecessarily, the economics of operations. Development of more experimental data could also result in a decrease in the probability of a criticality occurring through this path. Use of the tree would permit selection of the optimum way to control the program.

The right-hand branch of the tree illustrates that even technically correct CCR could contribute to the probability of criticality occurring if they were poorly selected

for the operations involved. The data bank for this branch would be the records of CCR violations that have occurred previously. Detailed examination of these records would be required to determine the probabilities for the two "Lack of Understanding" events. However, development of the probabilities for these events would permit selection of the proper program to improve the CCR.

The above examples illustrate some advantages of FTA applied to criticality safety programs. A top-level fault tree model for determining the most probable types of criticality safety limit violations (e.g., geometry, concentration, volume, spacing limit), which will contribute to a future accidental criticality, will also be presented.

Fuel Fabrication

Plutonium Storage Study, A. Uriarte Hueda, F. Rufz Sánchez (Junta de Energia Nuclear-Spain)

INTRODUCTION

This document contains a study on the storage of the plutonium that will be produced in Spain by nuclear power plants, as a previous step before deciding its subsequent utilization.

A study of this type must take several factors into account, such as: evaluation of the plutonium produced in terms of the installed nuclear electrical power, and of the type and characteristics of planned nuclear reactors, chemical form, and isotopic composition of the plutonium to be stored, possible storage systems, and safety standards to be adopted (criticality, fire protection, shielding, flooding, heat elimination, earthquakes, etc.). The expectations of the Spanish Electricity Plan have been used as a basis for this study, which assumes an installed nuclear electrical power of 23,500 MW(e) by the year 1985.

Taking into account the number, type, and characteristics of the planned nuclear reactors, for an average production of irradiated fuels of 35 MTU/yr per 1000 installed MW(e), it has been estimated that the total amount of plutonium recovered in the treatment of irradiated fuels will be ~57 MT up to 1990. For this study, the following isotopic composition has been taken: 1.8% (^{238}Pu); 59.2% (^{239}Pu); 23.2% (^{240}Pu); 12.0% (^{241}Pu), and 3.8% (^{242}Pu).

MATERIAL FOR STORAGE

With the Purex process used in the treatment of the irradiated fuels, the final product will be a concentrated solution of plutonium nitrate (100 to 250 g Pu/liter). However, plutonium oxide (PuO_2) has been adopted as the material for storage with a view to safety standards because of its characteristics (chemical stability, physical state, etc.) from the viewpoints of preparation, shipping, and storage.¹

RISK EVALUATION AND SAFETY MEASURES ADOPTED

Criticality

The study of installation criticality has been done in three stages. First of all, the maximum quantity of plutonium material in each "individual storage unit" has been determined, which will be 4.5 kg of PuO_2 (with the oxide density not being $<2 \text{ g/cm}^3$). This value entails 45% of the critical mass for entirely reflected systems.² The primary-containment enclosure will be a stainless-steel cylinder with a diameter of 14.5 cm and a length of 27.3 (inside dimensions), with a minimum wall thickness of 6 mm. The secondary container will be another stainless-steel cylinder.

The second phase of the study has been calculation of the maximum number of "individual storage units" and their distribution in each "storage module." It has been determined³ that, at most, there will be 125 (with a total approximate content of 500 kg of plutonium). The "storage module" will be a 20.5-m-long $\times$ 3-m-wide $\times$ 3-m-high enclosure, with surfaces lined with 30 cm of concrete and a shielded and fireproof access door.

Finally, with regard to the "general storage installation," considering that 57 MT of plutonium will have been accumulated by the year 1990, it will be necessary to have 114 "storage modules" in an approximately 250- $\times$ 50-m building (Fig. 1).

Gas Generation

A factor to be considered in each "individual storage unit" is the accumulation of gases. Taking into account the amount of plutonium stored in each "individual storage unit" and its isotopic composition, it turns out that the accumulation of gases in the free volume of the cylinder (50% of the total volume) will produce a pressure overload of 60 kg/cm² at 100°C at the end of 700 yr storage. This is for the most unfavorable case in which it is assumed that all the ^{241}Pu has become ^{241}Am .

Fig. 1. Plutonium storage facility.

Radiological Shielding

For calculating the shield of each "individual storage unit," the gamma photons and neutrons must be taken into account. As far as the former is concerned and assuming long-term storage, the total gamma activity of each "individual storage unit" will be 2054 Ci with an average energy of 0.045 MeV. The neutron intensity will be $\sim 1.4 \times 10^6$ n/sec, with an average energy of 0.7 MeV.

Heat Generation

One of the risks presented by the storage of plutonium is the generation of heat, which must be continuously eliminated from each storage module. In this way a high-temperature increase is prevented, both in the storage areas and in the mass of stored plutonium itself. The amount of heat generated by each "individual storage unit" has been calculated⁴ in terms of the isotopic composition of the stored plutonium by assuming that all the ^{241}Pu has converted into ^{241}Am , giving an approximate value of 89 kcal/h of heat generated in each "storage unit."

The heat generated is eliminated by providing adequate ventilation for the installation with the result that with 15 air changes/hour for each "storage module," the ambient temperature will never be $> 35^\circ\text{C}$, and that of the stored plutonium oxide not higher than 120°C .

Other Risks

The risks due to flooding, fires, explosions, and all others related to the characteristics of the site must also be evaluated.

The criteria for selecting the installation site shall be limited by a set of considerations of a varied nature, such as: geography, topography, meteorology, etc., in addition to economics and safety.

DESCRIPTION OF THE INSTALLATION

The storage installation (Fig. 1) shall be located in a 250- x 250-m building consisting of three floors—two underground and one at ground level.

On the bottom floor will be located the areas for collecting and temporarily storing the liquid effluents to enable performing a strict check of all the water evacuated to the environment.

Some Aspects of Ultimate Disposal

Plutonium Criticality Considerations in Reprocessing Wastes and Contaminated Soils, Roger D. Carter (ARHCO-USA)

A combination of reduced water density and addition of materials that have poor neutron absorbing properties but are good isotropic scatterers (for example, oxygen, aluminum, silicon, etc.) can result in mixtures that have critical masses at lower plutonium densities than the 7.0 g/liter which is the limiting critical concentration for plutonium solutions.¹ At the Hanford plant of the U.S. Atomic Energy Commission we have recently performed an extensive characterization of the plutonium in the Z-9 covered trench in which this type of system occurs.² This trench is a concrete-covered excavation with a soil bottom where large quantities of waste solutions were discarded to the ground prior to 1962. Although analyses of the waste solutions before discarding to the Z-9 trench showed very low plutonium concentrations, interaction with the soil resulted in concentration of the plutonium. When soil tests indicated as many as 100 kg of plutonium might be contained in the soil, some concern was felt that a change in the water content of the soil might create a hazard, and the study was begun. This has led to a broader consideration of the effects of plutonium density and hydrogen-to-plutonium (H/Pu) atomic ratios on critical mass.

The "minimum concentration" limit, of course, is not truly a concentration limit but an H/Pu ratio limit at a maximum of about 3600. If, by some method, a normal solution with 10 g Pu/liter were to be expanded by a factor of 10, the plutonium concentration would become 1.0, but the H/Pu ratio and the k_{∞} would remain the same and a critical mass would still be possible, although much larger. Therefore, any system containing dilute plutonium below 7 g/liter cannot be dismissed arbitrarily as being safe as long as H/Pu ratios are <3600.

There are a number of situations where the existence of dilute plutonium with low H/Pu ratios might have to be considered to effectively control criticality. Among these are the dispersion of plutonium in soil, the accumulation of plutonium in fuel processing wastes, and the accumulation in filter materials. These systems may not only have finite critical masses at plutonium concentrations <7 g/liter, but may also contain added scatterers which can reduce the critical masses of the systems to several times less than those of pure plutonium-water systems.

The decrease in critical mass caused by poorly absorbing neutron scatterers has been shown before.^{3,4} These materials tend to decrease the critical mass because they scatter neutrons isotropically, thus tending to hold them within the system. The solid of the Hanford plant contain a number of these materials (oxygen, silicon, aluminum, etc.). Calculations show that if plutonium is dispersed within the soil, the soil components cannot be ignored.

Based on limited 1969 Z-9 trench soil analyses, the material properties, k_{∞} , and buckling were calculated with the GAMTEC II computer code for soil with 30 or 40% void volumes, 0.5 to 15 g Pu/liter, and various water contents.⁵ These calculations showed that although all the k_{∞} first increased compared to the saturated cases, reaching a maximum at H/Pu values of 200 to 300, the material bucklings were highest in the saturated cases except below 5 g/liter (where a drying system might go critical). Maximum $k_{\infty} > 1.0$ were found to below 2.0 g/liter.

The more elaborate 1973 Z-9 characterization defined the three-dimensional distribution below the 9- x 18-m floor. Maximum plutonium concentration was 20 g/liter in a small, thin region, with 2 to 10 g/liter in large areas of soil down to a depth of 30 cm. The higher concentration regions were found to have water volume fractions up to 60 vol%. A detailed three-dimensional analysis was performed using the GEM4 and KENO Monte Carlo computer codes, with the k_{∞} of the trench calculated to be 0.5 (Refs. 6 and 7).

The questions raised during the Z-9 soil analysis indicated there might be mixtures more restrictive than those found in the Z-9 trench. Accordingly, a general parameter study over a range of 2 to 30 g Pu/liter was

Fig. 1. Sphere critical mass, full H₂O reflection.

Fig. 2. Critical areal concentration, infinite slab—full H_2O reflection.

made with normal Pu- H_2O mixtures as a starting point, with the water then being reduced and "soil" added to replace the water that was removed. The soil composition was simplified by using only the oxygen component of Hanford soil since calculations showed that removal of all other components was nearly self-canceling, the removal of iron having a slightly greater positive effect on reactivity than the negative effect of the removal of silicon and aluminum. Sphere critical masses (Fig. 1) and slab areal concentrations (Fig. 2) for fully water-reflected systems were calculated using the HFN diffusion theory code and cross-section sets generated by the GAMTEC II code.⁸ In addition to the simple mixtures, additional parameters were calculated showing the effects of adding cadmium to the mixture to simulate nuclear poison control.

An interesting result was obtained for the addition of cadmium. When graphed, as shown in Fig. 1, a smaller critical mass could be found for a mixture that had more cadmium and less plutonium than another because of differences in the H/Pu ratios of the mixtures. For example, at 6 g Pu/liter, a mixture with 0.2 g Cd/liter and 20% water has a critical mass of 7.6 kg, while a mixture at 15 g Pu/liter and no cadmium has a critical mass of 10.6 kg at 10% water.

From these data, an envelope of minimum values can be obtained. As can be seen, critical systems extend to <2 g/liter. (However, if the critical mass is plotted as a function of the H/Pu ratio, the normal solution curve is the most limiting.) Although the absolute minimum critical mass is still at 30 g/liter and in a normal solution, the critical areal concentration for the slab is less for the low-plutonium-concentration/low-water-content mixtures than for the normal solution.

These data show that low concentration mixtures cannot be ignored as potential criticality hazards. The mixtures used are certainly not the most reactive that can be postulated. Soils that are low in iron content but high in oxides of silicon and aluminum, for example, could have significantly smaller critical dimensions. Spillage of plutonium solutions into such soils might require careful consideration of potential criticality. Fuel reprocessing wastes, normally very low in plutonium may require careful analysis if concentration occurs with self-boiling

or is induced by some other process. Another system with potentially low concentration plutonium, low water content (from fire fighting perhaps), and added scatterers is a bank of high efficiency filters. Any of these systems, along with others of similar composition, may require special analysis to assure criticality safety.

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Criticality Safety Parameters for Advanced LMFBR Fuels

Criticality of Pu-U Nitrate Solution Containing Glass Raschig Rings, R. C. Lloyd, E. D. Clayton (BNW)

Criticality data on the effect of Raschig rings is needed to more accurately establish nuclear criticality safety limits in storing and processing fissile solutions. Experiments being performed provide data on Pu-U systems typical of the LMFBR fuels. These experiments make use of Raschig rings in Pu-U nitrate solutions. The data help establish the effectiveness of the fixed neutron poison as a method of criticality control. The use of Raschig rings permits handling larger quantities of material both safely and economically. These experiments also provide experimental benchmarks for validating calculational techniques and cross-section sets.

The experiments were performed utilizing a 61.03-cm-i.d. cylindrical vessel which was fully reflected with water on the sides and bottom. The wall thickness of the

stainless-steel cylinder was 0.079 cm. (A diagram of the experimental setup is shown in Fig. 1.)

The Pu-U nitrate solutions contained between 30 to 38 wt% Pu in the Pu+U (^{240}Pu content of ~5.7 wt% and ^{235}U content of U, 0.66 wt%). The Pu concentrations in the nitrate solutions ranged between 83 to 92 g/liter. The glass Raschig rings were 1.5-in. o.d., 1.25-in. i.d., and 1.7 in. long. The glass contained boron in the form of B_2O_3 (1.5 wt%). This is equivalent to a nominal $\frac{1}{2}$ wt% boron loading in the rings. This concentration was chosen for the experimental program instead of the more commonly used 4 wt% boron glass so that criticality could be achieved with vessel sizes compatible with available system equipment; i.e., 61.03-cm-diam cylinder. The volume displacement of these rings in the vessel was 19.3%.

The criticality data are summarized in Table I where the critical solution height is given for the various solutions used. The isotopic analyses for the fissionable

TABLE I
Criticality of U-Pu Solutions in Raschig Ring-Filled Tank*

Experiment No.	wt% Pu in U+Pu	Pu Conc ^{a,b} (g/liter)	U Conc ^c (g/liter)	Acid Molarity	Total NO ₃ (g/liter)	Specific Gravity	Critical Height (cm)
106	38.6	92.3	147	4.2	433	1.4691	57.56
107	36.4	89.9	157	4.1	425	1.4778	63.98
108	33.2	86.1	173	3.9	418	1.4820	70.46
109	31.6	83.7	181	3.8	414	1.4850	76.07
110	30.1	82.4	191	3.6	409	1.4995	79.78
111	30.5	81.7	186	3.5	394	1.4856	79.93
112	30.4	80.8	185	3.5	391	1.4812	81.51
113	30.4	79.4	182	3.4	384	1.4731	85.70

*Description of glass Raschig rings:

Boron Nominal (wt%)	Dimensions, in.			Density (g/cm ³)	Chemical Composition, wt%						Ring Displacement
	i.d.	o.d.	Length		SiO ₂	B ₂ O ₃	Al ₂ O ₃	Fe ₂ O ₃	Na ₂ O ₃	K ₂ O	
0.5	1.25	1.50	1.70	2.51	91.55	1.5	2.25	0.05	3.50	1.15	19.3%

$$\frac{^{240}\text{Pu}}{\text{Pu}} = 2.4 \times 10^{-4} \text{ by wt, } \frac{\text{Fe}}{\text{Pu}} = 1.66 \times 10^{-2} \text{ by wt.}$$

^bIsotopic analysis of plutonium in the fissile solution (wt% of Pu):

	Sample 692 (Exp. 106-109)	Sample 694 (Exp. 110)	Sample 697 (Exp. 111-114)
^{239}Pu	93.951	93.901	93.840
^{240}Pu	5.621	5.663	5.710
^{241}Pu	0.348	0.362	0.376
^{242}Pu	0.061	0.059	0.058
^{238}Pu	0.019	0.015	0.016

^cIsotopic analysis of uranium in the fissile solution (wt% of U):

^{238}U	99.317
^{235}U	0.012
^{236}U	0.665
^{234}U	0.006

Fig. 1.

materials, and the chemical composition of the glass Raschig rings, also, are included in the table.

There are currently few data for the effectiveness of Raschig rings on the criticality of Pu-bearing solution systems. The measurements reported herein provide the only available data for code validation on Pu-U nitrate solutions with Raschig rings.

The Criticality Implications of Pu-U Carbide and Pu-U Nitride Fuel Mixtures, S. R. Bierman, B. W. Howes, E. D. Clayton (BNW)

To date, the principal emphasis has been on the development of mixed-oxide fuels for LMFBRs, and the criticality safety efforts have been directed toward providing data for criticality control on mixed oxides in the fuel cycle. However, other types of fuels that may prove to have advantages over the oxides are currently being considered. Two of these consist of mixtures of either plutonium-uranium monocarbides or plutonium-uranium mononitrides.

Although there are some limited calculational results available¹ for plutonium carbide and for plutonium nitride in the undermoderated range (0 to 20 H/Pu), there is no information for plutonium-uranium carbide or nitride mixtures. No experimental criticality data that are appropriate to fuel cycle operations outside reactors exist as yet for either fuel. To provide the necessary insight for selecting and planning experiments, and to provide the interim guidance needed in development of the carbide and nitride fuels, a series of survey calculations was initiated. The results of these calculations are described and com-

Fig. 1. Spherical critical mass 15 wt% Pu + U.

TABLE I

Calculated Critical Conditions for Water-Reflected, Homogeneous Carbide, Oxide, and Nitride Mixtures of Plutonium-Uranium

Concentration (kg Pu+U/liter)	²³⁹ PuC-U(0.71)C-Water				²³⁹ PuO ₂ -U(0.71)O ₂ -Water				²³⁹ PuN-U(0.71)N-Water						
	H		Sphere Mass (kg Pu+U)	Sphere Radius (cm)	Cylinder Radius (cm)	H		Sphere Mass (kg Pu+U)	Sphere Radius (cm)	Cylinder Radius (cm)	H		Sphere Mass (kg Pu+U)	Sphere Radius (cm)	Cylinder Radius (cm)
	²³⁹ Pu+ ²³⁵ U (atom)					²³⁹ Pu+ ²³⁵ U (atom)					²³⁹ Pu+ ²³⁵ U (atom)				
Plutonium-Uranium Containing 15 wt% Pu															
12.947	0.0		315.8	17.99	11.95	--	--	--	--	--	--	--	--	--	--
9.680	4.4		236.5	18.00	11.9	0.0	414.3	21.70	14.6	--	--	--	--	--	--
6.562	12.9		143.1	17.33	11.57	8.4	219.9	20.00	13.9	--	--	--	--	--	--
5.352	18.6		108.2	16.9	11.4	14.2	156.2	19.1	13.4	19.2	100.7	16.50	10.95	--	--
3.761	32.1		68.6	16.33	10.82	27.6	87.4	17.70	12.4	--	--	--	--	--	--
1.553	96.4		20.4	14.63	9.61	91.9	22.8	15.20	9.9	--	--	--	--	--	--
0.826	193		9.5	14.02	9.22	188	10.3	14.40	9.3	193	9.1	13.79	9.06	--	--
0.508	321		6.2	14.25	9.45	317	6.4	14.41	9.57	--	--	--	--	--	--
0.343	482		4.9	15.02	10.10	478	4.9	15.0	10.15	--	--	--	--	--	--
0.131	1286		4.6	20.25	14.16	1280	4.6	20.32	14.22	--	--	--	--	--	--
0.066	2571		17.6	39.93	29.22	2568	7.6	39.93	29.22	--	--	--	--	--	--
0.050	3386		"	"	"	3382	"	"	"	3387	"	"	"	"	"
Plutonium-Uranium Containing 30 wt% Pu															
12.947	0.0		80.8	11.42	7.28	--	--	--	--	--	--	--	--	--	--
9.752	2.2		78.8	12.45	8.1	0.0	107.0	13.80	8.9	--	--	--	--	--	--
6.543	6.6		64.6	13.31	8.37	4.4	81.0	14.35	9.4	--	--	--	--	--	--
3.757	16.4		40.1	13.66	8.88	14.2	47.0	14.40	9.55	--	--	--	--	--	--
3.336	19.4		35.2	13.6	8.8	17.2	41.7	14.4	9.5	19.7	33.5	13.39	--	--	--
1.553	49.3		14.2	12.98	8.42	47.1	17.7	13.95	9.15	--	--	--	--	--	--
0.826	98.6		7.0	12.67	8.19	96.4	8.3	13.4	8.8	--	--	--	--	--	--
0.508	164		4.6	12.92	8.42	162	4.9	13.20	8.62	--	--	--	--	--	--
0.259	329		2.7	13.56	8.97	327	2.9	13.8	9.1	--	--	--	--	--	--
0.131	658		1.9	15.10	10.20	655	1.9	15.15	10.24	--	--	--	--	--	--
0.033	2630		7.7	38.25	27.94	2627	7.8	38.33	28.00	--	--	--	--	--	--
0.029	--		--	--	--	--	--	--	--	2950	18.1	52.79	39.07	--	--
0.025	3473		"	"	"	3471	"	"	"	3473	"	"	"	"	"

pared with corresponding validated calculational results on current LMFBR fuel mixtures of PuO₂-UO₂.

The calculations were made with ENDF/B-III cross-section data which were processed by the computer codes ETOG and FLANGE and averaged over 17 epithermal energy groups and 1 thermal energy group (0 to 0.683 eV) by the code EGGNIT for use in either the neutron diffusion theory code HFN or the neutron transport theory code DTF-IV. The critical sizes for the homogeneous systems having concentrations below about 400 g Pu+U/liter were calculated by HFN. DTF-IV was used for concentrations above about 800 g Pu+U/liter.

Calculations have encompassed homogeneous mixtures in spherical and cylindrical geometries containing 15 and 30 wt% plutonium in the Pu+U. The results are presented in Table I together with data on mixed oxides for comparison.

Curves of water-reflected spherical critical masses for 15 wt% Pu mixtures are shown in Fig. 1 as a function

of Pu+U concentration. As these curves show, the carbide and the nitride Pu-U mixtures are generally more reactive than the Pu-U oxide mixtures. For the dry powder the critical masses for the carbide mixtures are 30 to 40% less than that of the oxides. This difference tends to decrease as the mixtures are diluted with water. At concentrations below about 200 g Pu+U/liter the critical masses become nearly identical. The calculations completed to date with the Pu-U nitride mixtures indicate critical masses of these mixtures to be about 5% less than those of the corresponding density carbide mixtures.

There are currently no experimental data on carbide or nitride mixtures with which to verify the calculations. The validity of the calculated results can only be inferred by comparison with corresponding mixed-oxide data that are available.

1. L. E. HANSEN and E. D. CLAYTON, "Criticality of Plutonium Compounds in the Undermoderated Range, H/Pu ≤ 20," *Nucl. Appl.*, 3, 481 (1967).

Nuclear Fuel Fabricating and Reprocessing in 1975—A Regulated Industry

Status Report: Licensing of Nuclear Fuel Cycle Plants, *Howard J. Larson, R. E. Cunningham (NRC)*

The Energy Reorganization Act of 1974 established the Nuclear Regulatory Commission as an independent Regulatory commission. Within the Commission, the Office of Nuclear Material Safety and Safeguards was established as one of the three statutory offices provided by the Act. The Commission, in announcing its transitory organization on January 20, 1975, provided two principal divisions within the Office: the Division of Materials and Fuel Cycle Licensing and the Division of Safeguards. Together, these divisions are responsible for the principal licensing and regulation of all facilities and materials associated with the processing, transporting, and handling of nuclear materials including safeguards to protect against threat, theft, and sabotage of certain facilities and materials. The Office is further responsible for recommending legislation necessary for the discharge of these functions by the Commission and for developing policy options for Commission consideration.

Estimated time spans from receipt of an application to licensing the startup of uranium fuel fabrication and mixed-oxide fuel fabrication plants and the construction of reprocessing plants are 10, 14, and 16 months, respectively. An environmental statement with a limited hearing for a reprocessing plant requires about 15 months. There is, however, a close interface between licensing both mixed-oxide fuel fabrication plants and reprocessing plants with pending decisions concerning use of recycled plutonium, additional safeguards options, integrated fuel cycle sites, and waste management. The industry has experienced delays in selection of sites, establishing final design, and starting construction of these plants because of ongoing studies in these areas and resolution of pending decisions. The licensing review process has also been stretched out in some instances because of these matters. Experience over the past year with intervenors concerned with licensing of reprocessing plants indicates that, in the near term, construction of certain types of fuel cycle plants are likely to be contested and licensing delays may be experienced.

Licensing casework is planned, scheduled, and monitored by the Office through its computerized project control and manpower utilization system which, in addition, periodically seeks information from applicants about their plans or changes of plans for submission of new applications. Industry participation in this scheduling program is very important for planning manpower requirements and minimizing delays in the license review process. Long-term projections for casework are made by translating projected nuclear power production to the demands it makes on the fuel cycle. The type of fuel cycle plants and the need for additional irradiated fuel storage capacity and plutonium nitrate storage capacity will depend on the timing and the type of decisions that are made with respect to widescale use of recycled plutonium and waste management.

Standards Program for the Nuclear Fuel Cycle, *Robert B. Minogue (NRC)*

This paper presents a report on some aspects of the national nuclear standards program related to developing rules and guides that define acceptable methods for resolving some of the major problem areas in the nuclear fuel cycle. This program is being carried out by the technical staff of the Nuclear Regulatory Commission (NRC) and with substantial input from and use of standards developed within the national consensus standards program coordinated by the American National Standards Institute (ANSI).

Our objective is to identify and promulgate solutions to generic issues of safeguards, siting, and safety associated with regulation of nuclear fuel cycle facilities and materials. The basic safety standards that set policy on acceptable levels of risk are mostly incorporated in the Code of Federal Regulations, under Title 10. In addition to the Regulations, there are the Regulatory Guides developed by the NRC staff. Certain of these guides indicate to applicants the information that should be submitted in support of license applications and form part of the basis of licensing decisions. Other guides provide information on techniques used by the Regulatory staff in evaluating safety problems. Finally, there are the important Regulatory Guides that identify good engineering practice; these are the guides in which ideally we use the product of the national consensus standards program to define how the required level of performance is to be achieved.

A major potential problem area in the nuclear fuel cycle relates to the disposition or reuse of plutonium produced in power reactors. We are developing improved techniques for control and accountability of special nuclear materials at fixed sites and in transit and upgrading physical security requirements for plants and materials. The toxicity of plutonium and the other transuranics requires careful consideration in developing the technological bases for siting facilities. A study is under way to systematically assess this problem to provide a basis for future siting regulations, standards, and guides.

Other important areas of standards needs are related to the appropriate level of application of seismic criteria and seismic analysis methods to fuel reprocessing and mixed-oxide fuel fabrication plants. Engineering studies and assessment of the applicability of existing standards and techniques have been initiated to provide a basis for these standards. Decommissioning criteria are needed for all fuel cycle facilities. Establishment of numerical guidelines for design objectives and limiting conditions for operation to meet "as-low-as-practicable" criteria for effluents from fuel cycle facilities is another problem area that is being dealt with by our technical staff. Standardized methods of effluent monitoring are being developed.

Existing Regulatory programs and standards for shipping containers are being continuously evaluated, modi-

fied, and upgraded as appropriate to assure a high level of protection to the public during transportation of increasing numbers and sizes of radioactive material packages.

Studies will be initiated to provide bases for rules and guides relating to siting, design, construction, and operation of waste storage facilities, structural requirements for containers and shipping casks, and other aspects of waste management.

In summary, I have identified some of the major problem areas in the nuclear fuel cycle and the related plans in the nuclear standards development program. Resolution of the issues involved and the timely development of the needed standards will require the combined efforts of the NRC technical staff and the ANSI national standards groups.

Safety Control of Nuclear Processing Installations in the United Kingdom, E. C. Williams (British Nucl Installations Insp)

One of the principal safety controls on nuclear processing sites with mixed chemical or metallurgical processing plants is to require the licensee to produce a site plan showing every building or plant on the site that may affect nuclear safety, together with a specification of the materials and operations of those buildings or plants. The specifications are relatively simple and descriptive and do not go into technical detail of the plant or detailed safety provisions. The identification of plants that may affect safety can include non-nuclear plants where a malfunction of such a plant could affect a nuclear operation; e.g., a gas pressurizing plant or a flammable liquid plant. Such a site plan must be kept up to date in consultation with the Inspector for the site and must also be reviewed each year.

The site plan concept provides an overall view of plants requiring safety inspection and establishing an early warning system for possible changes of use or process. Within this general control, detailed controls may be elaborated. If it is desired to introduce a new plant on the site, a more detailed plant specification must then be produced. This specification must indicate the type of process to be introduced, the nature of the materials to be processed, the nature and volume of likely wastes, the main nuclear hazards, and a Design Safety Report, which is a statement of the principles and means for the control of safety. This information must be submitted to the Inspectorate before the licensee begins to construct the new plant on his site.

Once construction of the plant has commenced, field inspection concentrates on the quality of materials of construction, fabrication and inspection methods, and component testing. By this time a plant commissioning program that details the plant tests important to safety is formally required. No product material may be intro-

duced into the plant other than for the demonstration of a commissioning or test procedure. During this phase of the development of the plant the main operating rules relating to safety must be developed. In the case of plants processing fissile materials, these operating rules will include the issue of Nuclear Safety Certificates which incorporate both a criticality analysis of a process vessel or vessels and a statement of operating procedures and control limits. On satisfactory completion of all the commission tests and the production of safety documentation, the operator is then formally authorized to commence production operation of the plant.

It is characteristic of processing and metallurgical plants that processes can be and are frequently modified, either to increase production or to produce a modified product. In many plants, even those with highly active processes, equipment also can be modified and replaced. It is important that such modifications should not affect the basic safety principles of the plant or that the safety principles should be extended or adapted to the new process.

In larger plants the licensee may be required to keep a list of projects involving plant or process modification. These are discussed with the Nuclear Inspector, and those identified as being of major significance to safety are treated in the same manner as new plants or processes. The same specification, commissioning, and operating procedures must be followed. Where fissile materials are concerned, this latter may be accomplished by the issue of a new Nuclear Safety Certificate. Minor modifications can be more appropriately dealt with by the licensee's own administrative procedures.

The licensee at all times is fully responsible for safety of the site and one of the requirements of a nuclear site license is that the licensee must be suitably organized administratively to deal with the various technical aspects of safety. For this purpose he is required to form a Safety Committee to consider matters affecting safety on the site and to submit safety issues to this Committee and receive its advice. For complex sites or on matters involving highly specialized construction, the licensee may institute subcommittees or specialist technical groups to assist the Safety Committee. On smaller sites the licensee may elect to engage consultants of adequate technical repute. The Inspectorate is kept fully informed of these matters by the formal requirement that all Safety Committee or subcommittee minutes and papers must be furnished within 14 days of any meeting.

PANEL DISCUSSION

Moderator: S. H. Smiley (NRC)

Panelists: R. E. Cunningham (NRC)

R. B. Minogue (NRC)

E. C. Williams (British Nucl
Installation Insp)

D. F. Knuth (NRC)

H. J. C. Kouts (NRC)

Physics of Thermal Reactors

Benchmark U-D₂O Experiments and Comparison with Computations, D. J. Pellarin, B. M. Morris (duPont-SRL)

Material bucklings and reaction rate parameters were measured for heavy-water (D₂O)-moderated uniform lattices in the Savannah River Laboratory experimental facility, the SE.¹ Two different slightly enriched tubular uranium fuel assemblies (Table I) were examined over a wide range of triangular pitches. The results are compared with multigroup integral transport theory cell computations using ENDF/B-IV cross sections and the Nordheim resonance treatment.

Previous analyses of benchmark U-D₂O data have been restricted to simple rod lattices.² The purpose of this

work is to expand the benchmark experimental data base to include uniform lattices of coaxial fuel assemblies. Separate parameter measurements are reported for inner and outer fuel to provide a more detailed and rigorous comparison with computation than previously available.

The reaction rate parameters measured are: ratio of epithermal-to-thermal ²³⁸U captures (ρ_{28}); ratio of epithermal-to-thermal ²³⁵U fissions (δ_{28}); ratio of ²³⁸U captures to ²³⁵U fissions (C^*); ratio of ²³⁸U fissions to ²³⁵U fissions (δ_{28}); and ratio of ¹⁷⁶Lu-to-⁶³Cu subcadmium captures in the fuel normalized to the same ratio in a Maxwellian thermal flux (R). Measurement and analysis techniques used in this study were identical to those described in Ref. 3.

The measured results are compared in Table II with RAHAB⁴ computations. For uniform lattices, RAHAB is similar to the HAMMER⁵ code. The following conclusions are noted:

1. Calculated values of $\delta_{28}^{\text{inner}}$ and $\delta_{28}^{\text{outer}}$ for both fuel types are ~23 and ~13% higher than measured values, respectively, indicating too much epithermal fission in ²³⁸U.

2. Calculated values of ρ_{28}^{inner} for both fuel types are ~20% higher, ρ_{28}^{outer} values for Type II fuel are ~10% higher, and Type I fuel values are ~2% higher than measured values. Also, calculated values for C_{inner}^* for both fuel types are ~10% higher, C_{outer}^* values for Type II fuel are ~5% higher, and for Type I fuel values are ~1% higher than measured values. These facts imply that ²³⁸U epithermal captures are overestimated.

3. Calculated δ_{28} values are consistently lower by ~10% for inner fuel and ~7% for outer fuel for both fuel types.

4. Calculated R values are ~5% higher for Type II fuel (fuel/coolant volume ratio = 1.04), and ~2% higher for Type I fuel (fuel/coolant volume ratio = 1.94) than measured R values.

5. The calculated reactivities for both fuel types are consistently low by about 5%, a significantly larger error than observed for benchmark rod lattices.⁶

These experiments provide a set of consistent data over a range of D₂O lattice conditions that serve as benchmarks for improvement and development of lattice cell physics codes.

TABLE I
Assembly Geometry and Fuel Composition

Geometry, inches	Type I	Type II
Al Inner Housing		
O. D. ^a	0.8706	1.6512
I. D. ^b	0.5320	1.4730
Inner Fuel Slug		
Al Cladding^c		
O. D. ^a	1.9975	2.6787
I. D. ^b	1.166	1.958
Fuel		
O. D. ^a	1.914	2.605
I. D. ^b	1.226	2.018
Outer Fuel Slug		
Al Cladding^c		
O. D. ^a	3.076	3.700
I. D. ^b	2.400	3.105
Fuel		
O. D. ^a	3.016	3.640
I. D. ^b	2.460	3.165
Fuel Composition, wt %		
²³⁵ U	0.860	1.10
²³⁸ U	99.101	98.877
²³⁹ U	0.032	0.023

- a. Outside diameter, includes rib volumes.
b. Inside diameter.
c. A thin (0.5 mil for Type I fuel; 0.41 mil for Type II inner fuel; 0.47 mil for Type II outer fuel) nickel flashing exists at the fuel-cladding interface and was homogenized in the cladding for the calculations.

1. A. E. DUNKLEE and C. E. JEWELL, "Zero Power Measurements on a High-Flux Demonstration Lattice," DP-1076, E. I. du Pont de Nemours and Co., Savannah River Lab. (1967).
2. H. C. HONECK and J. L. CRANDALL, "The Physics of Heavy Water Lattices," in *Reactor Technology, Selected Reviews - 1965*, L. E. LINK, Ed., Supt. of Documents, U.S. Gov. Printing Office, Washington, DC; also issued as BNL-8253 (June 1964).
3. J. L. JARRIEL and D. J. PELLARIN, "SRL Experiments for the Japanese Power Reactor and Nuclear Fuel Development Corporation," DP-1219, E. I. du Pont de Nemours and Co., Savannah River Lab. (1970).
4. H. C. HONECK, "The JOSHUA System," DP-1380, E. I. du Pont de Nemours and Co., Savannah River Lab. (to be issued).

TABLE II
Measured^a and RAHAB Computed Parameters

A. Type I Lattices:

Pitch ^b Parameter	5.5				6.0				7.0				8.0			
	Meas.		Calc.		Meas.		Calc.		Meas.		Calc.		Meas.		Calc.	
R	1.757 ±0.017	1.628 ±0.024	1.811 ±0.013	1.673 ±0.022	1.759 ±0.017	1.575 ±0.023	1.749 ±0.017	1.604 ±0.023	1.656 ±0.016	1.477 ±0.022	1.655 ±0.015	1.504 ±0.021	1.547 ±0.015	1.407 ±0.021	1.590 ±0.015	1.439 ±0.021
δ_{20}	0.254 ±0.003	0.205 ±0.004	0.313 ±0.012	0.229 ±0.003	0.203 ±0.002	0.157 ±0.003	0.249 ±0.002	0.178 ±0.003	0.142 ±0.0015	0.109 ±0.002	0.174 ±0.0015	0.121 ±0.002	0.105 ±0.001	0.0787 ±0.0015	0.132 ±0.001	0.0901 ±0.0015
δ_{20}	0.134 ±0.005	0.0928 ±0.004	0.125 ±0.005	0.0903 ±0.0035	0.128 ±0.005	0.0878 ±0.0035	0.117 ±0.005	0.0822 ±0.0035	0.121 ±0.005	0.0777 ±0.0030	0.109 ±0.005	0.0734 ±0.0030	0.115 ±0.0045	0.0740 ±0.0030	0.104 ±0.0045	0.0685 ±0.0030
C*	1.640 ±0.016	1.627 ±0.024	1.777 ±0.016	1.634 ±0.024	1.433 ±0.014	1.440 ±0.025	1.570 ±0.014	1.425 ±0.025	1.178 ±0.012	1.128 ±0.017	1.309 ±0.012	1.170 ±0.017	1.054 ±0.011	0.992 ±0.015	1.154 ±0.011	1.026 ±0.015
ρ_{20}	2.575 ±0.019	2.455 ±0.016	3.008 ±0.019	2.504 ±0.016	1.990 ±0.013	1.941 ±0.050	2.396 ±0.013	1.952 ±0.050	1.402 ±0.012	1.303 ±0.028	1.687 ±0.012	1.330 ±0.028	1.017 ±0.016	0.914 ±0.034	1.299 ±0.016	0.998 ±0.034
$B^0(\text{cm}^{-2})$	-0.000465 ±0.000050		-0.001095 ±0.000050		0.000056 ±0.000050		-0.000457 ±0.000050		0.000684 ±0.000040		0.000288 ±0.000040		0.000922 ±0.000030		0.000630 ±0.000030	

B. Type II Lattices:

Pitch ^b Parameter	6.35				7.0				8.08				9.25				14.0			
	Meas.		Calc.		Meas.		Calc.		Meas.		Calc.		Meas.		Calc.		Meas.		Calc.	
R	1.737 ±0.017	1.586 ±0.023	1.800 ±0.013	1.673 ±0.022	1.644 ±0.016	1.539 ±0.023	1.734 ±0.016	1.597 ±0.023	1.589 ±0.016	1.419 ±0.021	1.649 ±0.015	1.506 ±0.021	1.530 ±0.015	1.378 ±0.021	1.585 ±0.015	1.440 ±0.021	1.392 ±0.014	1.267 ±0.019	1.459 ±0.014	1.318 ±0.019
δ_{20}	0.252 ±0.003	0.199 ±0.004	0.307 ±0.012	0.231 ±0.003	0.195 ±0.002	0.155 ±0.003	0.242 ±0.002	0.177 ±0.003	0.148 ±0.0015	0.108 ±0.002	0.177 ±0.0015	0.125 ±0.002	0.112 ±0.001	0.0809 ±0.0015	0.137 ±0.001	0.0938 ±0.0015	0.0648 ±0.0010	0.0441 ±0.0010	0.0783 ±0.0010	0.0504 ±0.0010
δ_{20}	0.115 ±0.004	0.0841 ±0.003	0.102 ±0.004	0.0773 ±0.003	0.105 ±0.004	0.0731 ±0.003	0.0962 ±0.004	0.0705 ±0.003	0.102 ±0.004	0.0672 ±0.003	0.0909 ±0.004	0.0640 ±0.003	0.103 ±0.004	0.0674 ±0.003	0.0884 ±0.004	0.0604 ±0.003	0.0954 ±0.004	0.0620 ±0.0025	0.0869 ±0.004	0.0573 ±0.0025
C*	1.347 ±0.013	1.281 ±0.020	1.470 ±0.013	1.340 ±0.020	1.184 ±0.012	1.093 ±0.016	1.289 ±0.012	1.153 ±0.016	1.005 ±0.010	0.919 ±0.014	1.096 ±0.010	0.962 ±0.014	0.896 ±0.009	0.805 ±0.012	0.972 ±0.009	0.844 ±0.012	0.755 ±0.008	0.663 ±0.010	0.789 ±0.008	0.678 ±0.010
ρ_{20}	2.734 ±0.026	2.416 ±0.076	3.235 ±0.026	2.688 ±0.076	2.172 ±0.013	1.902 ±0.034	2.558 ±0.013	2.058 ±0.034	1.583 ±0.013	1.328 ±0.024	1.896 ±0.013	1.462 ±0.024	1.245 ±0.015	0.996 ±0.029	1.494 ±0.015	1.113 ±0.029	0.833 ±0.019	0.620 ±0.043	0.945 ±0.019	0.648 ±0.043
$B^0(\text{cm}^{-2})$	0.000290 ±0.000050		-0.000264 ±0.000050		0.000798 ±0.000040		0.000324 ±0.000040		0.001182 ±0.000030		0.000857 ±0.000030		0.001318 ±0.000030		0.001081 ±0.000030				0.000889 ±0.000030	

- a. Small calculated corrections have been applied to convert computed values to 99.75 mol % D₂O moderator purity.
b. Triangular lattice pitch, inches.
c. Inner fuel.
d. Outer fuel.

5. J. E. SUICH and H. C. HONECK, "The HAMMER System: Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors," DP-1064, E. I. du Pont de Nemours and Co., Savannah River Lab. (1967).
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Nuclear Criticality Safety Guidance

A 218-Neutron-Group Master Cross-Section Library for Criticality Safety Studies, W. E. Ford, III, R. M. Westfall, C. C. Webster (ORNL)

The AMPX system¹ was used to generate a P₃ 218-neutron-group master² cross-section library from ENDF/B-IV data for the fuel, structural, and neutron-absorbing materials listed in Table I. The library is the data base for the generation of broad-group cross sections for

^aThe AMPX system uses a fine-group master cross-section library as input to generate fine- or broad-group coupled neutron-gamma or uncoupled cross sections in formats for several codes (e.g., ANISN, VENTURE, MORSE, DOT, SWANLAKE) at a relatively low cost and in a relatively fast computer run.

shipping cask calculations and other criticality safety analyses using codes such as KENO³ and ANISN.³

Selection of the fine-group energy structure for the 3-eV to 20-MeV energy range included consideration of the resonance structure of prominent nuclei, the thresholds of important reactions, and the fission spectra. For $10^{-5} \leq E_n < 3$ eV, 78 closely spaced thermal groups were chosen to examine the effects of low-energy resonances and thermal-neutron upscatter. Distribution of the 218 groups within the Hansen-Roach⁴ 16-group boundaries is shown in Table II.

Adequacy of the group structure and validity of selected data sets from the library were tested by P₃S₈ XSDRNPM^b calculations of k-eff for two benchmark

^bXSDRNPM is the AMPX module with one-dimensional S_n capability. See Ref. 1.

TABLE I
Fine Group* Master Cross-Section Library for Criticality Safety Analysis

Nuclide	ENDF/B-IV MAT No.	Master Library ID Number	Nuclide	ENDF/B-IV MAT No.	Master Library ID Number
H	1269 ^a	1269	¹⁰⁷ Ag	1138 ^b	113802
⁴ He	1270	127003	¹⁰⁸ Ag	1139 ^b	113902
⁶ Li	1271	127103	Cd	1281	128103
⁷ Li	1272	127203	Sn	7039 ^c	703903
⁹ Be	1289	128903	Ba	7040 ^c	704003
¹⁰ B	1273	127303	Gd	1030 ^b	103002
¹¹ B	1160	116003	¹⁶⁴ Dy	1031 ^b	103102
¹² C	1274	127403	¹⁷⁵ Lu	1032 ^b	103202
¹⁴ N	1275	1275	¹⁷⁶ Lu	1033 ^b	103302
¹⁶ O	1276	1276	Hf	1034 ^b	103402
F	1277	1277	¹⁸¹ Ta	1127 ^b	112702
²³ Na	1156 ^b	115602	¹⁸² W	1128 ^b	112802
Mg	1280	128003	¹⁸³ W	1129 ^b	112902
²⁷ Al	1193	1193	¹⁸⁴ W	1130 ^b	113002
Si	1194	119403	¹⁸⁶ W	1131 ^b	113102
³¹ P	7019 ^c	701903	¹⁸⁵ Re	1083 ^b	108302
³² S	7020 ^c	702003	¹⁸⁷ Re	1084 ^b	108402
Cl	1149	114903	¹⁹² Au	1283	128303
K	1150	115003	Pb	1288	128803
Ca	1195	119503	²³² Th	1296 ^b	129602
Ti	1286	128603	²³³ U	1260 ^b	126002
V	1196	119603	²³⁴ U	1043 ^b	1043
Cr	1191 ^b	119102	²³⁵ U	1261 ^b	1261
⁵⁵ Mn	1197 ^b	119702	²³⁶ U	1163 ^b	116302
Fe	1192 ^b	119202	²³⁸ U	1262 ^b	1262
⁵⁹ Co	1199 ^b	119902	²³⁷ Np	1263 ^b	126302
Ni	1190 ^b	119002	²³⁸ Pu	1050 ^b	105002
Cu	1295 ^b	129502	²³⁹ Pu	1264 ^b	126402
Zn	4530 ^d	453003	²⁴⁰ Pu	1265 ^b	126502
⁹⁵ Zr ^b	1202	120202	²⁴¹ Pu	1266 ^b	126602
ZIRC-2	1284 ^b	128402	²⁴² Pu	1161 ^b	116102
⁹³ Nb	1189 ^b	118902	²⁴¹ Am	1056 ^b	105602
Mo	1287 ^b	128702	²⁴⁴ Cm	1162 ^b	116202

*The 218 neutron groups span the range $10^{-5} \leq E_n \leq 2 \times 10^7$ eV. Seventy-eight "thermal" groups are in the range $10^{-5} \leq E_n \leq 3.05$ eV.

^aENDF/B MAT 1002 was used for water-bound thermal data.

^bThis is a resonance material. A potential scattering cross section of 5×10^4 b/atom was used in the unresolved resonance calculation.

^cENDF/B formatted data were obtained from Howerton's library.⁸

^dENDF/B formatted data were obtained from RSIC's Defense Nuclear Agency Working Cross-Section Library.⁹

TABLE II
Correspondence of 218n and Hansen-Roach
16-Group Structures

Hansen-Roach		218n Group Numbers	Number of 218n Groups per Hansen-Roach Group
Group Number	Energy Range		
1	20-3 MeV	1-4	4
2	3-1.4	5-9	5
3	1.4-0.9	10-17	8
4	900-400 keV	18-31	14
5	400-100	32-37	6
6	100-17	38-48	11
7	17-3	49-55	7
8	3-0.55	56-66	11
9	550-100 eV	67-78	12
10	100-30	79-108	30
11	30-10	109-124	16
12	10-3	125-141	17
13	3-1	142-182	41
14	1-0.4	183-196	14
15	0.4-0.1	197-208	12
16	0.1-10 ⁻⁵	209-218	10

critical experiments. First, the 4.9%-enriched $\text{UO}_2\text{F}_2\text{-H}_2\text{O}$ solution critical sphere experiment of Johnson and Cronin⁵ was analyzed. This system had a critical ^{235}U mass of 4.14 kg with an $\text{H}/^{235}\text{U}$ atom ratio of 1002. The solution was contained within a 0.32-cm-thick Al shell having an outer radius of 34.82 cm. A k-eff of 0.9998 was calculated for the low-enriched thermal system.

Second, the 93.2%-enriched $\text{UO}_2\text{F}_2\text{-H}_2\text{O}$ solution critical sphere experiment of Fox⁶ was analyzed. This system had a critical mass of 3.25 kg ^{235}U with an $\text{H}/^{235}\text{U}$ atom ratio of 1393. The solution was contained within a 0.32-cm-thick Al shell having an outer radius of 34.92 cm. A k-eff of 1.0046 was calculated for the high-enriched thermal system.

The master cross-section library is available on magnetic tape from the Radiation Shielding Information Center.^c Efforts to determine a broad-group subset of the 218 neutron groups for collapsing data sets from the library for shipping cask criticality studies are described in Ref. 7.

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Procedure for Determining Broad-Group Energy Structure for Criticality Safety Calculations, R. M. Westfall, D. M. Plaster, W. E. Ford, III (ORNL)

The treatment of basic cross-section data preparatory to the calculation of criticality by computer codes has been demonstrated to influence the results.¹ A procedure has been developed for establishing the most appropriate broad-group neutron energy structure for criticality safety analyses from a standard 218-neutron-group structure and master cross-section library described in Ref. 2. Emphasis is being placed on a group structure for shipping cask calculations.

The procedure gives primary consideration to the influence on calculated k-eff's and to the cross-section characteristics of the following nuclides: H , ^{10}B , ^{11}B , C , O , N , F , Na , Fe , Ni , Cr , Al , Mn , Co , Zr , Mo , Ag , In , Cd , Sn , Gd , Hf , ^{232}Th , ^{233}U , ^{235}U , ^{238}U , Pb , ^{239}Pu , ^{240}Pu , and ^{241}Pu . Trial broad-group cross-section sets for these nuclides are collapsed from the standard set and a series of model calculations indicates which broad-group sets meet an acceptance criterion of $\Delta k_{\text{eff}}/k_{\text{eff}} \leq 0.3\%$ when compared to reference k-eff's resulting from 218-group analyses.

A three-region spherical model is utilized to examine the effect of specific nuclides on the resulting k-eff. The inner and outer regions contain the nuclide of interest at an atomic density corresponding to the possible theoretical density compatible with the physical state. The intermediate region contains a uranium metal-water mixture used to produce criticality. Various neutron spectrum ranges are emphasized by the selection of appropriate $\text{H}/^{235}\text{U}$ atom ratios. The median fission energy (MFE) is used as a spectral index to characterize four spectral ranges: (a) $\text{MFE} < 1 \text{ eV}$, (b) $1 \text{ eV} < \text{MFE} < 100 \text{ eV}$, (c) $50 \text{ keV} < \text{MFE} < 200 \text{ keV}$, and (d) $\text{MFE} > 200 \text{ keV}$. The characteristics of thermal- and epithermal-neutron spectra are established with the model having both low (3.3%) and high (93.2%) ^{235}U enrichment to amplify the effects of ^{235}U and ^{238}U . The nuclide of interest appears in the intermediate region at 10% of the ^{235}U atom density.

An example of the group selection procedure considers ^{238}U as the nuclide of interest in a low-enriched thermal model. The AMPX system³ was used to collapse the necessary data sets from the fine-group master cross-

TABLE I
Broad-Group Determination with ^{238}U as
the Nuclide of Interest

Number of Groups	k-eff	% $\Delta k_{\text{eff}}/k_{\text{eff}}$
25 ^a	1.00000	0.76
218 ^b	0.99242	N/A
16 ^c	1.00540	1.31
19 ^d	1.00156	0.92
25 ^e	0.99674	0.44
25 ^f	0.99515	0.27 ^g

^aInitial calculation with trial group structure consisting of the first 15 Hansen-Roach groups plus 10 thermal groups. Calculation was made to establish the critical dimensions for the system.

^bCalculation to establish the reference effective multiplication factor.

^cHansen-Roach 16-group structure.

^dThe first 15 Hansen-Roach groups plus four thermal groups.

^eThe 19 groups plus six extra groups in the resonance range.

^fThe above 25 groups with adjusted resonance group boundaries.

^gAcceptance criterion satisfied.

section library into a trial 25-group library and to perform the appropriate Nordheim resonance treatments. This 25-group structure corresponded to the first 15 groups of the Hansen-Roach⁴ 16-group structure plus 10 groups below 0.1 eV.

The analysis was performed with the XSDRNPM discrete ordinates module from the AMPX system in the P_3S_8 approximation. Initially, the trial 25-group subset was used to conduct a critical zone width search on the intermediate region. The resulting critical dimensions for the model were then analyzed with the 218-group structure to establish the near-critical reference value of the effective neutron multiplication factor. Subsequent analyses, summarized in Table I, demonstrated the effects of adding thermal and resonance region groups to the Hansen-Roach 16-group structure. The final 25-group structure includes a relatively narrow group between 6.25 and 7.0 eV to bracket the large 6.67-eV ^{238}U resonance level. These results demonstrate one step in a consistent procedure to establish a neutron-energy-group structure applicable to a wide range of criticality safety analyses. At the conclusion of this procedure, a master broad-group library will be generated for shipping cask calculations.

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Effect of Gadolinium on Criticality of U+Pu Solutions, R. C. Lloyd (BNWL)

Experiments were recently completed to provide data on the effectiveness of soluble poisons on the criticality of plutonium-uranium solution mixtures. Although some data have been reported on Pu solutions, none have been available for U + Pu solutions.¹ The current experiments provide data on systems typical of LMFBF fuel compositions. Gadolinium was chosen for this study because of its high neutron cross section, high solubility, and compatibility in the separation process; i.e., gadolinium is more easily separable from Pu than cadmium or boron. The data can have applications in criticality control and prevention and serve also as experimental benchmarks for validating calculational techniques and cross-section sets.

The experiments were performed in a 61.03-cm-diam, water-reflected, stainless-steel cylindrical vessel. The gadolinium was in the form of gadolinium nitrate mixed homogeneously with the U + Pu nitrate solution. The plutonium concentration was ~76 g Pu/liter, which amounts to ~30% of the total U + Pu. The ^{240}Pu content was 5.77 wt%; the ^{238}U content was 0.66 wt%. Gadolinium concentration was varied up to 1.06 g/liter, in steps, for which the critical height on the cylindrical vessel was determined. The data from these experiments are given in Table I (see following page). The results demonstrate the effectiveness of gadolinium as a neutron poison in the U + Pu solution system.

Calculational correlation to the experiment data, using KENO Monte Carlo codes, is now in progress. These calculations will be presented and discussed.

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Subcritical Measurements with Irradiated and Unirradiated Uranium Metal Fuel Elements, Hans Toffer, Larry A. Nielsen (United Nucl Ind)

With the lack of spent-fuel reprocessing capacity, an increased emphasis is placed on more efficient utilization of space in spent-fuel storage pools. Each year, United Nuclear Industries, the operator of the Hanford N Reactor, must store 300 MT of spent fuel. The available storage pool capacity has been constrained by applying unirradiated fuel critical mass limits to fuel storage configurations.¹ Incorporating fuel irradiation effects on storage parameters represents a basic change in criticality control philosophy that might require experimental verification of irradiated fuel critical mass parameters.

Analytical studies show that unnecessarily large margins of safety may be applied to irradiated fuel storage by assigning unirradiated fuel critical mass parameters.² Subcritical measurements with irradiated and unirradiated fuel elements (see Table I for geometric and isotopic details) were undertaken by United Nuclear Industries, with technical assistance supplied by the Battelle Pacific Northwest Critical Mass Laboratory, to confirm the analytical results.³ No similar data have been obtained for low-enrichment, spent, uranium-metal fuel elements

Gd-Poisoned Homogeneous U+Pu Solution

Exp. No.	Critical Height (cm)	Pu Concentration (g/liter)	U Concentration (g/liter)	Acid Molarity	Total NO ₃ (g/liter)	Specific Gravity	Gd (g/liter)
127	19.10	75.6	171.9	3.27	371.4	1.446	0.042
128	21.93	76.3	173.4	3.26	377.0	1.452	0.180
129	24.61	76.3	174.6	3.42	383.8	1.450	0.288
130	29.47	76.6	174.7	3.07	375.8	1.456	0.459
131	33.42	76.6	175.3	3.05	365.2	1.454	0.581
132	37.15	77.0	175.9	3.14	370.8	1.456	0.679
133	43.71	77.1	175.1	3.13	380.7	1.456	0.800
134	53.12	77.3	177.7	3.24	376.4	1.461	0.923
135	62.12	77.2	176.3	3.27	381.3	1.455	1.01
136	75.32	77.3	176.8	3.25	383.8	1.462	1.06
Isotopic Analysis - wt%							
		²³⁸ Pu - 0.012	²³⁸ U - 99.317				
		²³⁹ Pu - 93.789	²³⁶ U - 0.012				
		²⁴⁰ Pu - 5.766	²³⁵ U - 0.665				
		²⁴¹ Pu - 0.371	²³⁴ U - 0.006				
		²⁴² Pu - 0.062					

TABLE I (Paper 4)

Experimental and Calculated Results of the Subcritical Measurements

Type of Fuel	Assembly Loadings Number of Fuel Tubes	Calculated ^a k-eff	Experimental ^b k-eff
Unirradiated Mark IA ^c	43	0.95	
	49	0.964	
	51	0.968	0.91
	54	0.974	0.94
	72 ^d	0.999	1.0075
Irradiated (average exposure 3020 MWd/MT) Mark IA	73	0.961	0.90
	85	0.971	0.93
	91	0.975	0.95
	158 ^d	1.000	1.0064

^aBounds calculated using the HAMMER code.^bPulse-neutron source data corrected for kinetic distortion and using $\beta_{eff} = 0.0075$ and $\beta_{eff} = 0.0064$.^cZircaloy-2-clad uranium tubesOuter tube clad/core o.d. = 6.11 cm/5.98 cm, clad/core i.d. = 4.49 cm/4.62 cm
Enrichment = 1.25 wt% ²³⁵UInner tube clad/core o.d. = 3.16 cm/2.96 cm, clad/core i.d. = 1.24 cm/1.12 cm
Enrichment = 0.95 wt% ²³⁵U^dExtrapolated from prompt-neutron decay rate data.

Fig. 1. Partially loaded test assembly under 6 m of water showing detector tube locations and a fuel tube being moved into position.

in a geometry readily reproducible by calculational methods.

Both unirradiated and irradiated fuel measurements were performed in the aluminum test assembly shown in Fig. 1 located under 6 m of water in the N Reactor fuel storage basin. The fuel was loaded into the test assembly in aluminum tubes (7.62-cm o.d.) containing two (53-cm-long) fuel elements. The fuel tubes were held in a hexagonal lattice arrangement with a pitch of 8.13 cm.

The reactivity of the fuel loadings was monitored with three detectors: one proportional counter and two fission chambers. For spent-fuel measurements, three fission chambers were used. Approach-to-critical and pulse-neutron source data were recorded during loading and unloading of fuel tubes. A k -eff value of 0.94 was obtained with a loading of 54 tubes of unirradiated fuel, compared with a value of 0.95 obtained with 91 tubes of fuel irradiated to an average exposure of 3020 MWd/MT. The loadings were limited to a maximum k -eff value of 0.97 established as the test safety limit.⁴

Preliminary calculations using the HAMMER code⁵ agree with the experimentally determined k -eff and critical mass values. The measured ratio of irradiated-to-unirradiated fuel critical mass of 2.0 compares with the calculated value of 2.0 and confirms that application of unirradiated fuel critical mass limits to spent-fuel operations introduces undue conservatism. Updating criticality control limits based on the test results will permit maximum utilization of spent-fuel storage space⁶ and will expedite irradiated fuel handling operations.

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Criticality Calculations on Maine Yankee Spent-Fuel Racks Containing Boral, D. J. Denver, B. F. Momsen, E. E. Pilat (YAEC)

Due to the delays in the availability of reprocessing, many nuclear power plant operators have had to contend with a shortage of on-site storage space for spent-fuel assemblies. This paper describes the criticality calculations that were performed in support of the licensing effort for the increased capacity Maine Yankee spent-fuel storage racks which utilize Boral as a fixed absorber, thereby reducing the nominal assembly pitch from 20.5 to 12.0 in. and increasing the available storage space by a factor of 3.

The rack employs two 0.174-in. Boral sheets between each row and each column of assemblies. The Boral sheets are composed of a 0.084-in. core of 35 wt% natural B_4C in Al_2O_3 , clad with 0.045 in. of aluminum. The two Boral sheets between assemblies are held in place, 2.714 in. apart, by a series of 0.125-in. aluminum channels, which also serve to confine the 8.12-ft-sq fuel assembly when it is in the rack. The mechanical design of the racks has been described by Allison¹ and thermal-hydraulic considerations have been discussed by Metevia et al.²

Extensive criticality calculations have been done for this rack to ensure that it meets the applicable criteria³ under all conditions. The calculations were performed using a four-energy-group, X-Y representation of the racks in the PDQ-07 (Ref. 4) program. Cross sections for the PDQ-07 model were obtained from LEOPARD (Ref. 5) super cell calculations, which also yielded fast and thermal spectra for use in RODWORTH (Ref. 6), a blackness theory code which was modified to account for boron particle self-shielding. The resulting cross sections were then used in a PDQ-07 calculation which explicitly modeled the system geometry.

The accuracy of the PDQ-07 model was checked in two ways: (a) the model was used to calculate one of the unpoisoned critical experiments described below; and (b) the PDQ-07 analysis of the Maine Yankee spent-fuel

TABLE I
Results of Criticality Calculations

Description of System	Method	$K_{eff} \pm$ KENO Standard Error
1. Critical lattice, 203 rods ^a	KENO-II	1.0103 $\pm$ 0.0062
2. Same System as 1	PDQ-07	0.9808
3. Critical lattice, 245 rods with large depleted U block adjacent to lattice ^b	KENO-II	1.0221 $\pm$ 0.0068
4. Critical lattice, 324 rods with depleted U block adjacent to lattice and Boral sheet through lattice ^c	KENO-II	1.0175 $\pm$ 0.0063
5. Maine Yankee spent-fuel racks (68°F)	PDQ-07	0.7558
6. Same System as 5	KENO-II	0.7821 $\pm$ 0.0057
7. Maine Yankee spent-fuel racks (150°F)	PDQ-07	0.7447
8. Maine Yankee spent-fuel racks (68°F) with most reactive pitch allowed by fuel rack walls	PDQ-07	0.7867
9. Maine Yankee spent-fuel racks (68°F) with one Boral plate missing	PDQ-07	0.8317

^a4.95 wt% ^{235}U metal rods, 0.762-cm o.d., 30 cm long.

^bDepleted uranium block was a 640-kg casting of 0.185 wt% ^{235}U .

^cBoral sheet consisted of 0.1689-in.-thick B_4C in Al_2O_3 (38.9 wt% B_4C , 0.0949 lb/in.³) clad with 0.041-in. Al.

rack in the nominal condition was compared with a similar analysis performed using the Monte Carlo program KENO-II (Ref. 7). Additionally, the KENO-II model, which utilized the Knight modified 16-group Hansen-Roach cross-section set,⁸ was verified by analyzing three critical experiments (one containing Boral) from a series of unpublished criticals performed by Johnson at the Oak Ridge Critical Experiments Facility.⁹ The experiments were performed with unclad 4.95 wt% ²³⁵U metal rods that were 0.762 cm in diameter and 30 cm long. The rods were placed in a uniform lattice having a pitch of 2.05 cm. The results of the calculations are shown in Table I.

The PDQ-07 model was also used to examine off-nominal fabrication and pool conditions. To ensure conservatism and satisfy licensing requirements, the fuel was assumed to be undepleted, to have a higher initial enrichment than anticipated, to contain no fixed burnable poison, and to be moderated by water containing no soluble poison. Off-nominal situations investigated include those with different pool temperatures, potential missing Boral sheets, more reactive fuel assemblies, and other abnormal configurations. In no case did the system multiplication factor exceed the self-imposed criterion of 0.90.

In summary, an effective spent-fuel rack, which tripled the spent-fuel storage capacity at Maine Yankee, was designed and analyzed using conventional methods that are generally available.

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Tutorial Session: Theoretical Developments in Nuclear Criticality Safety and Risk Analysis

Probabilistic Safety Analysis of Fuel Cycle Risks, R. C. Erdmann, R. R. Fullwood, J. Maly (SAI)

The intent of this review is to indicate how the methods of probabilistic safety can be applied to the analysis of potential accidents in the nuclear fuel cycle, excluding reactors. There has been considerable effort spent in analyzing selected aspects of the fuel cycle from a probability viewpoint, but there has been little work done to analyze the complete cycle with the intent of either balancing the risks from the various components or quantifying the total risk.

In reactor safety studies of this nature, an attempt is made to construct a table or plot of accident consequences (isotope amounts released) vs likelihoods. The basis for this work began with Farmer,¹ was continued in studies such as that by Otway and Erdmann,² and has recently been utilized by the Atomic Energy Commission,³ with substantial methods development. A similar approach was used for a reprocessing plant safety analysis⁴ for a mixed-oxide fuel fabrication plant,⁵ in the transportation of nuclear materials,⁶ and in considerations of waste storage.⁷

The predominant method of evaluation in these studies has been fault tree analysis. The Reactor Safety Study³ also utilized event tree analysis to conveniently document accident sequences and to link the subsystem fault trees into a plant analysis. Failure Mode and Effects Analysis, used extensively in fast reactor safety, is the recommended method for preliminary analysis.⁸ All of these methods have application to the analysis of the fuel cycle including the problems of safeguarding special nuclear material.⁹

To evaluate fuel cycle risks and their composition, it is first necessary to define the fuel cycle. This will be variable in time due to changing constituents in the fuel composition as well as in the geographical location of the cycle plants and in the modes of shipment. Decisions on the proximity of waste storage sites to reprocessing plants will impact transportation risks. Plutonium recycle and spiking of reprocessed fuel can also impact the magnitude of potential accident risk.

Whatever the industry composition in the years ahead, however, it is expected that the consequence/probability approach to accident quantification will provide a unified method for balancing risks. Of course this requires as much information as can be made available regarding plant and equipment design and operational procedures. The more one extrapolates from known to inferred systems, the less certainty one has in the results.

From the definition of the fuel cycle(s), one can identify key accident sequences, appropriate time scales, and the types of radiological results expected. Both internally as well as externally induced events should be included in this type of work and the material distribution

throughout the cycle will tend to emphasize certain aspects of the cycle over others. Usually this type of analysis is aimed at determining involuntary risk but it can be used to help estimate voluntary risk as well.

Nuclear reactors contain large amounts of stored energy which, if uncontrolled, drives the accident and provides the dispersal mechanism in the much-studied blowdown accidents. The other parts of the fuel cycle do not have this stored energy, and other mechanisms such as criticality, fire, uncontrolled chemical reactions, transportation collisions, and long-term fission product motion require careful investigation. From the studies done to date, it appears that the fuel cycle can and is being developed in a safe way. However, it is unclear at present how the dominant contributors to fuel cycle risk are distributed among the constituent parts.

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Cross Sections for Criticality Safety Computations, H. K. Clark (DuPont-SRL)

Advances in cross-section processing have made criticality safety computations increasingly complex. The criticality safety analyst must select a suitable processing scheme and describe it well.

The proposed ANSI Standard for the Validation of Computational Methods for Nuclear Criticality Safety defines a Method as "the mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections), and calculational procedures which yield the calculated results." The proposed Standard further states, "Nuclear parameters such as cross sections should be consistent with experimental measurements of these parameters." Care clearly is required in specifying what constitutes a Method. It is obviously not acceptable merely to specify, say, KENO.¹

But neither is it acceptable to specify KENO and ENDF/B-III or KENO and Hansen/Roach.² ENDF/B cross sections must be averaged in some manner to yield the multigroup cross sections required by KENO. How this operation is performed may cause significant differences in the multiplication constant calculated for a system.³ With Hansen-Roach cross sections, one must specify the method for selecting resonance absorption and fission cross sections, the cross-section tables selected for hydrogen, whether ²³⁵U cross sections are those as modified at Oak Ridge by Knight, and how nuclides not in the Hansen-Roach set are treated.

In the early days of criticality safety computations, when a two-group diffusion theory calculation in cylindrical or spherical coordinates by desk calculator was a tedious and somewhat formidable task and when cross-section data were more sparse, the selection of cross sections was perhaps a simpler task. The four-factor formula was widely used for moderated systems. A factor was used to indicate the deviation from $1/v$ behavior of an absorption cross section at thermal energy. Thermal and epithermal cross sections were related to the integral parameters, diffusion area, and neutron age.

With advances in techniques for measuring cross sections and with increases in computer speed and core size, cross sections could be treated in greater detail and more exactly. Computer codes became more complex. Many options were introduced. Instead of a single thermal group, the range over which a neutron might gain as well as lose energy in a collision was divided into many groups. The epithermal range was similarly divided, and group skipping was permitted. Inelastic scattering, anisotropic elastic scattering, and resonance absorption were treated more exactly.

Such developments do not always improve the agreement with integral experiments. Where agreement is poor, cross sections can be adjusted within the limits of experimental error to produce agreement. Agreement does not only mean agreement in terms of the multiplication constant, but also in terms of various flux ratios. Producing such agreement is likely to be difficult when the number of groups is large and when the range of compositions is wide. The alternative is to accept the bias produced by the cross sections and the computational technique. Fortunately, in the criticality safety field, the chief concern is the bias in k_{eff} , or how much k_{eff} calculated for a critical experiment deviates from unity.

The increased complexity of cross-section processing presents the criticality safety analyst with a real problem. He may not be, and really should have no need to be, well versed in reactor theory. On the other hand, because the processing is performed by computer, he is able to "push a set of buttons" that will give him an answer. With the various options available, however, this may not be the only set, and hence not the only answer he can get for a particular configuration of fissile material. He must take care, through advice from experts, that he pushes a proper set and that he uses this same set both for establishing bias and for performing the criticality safety evaluation.

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Comments on the Use of the Monte Carlo Method for Criticality Calculations, G. E. Whitesides (ORNL)

As evidenced by recent papers given at Nuclear Criticality Safety Division meetings, the use of the Monte Carlo method has become a very popular computational tool. The ease of use has undoubtedly been a primary reason for this popularity. This ease of use, however, may lead to a false sense of security when using the method. It is the purpose of this paper to offer some guidance on the effective use of the method and to provide some suggestions on how to avoid some of the pitfalls that can occur.

To minimize the statistical error per unit computer time, a number of modifications are made to the analog Monte Carlo procedure. In almost all cases these modifications can have "side effects" that may be undesirable and in some cases may lead to erroneous conclusions.

The most common technique used to minimize the error per unit time is the use of weighted tracking. When using weighted tracking, neutrons are not allowed to die by absorption. Instead, an initial weight is assigned, and the weight is reduced at each collision site by the absorption probability. In this procedure (described more fully in Ref. 1), a weight, which we shall call WTLOW, is chosen. When the neutron falls below this weight, Russian Roulette is played to determine if it dies or if it survives with a weight WTSUV. The choice of WTLOW and WTSUV has been shown to exhibit a minimum for a given combination of values.¹ The optimum choice of the value of the variables depends on a number of factors. Fortunately, the "side effects" of choosing values of WTLOW between 0.02 and 0.3 and WTSUV between 0.2 and 1.0 seem only to be a variation in the computed statistical error and almost no effect on the computed k_{eff} . Unfortunately, when using adjoint biasing,² values of WTLOW and WTSUV may be assigned outside this "safe" range for certain portions of the system being computed. The purpose of adjoint biasing is to minimize the computer time spent tracking neutrons in regions of low importance. The side effect observed when using the method correctly is the failure to accurately compute effects such as absorptions and leakage in these regions of low importance. By using values of WTLOW and WTSUV that are too high in the regions of low importance, the effect almost always will be to compute a k_{eff} that is too low. Care should be taken to stay on the low side when choosing and applying adjoint biasing parameters.

In the original paper on the use of differential albedos,³ a tendency of the method to overestimate k_{eff} in cases where the reflector face dimensions were too small was reported. Long,⁴ in a later paper, pointed out that the error can be quite large, particularly for arrays in which the fissile material touches the reflector face. Where applicable, the differential albedo can be a very effective saver of computer time. The side effect of this method has been to overestimate k_{eff} for all cases observed. Theoretically, the case where this is not true may exist. When beginning a study using a material and configuration whose behavior is unfamiliar to the user, a safe practice would be to check the use of the differential albedo against actual tracking for a few representative cases.

Another area of concern when using the Monte Carlo method is determining whether adequate sampling in important regions has occurred. In a paper dealing with this subject,⁵ it was pointed out that it is possible to compute the wrong answer with no hint of trouble if only the k_{eff} 's as a function of generation are observed. While

no foolproof method to prevent this from occurring can be described for the general case, the user can minimize the failure to recognize the problem by observing activities, such as fission densities, in localized sections of the system. Significant differences in the fission density between regions of the system, particularly large regions with low fission densities in a system with a small region with a high fission density, should be examined carefully for adequate sampling. Limiting the problem description to look at only a portion of the system often can provide valuable information.

An unresolved difficulty with Monte Carlo calculations which continues to cause concern is the inability to compute accurately the error estimates for the differential quantities (such as flux, fission densities, etc.) as a function of region and energy group.⁶ While there is no indication of error in computing the differential quantities themselves, the standard assumption when computing the statistical error that the "sample estimates" are independent is often not valid. To be correct, the statistical error calculation must take into account the correlation between "sample estimates." There is currently no general method to do this. While research on this problem continues, error estimates computed by standard techniques should be used with caution.

The Monte Carlo method provides the criticality safety specialist a rigorous, easy-to-use technique for evaluating many problems. A good understanding of the method and its limitations is essential if the user is to escape the pitfalls, which can lead to erroneous results. Undetected, these erroneous results could lead to erroneous safety recommendations.

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Remarks on Surface Density and Density Analog Representation of Array Criticality, J. T. Thomas (ORNL)

The body of information on the criticality of sub-critical components of fissile materials arranged in reflected critical arrays has grown sufficiently in the past ten years to warrant examination of density techniques and understanding of concepts employed in nuclear criticality safety. Monte Carlo calculations of experimental data have permitted valid extensions of these limited data. Correlations¹ of the data have been effected and have resulted in an analytic expression relating the total number of units, N , in an array, the edge dimension,

d , of cubic cells, and the mass, m , of the fissile material centered in a cell. The mass necessary for criticality of such systems satisfies the equations

$$\sigma(m) = c_2 (m_0 - m) = \frac{nm}{d^2} (1 - c/\sqrt{N})^2 \quad (1)$$

In these relations, $\sigma(m)$ is a limiting surface density ($g \cdot cm^{-2}$); m_0 is the unreflected critical mass in the geometry of the unit; n^3 is N ; c is a constant characterizing the geometry of center spaced units and equals 0.55; c_2 is a constant dependent on the type of fissile material and is influenced by the unit shape, by the array shape, and by the array reflector material.

For cuboidal arrays of cubic cells, the mass m required for criticality satisfies the equation

$$\left(\frac{m_0}{m} - 1\right) = \left[\frac{11.924(\sqrt{N} - c)}{nd}\right]^2 (5R^{-0.672} - 1) \quad (2)$$

where the array shape is represented by the parameter

$$R = \frac{N^{1/3}}{3} \sum_{i=1}^3 \frac{1}{n_i} \quad \text{and } n_i \text{ is the number of units along the}$$

three directions of the reflected cuboidal array. The value of R should not exceed 5.34. The definition of criticality by these equations has been determined as conservative below $m = 0.1 m_0$ and as very good for greater values of m representing criticality to within 1% in k_{eff} .

The density analog^{2,3} representation of reflected array criticality is usually expressed in the form $M = \text{const} \times \rho^{-s}$. The power s is dependent on the size of the array approaching the value 2 as a limit and has been related to the mass of the unit. The above relations also may be expressed in this form. Cubing Eqs. (1) and (2) and rearranging the terms to yield

$$N(1 - c/\sqrt{N})^6 = \frac{[c_2(m_0 - m)]^3}{m} \frac{1}{\rho^s} \quad (\text{cubic}) \quad (3)$$

Fig. 1. Density analog representation of reflected array criticality for 20 kg U(93.2) metal spherical units.

and

$$N(1 - c/\sqrt{N})^6 = \frac{1}{(11.924)^6 m} \left(\frac{m_0 - m}{5R^{-0.672} - 1} \right)^3 \frac{1}{\rho^2} \text{ (cuboidal). } (4)$$

Application of Eqs. (3) and (4) to 20 kg U(93.2) metal spheres at a density of $\rho_0 = 18.76 \text{ g-cm}^{-3}$ for which c_2 has been evaluated as $1.762 \times 10^{-3} \text{ cm}^{-2}$, results in the relations shown in Fig. 1. The shape of the cubic arrays is typical of that obtained from experimental and calculated data displaying the total mass ($M = Nm$) as varying inversely with density to a variable power. The data are equally well displayed as the total mass [$M' = M(1 - c/\sqrt{N})^6$] varying inversely as the square of the density. The departure from cubic arrays is illustrated for arrays having variable shape, R, maintaining 2 cells in the vertical direction, beginning with an 8-unit array. For the same numbers of units in arrays, changing from cubic shape requires higher average densities. The arrays with constant noncubic shape were chosen to correspond to the shape of a reflected critical slab of U(93.2) metal.⁴ The latter example forcefully illustrates that as the average density approaches and achieves the fissile material density, the defined critical configuration is determined by the array shape and not the shape of the units in the array.

It may be said that the density methods can give complementary interpretations of criticality. The two constants, one for geometry and the other for fissile material as spherical units in arrays, are sufficient to represent criticality as

1. a continuous function of N
2. more properly depicting the total mass, or N, as varying inversely as the square of the density
3. not being limited to cubic arrays
4. capable of approximating reflected and unreflected single-unit criticality for cuboidal shapes.

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Special Criticality Considerations for Low-Enrichment Fuel Processing, Robert L. Seale, Charles Verdon (U of Ariz)

The criticality limits for a large dissolver tank containing nitric acid as used in the reprocessing of LWR fuel have been examined over a wide range of dissolved uranium concentrations. Some potentially autocatalytic criticality situations have been identified in processing low-enrichment fuel.

The chemistry of UO_2 in nitric acid determines the total uranium dissolved; the enrichment of the dissolver determines the criticality characteristics of the dissolver. For 5% enriched UO_2 in a 100-cm-diam tank, the critical height is plotted in Fig. 1 as a function of ^{235}U concentration. At a typical dissolved uranium concentration of $\sim 3 \text{ M}$, the ^{235}U concentration is $\sim 15 \text{ g/liter}$. In a 100-cm-diam tank, this corresponds closely to the asymptotic infinite length critical concentration. For a 127-cm-high solution inventory (1000 liters), the system is subcritical.

As fuel dissolves, heat is liberated in the solution as follows:

- $\sim 75,000 \text{ cal/MT}$ due to UO_2 dissolving;
- $\sim 17,500 \text{ cal/MT}$ due to fission heat decay.

For 5% enriched fuel at $15 \text{ g } ^{235}\text{U/liter}$ concentration, the heat release is $\sim 28 \text{ cal/liter}$. Thus, for dissolvers operated at 80 to 85°C , cooling must be maintained to prevent boiling.

Mechanisms have been identified by which a dissolver could be "forced" into an autocatalytic critical mode.

These mechanisms include:

1. Overbatching of the UO_2 inventory by 40% would increase the ^{235}U concentration to $\sim 21 \text{ g/liter}$, a critical condition when 1000 liters are contained in the vessel.
2. Failure of cooling and solution boiling to reduce solution height while increasing ^{235}U concentration to reach criticality.

Either mechanism will result in reaching the critical condition at a concentration-height combination such that the energy released due to fissions will boil off more solution and increase the uranium concentration even further.

Both situations are shown by the dotted lines on Fig. 1. Each results in a supercritical solution which expands to

Fig. 1. Critical height vs ^{235}U solution concentration for a 100-cm-diam dissolver tank. Boiloff paths for constant inventory (a) and 40% fuel overload (b) incidents are shown.

reduce density until the just-critical condition holds. Gas evolution, which is proportional to power level, is the primary factor in reducing the solution density. Power level traces to achieve the total yield, for 1000 liters of solution, of $\sim 10^{20}$ fissions have been computed.

To preclude the possibility of these mechanisms being realized, several options are available, including:

1. redundant density control to limit uranium concentration
2. use distributed poison in the dissolver to reduce the system multiplication.

It should be noted that the 5% enrichment used in the calculations is extreme and would only be approached if unirradiated fuel were reprocessed.

Analysis of Integral Experiments

Analysis of Mixed-Oxide-Fueled Critical Experiments Using ENDF/B-IV Data, H. S. Cheng (BNL)

Accurate prediction of criticality is essential not only for the quality design of plutonium recycle cores but also for the safety of handling plutonium assemblies. To assess the BNL capability in this regard, multigroup cross sections based on the latest nuclear data file, ENDF/B-IV, were used in conjunction with the lattice analysis code, HAMMER,¹ and the two-dimensional (2-D) multigroup discrete ordinate code, TWOTRAN,² to analyze a series of mixed-oxide-fueled critical experiments.^{3,4} These critical experiments which covered a wide range of water-to-fuel ratios and ²⁴⁰Pu compositions represent a set of excellent benchmarks for validation of both the analytical methods and nuclear data for criticality prediction of mixed-oxide fuel assemblies typical of plutonium recycle cores in LWRs.

The analysis of these experiments was carried out by TWOTRAN in (r,z) geometry using a four-group and S_4 approximation with transport-corrected isotropic scattering. The experimental assembly was represented by a finite cylinder surrounded by the radial and axial reflectors with a vacuum boundary condition specified at the outside edges of the reflectors. The effective radius of the experimental assembly was calculated from the numbers of fuel rods required for achieving criticality. The radial and axial reflector thicknesses were determined from the measured radial and axial bucklings, respectively. The macroscopic cross sections of the fuel region were obtained from the homogenized cell-average cross sections of the infinite lattice cell calculations by HAMMER, and the reflector cross sections are those of an infinite medium of pure or borated water at room temperature generated by HAMMER using the Haywood kernel. The effect of epithermal upscattering on the 1-eV ²⁴⁰Pu resonance absorption is important and has been taken into account by a simple approach proposed by Buslik.⁵ All the criticality calculations were performed in fine meshes on the order of one mean-free-path or less.

The results of the k_{eff} calculations are summarized in Table I along with the estimated experimental uncertainties on criticality. The uncorrected values are the results of direct TWOTRAN calculations without any correction applied. These results indicate that the TWOTRAN-HAMMER-ENDF/B-IV calculational system is capable of predicting the criticality of the mixed-oxide-fueled critical experiments to within about 1%. Further improvement is possible, if corrected for the effects of spacer grids, ²⁴¹Pu decay, resonance overlap, and finite plutonium grain sizes. Using the Δk corrections calculated by Goldstein and Windsor⁶ for these same

experiments, the corrected values of k_{eff} were determined. The corrections result in a small standard deviation of $\pm 0.36\%$ in k_{eff} for water-to-fuel ratios of 1.1 to 8.3 and a maximum discrepancy of only 0.49%. This is excellent agreement. It should be noted, however, that the corrections do slightly worsen the accuracy in k_{eff} of the tightest lattices ($H_2O/fuel = 1.1$) indicating that the TWOTRAN-HAMMER-ENDF/B-IV system tends to underpredict the criticality of tight lattices by a few-tenths of one percent. This is thought to be due to the approximations of the Wigner-Seitz cell and isotropic scattering used in HAMMER.

The effect of using more energy groups (8 with 5 thermal groups) and higher S_n orders (S_4 and S_{16}) in TWOTRAN tends to reduce k_{eff} toward unity, each by about $0.1 \approx 0.2\%$, which, although within the experimental uncertainty, does improve the accuracy in k_{eff} .

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TABLE I

Accuracy in Criticality Prediction by the HAMMER-TWOTRAN System
Using ENDF/B-IV Data for the Mixed-Oxide-
Fueled Critical Experiments

Case Number	Fuel Type	Pitch (in.)	Boron (ppm)	H ₂ O/Fuel	Experimental k_{eff}	TWOTRAN			
						Corrected		Uncorrected	
						k_{eff}	$\Delta k/k$ (%)	k_{eff}	$\Delta k/k$ (%)
1	8% ²⁴⁰ Pu	0.69	0	1.10	1.0±0.0032	0.9951	-0.49	0.9969	-0.31
2	8% ²⁴⁰ Pu	0.69	261	1.10	1.0±0.0020	0.9963	-0.37	0.9977	-0.23
3	8% ²⁴⁰ Pu	0.69	526	1.10	1.0±0.0020	0.9962	-0.38	0.9973	-0.27
4	8% ²⁴⁰ Pu	0.75	0	1.53	1.0±0.0026	0.9985	-0.15	1.0024	+0.24
5	8% ²⁴⁰ Pu	0.9758	0	3.45	1.0±0.0022	1.0040	+0.40	1.0088	+0.88
6	8% ²⁴⁰ Pu	0.9758	261	3.45	1.0±0.0021	1.0024	+0.24	1.0064	+0.64
7	8% ²⁴⁰ Pu	0.9758	526	3.45	1.0±0.0017	1.0008	+0.08	1.0043	+0.43
8	8% ²⁴⁰ Pu	1.0607	0	4.36	1.0±0.0030	1.0036	+0.36	1.0096	+0.96
9	8% ²⁴⁰ Pu	1.380	0	8.25	1.0±0.0008	0.9989	-0.11	1.0044	+0.44
10	24% ²⁴⁰ Pu	0.9758	0	3.45	1.0±0.0029	1.0047	+0.47	1.0115	+1.15
11	24% ²⁴⁰ Pu	1.0607	0	4.36	1.0±0.0019	1.0040	+0.40	1.0118	+1.18
12	16% ²⁴⁰ Pu	0.7470	0	1.64	1.0079	---	---	1.0080	+0.01

Calculations of Fast Critical Assemblies Using ENDF/B-IV Cross-Section Data, T. J. Hiron, R. E. Seamon (LASL)

The purpose of this paper is to present calculational results of several fast critical assemblies using ENDF/B-IV cross-section data. The principal materials studied include ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , Be, Fe, and W. A number of the fast critical assemblies that have been developed at Los Alamos¹⁻³ were used in the calculations. These assemblies are summarized in Table I. Calculations of many of these assemblies using earlier cross-section evaluations are described in Refs. 4 through 8. Integral quantities calculated in this present study include eigenvalues and central reaction ratios.

All calculations were performed using one-dimensional (1-D) transport theory; i.e., the DTF-IV code in spherical geometry. Appropriate correction factors were applied for anisotropic scattering and angular quadrature effects. Anisotropic scattering effects for many of these fast critical assemblies are discussed in detail in Ref. 8. Multigroup cross-section data were processed using the MINX code.⁹

The results of the k_{eff} calculations, after all correction factors have been applied, are given in Table I. With the exception of the larger thickness natural-uranium-reflected U and Pu cores (Assembly 4 for each of these cases) and the W-reflected U core, all calculated eigenvalues are unity to within approximately one-half percent. In Table II, calculated central reaction ratios for Godiva and Jezebel are compared with experimental values. The ratios calculated with ENDF/B-IV cross-section data give generally good agreement, particularly for the capture-to-fission ratio for ^{238}U ($^{238}\text{U}_C/^{238}\text{U}_F$). The calculated $^{238}\text{U}/^{235}\text{U}$ fission ratio in Jezebel is about 9% lower than the experimental value.

These calculations indicate that the ENDF/B-IV cross sections (a) accurately predict criticality for a large number of fast critical assemblies, and (b) give very good agreement with experiment for the central reaction ratio

TABLE II
Central Reaction Ratios

Reaction Ratio	Calculation	Experiment
<u>Godiva</u>		
$(^{238}\text{U}/^{235}\text{U})_F$	0.161	0.156 ± 0.005
$(^{239}\text{Pu}/^{235}\text{U})_F$	1.38	1.42 ± 0.03
$(^{238}\text{U}_C/^{238}\text{U}_F)$	0.47	0.47 ± 0.02
<u>Jezebel</u>		
$(^{238}\text{U}/^{235}\text{U})_F$	0.186	0.205 ± 0.008
$(^{239}\text{Pu}/^{235}\text{U})_F$	1.39	1.49 ± 0.03

rates in Godiva and Jezebel. Additional assemblies will be run in an attempt to identify any serious discrepancies in the ENDF/B-IV data or processing thereof.

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TABLE I
Critical Assembly Calculations

Assembly	Reference	k_{eff}
<u>Uranium Cores</u>		
1. Godiva	Ref. 3	1.001
2. 1.735 cm U(N)-refl	Ref. 3	1.003
3. 4.425 cm U(N)-refl	Ref. 3	1.006
4. 9.982 cm U(N)-refl	Ref. 3	1.008
5. 4.699 cm Be-refl	Refs. 1,2	1.001
6. 5.08 cm Fe-refl	Refs. 1,2	1.001
7. 5.08 cm W-refl	Refs. 1,2	1.008
<u>Plutonium Cores</u>		
1. Jezebel	Ref. 3	0.997
2. Dirty Jezebel	Ref. 3	0.999
3. 4.128 cm U(N)-refl	Refs. 1,2	1.001
4. 11.684 cm U(N)-refl	Refs. 1,2	1.011
5. 3.688 cm Be-refl	Refs. 1,2	1.001
6. 4.699 cm W-refl	Refs. 1,2	1.003
7. 1.664 cm Oy-refl	Ref. 3	0.998

Benchmark Testing Using ENDF/B-III and -IV, R. D. McKnight (ANL)

A series of benchmark calculations of critical experiments has been performed to assess the effects that recent changes in the ENDF/B data files have had on calculated LMFBR parameters. Three well-documented critical assemblies were studied using standard methods of fast reactor analysis [two-dimensional (2-D) multi-group diffusion theory] with both Versions III and IV of ENDF/B. A review of the changes in the principal cross sections incorporated in the latest evaluation has been made and was used to interpret the changes in calculated integral parameters.

The three critical assemblies included in this study were ZPR-6 Assemblies 6A and 7 (Ref. 1) and ZPR-3 Assembly 48 (Ref. 2). These three benchmark assemblies provide a range of reactor sizes, fuel types, and neutron

spectra (Assembly 7 being similar in composition to a large LMFBR). The method of analysis was consistent with an earlier study by Zolotar et al.³ The ETOE-2/MC²-2/SDX cross section processing codes⁴ were used to generate space-dependent cross sections for each assembly. The differences in the values reported in this study (as compared to Ref. 3) using Version III data reflect modifications in the cross-section processing codes, whereas the differences in the values reported herein using Versions III and IV reflect modifications in the basic nuclear data.

The principal results are shown in Table I. The calculated eigenvalues exhibit a negative bias of 0.9 to 1.6% δk , with only slight changes (0.0 to -0.4% δk) using Version IV data. Some of the principal cross-section modifications included in ENDF/B-IV were: decreasing σ_f^{25} (~2%), σ_f^{49} (~1 to 1.5%), σ_c^{28} (~3 to 4%) and increasing σ_f^{239} (~1 to 1.5%). The changes in calculated eigenvalues reflect these data

TABLE I
Calculated and Measured Integral Results for Three Benchmark Critical Assemblies

	Calculated Results*			Experiment ^a
	Version 3	Version 4	V.4/V.3	
<u>ZPR-6 Assembly 7</u>				
k_{eff}	0.9861	0.9844	0.998	1.00
β_{eff}	3.2184(-3)	3.3434(-3)	1.039	-
Prompt Neutron Lifetime, sec	5.1532(-7)	5.2580(-7)	1.020	-
Conversion Factor, $I_h/\% \delta k$	1011.2	958.6	0.948	-
$^{25}f/^{49}f$	1.1162	1.1099	0.994	1.061 ± 0.021
$^{28}c/^{49}f$	0.1547	0.1524	0.985	0.1400 ± 0.0028
$^{28}f/^{49}f$	0.02274	0.02344	1.031	0.02336 ± 0.00047
$^{49}c/^{49}f$	0.2790	0.2852	1.022	-
Central $\left\{ \begin{array}{l} ^{239}Pu \\ ^{235}U \end{array} \right.$	$\left\{ \begin{array}{l} 197.9 \\ 166.2 \end{array} \right.$	$\left\{ \begin{array}{l} 184.7 \\ 154.1 \end{array} \right.$	$\left\{ \begin{array}{l} 0.934 \\ 0.927 \end{array} \right.$	$\left\{ \begin{array}{l} 155.9 \pm 2.0 \\ 133.3 \pm 3.0 \end{array} \right.$
Worths, $\left\{ \begin{array}{l} ^{238}U \\ ^{23}Na \end{array} \right.$	$\left\{ \begin{array}{l} -12.57 \\ -8.733 \end{array} \right.$	$\left\{ \begin{array}{l} -11.15 \\ -8.752 \end{array} \right.$	$\left\{ \begin{array}{l} 0.887 \\ 1.002 \end{array} \right.$	$\left\{ \begin{array}{l} -10.91 \pm 0.46 \\ -6.205 \pm 0.315 \end{array} \right.$
I_h/kg	$\left\{ \begin{array}{l} ^{10}B \\ -3424 \end{array} \right.$	$\left\{ \begin{array}{l} -8.752 \\ -3241 \end{array} \right.$	$\left\{ \begin{array}{l} 0.947 \\ 0.947 \end{array} \right.$	$\left\{ \begin{array}{l} -2926 \pm 58 \\ - \end{array} \right.$
Δk_{het}	0.0161 ^b	0.0161 ^b	-	-
<u>ZPR-6 Assembly 6A</u>				
k_{eff}	0.9937	0.9893	0.996	1.00
β_{eff}	6.9538(-3)	7.0835(-3)	1.019	-
Prompt Neutron Lifetime, sec	5.1097(-7)	5.3599(-7)	1.049	-
Conversion Factor, $I_h/\% \delta k$	452.0	445.7	0.986	-
$^{25}f/^{49}f$	1.0344	1.0239	0.990	-
$^{28}c/^{49}f$	0.1469	0.01447	0.985	-
$^{28}f/^{49}f$	0.02352	0.02397	1.019	-
$^{49}c/^{49}f$	0.2819	0.2918	1.035	-
Central $\left\{ \begin{array}{l} ^{239}Pu \\ ^{235}U \end{array} \right.$	$\left\{ \begin{array}{l} 62.33 \\ 48.10 \end{array} \right.$	$\left\{ \begin{array}{l} 60.80 \\ 46.52 \end{array} \right.$	$\left\{ \begin{array}{l} 0.975 \\ 0.967 \end{array} \right.$	$\left\{ \begin{array}{l} 57.1 \pm 2.0 \\ 41.59 \pm 0.48 \end{array} \right.$
Worths, $\left\{ \begin{array}{l} ^{238}U \\ ^{23}Na \end{array} \right.$	$\left\{ \begin{array}{l} -4.225 \\ -0.8847 \end{array} \right.$	$\left\{ \begin{array}{l} -4.00 \\ -0.8324 \end{array} \right.$	$\left\{ \begin{array}{l} 0.946 \\ 0.941 \end{array} \right.$	$\left\{ \begin{array}{l} -3.29 \pm 0.01 \\ 0.160 \pm 0.040 \end{array} \right.$
I_h/kg	$\left\{ \begin{array}{l} ^{10}B \\ -1198 \end{array} \right.$	$\left\{ \begin{array}{l} -0.8324 \\ -1210 \end{array} \right.$	$\left\{ \begin{array}{l} 1.010 \\ 1.010 \end{array} \right.$	$\left\{ \begin{array}{l} -1231 \pm 54 \\ - \end{array} \right.$
Δk_{het}	0.0068 ^b	0.0081 ^b	-	-
<u>ZPR-3 Assembly 48</u>				
k_{eff}	0.9908	0.9911	1.000	1.000 ± 0.001
β_{eff}	3.3654(-3)	3.5237(-3)	1.047	-
Prompt Neutron Lifetime, sec	2.8707(-7)	2.9065(-7)	1.012	-
Conversion Factor, $I_h/\% \delta k$	982.9	934.4	0.951	-
$^{28}c/^{25}f$	0.1287	0.1268	0.985	0.137 ± 0.007
$^{28}f/^{25}f$	0.03083	0.03227	1.047	0.0326 ± 0.0016
$^{49}c/^{25}f$	0.2276	0.2321	1.020	-
$^{49}f/^{25}f$	0.9477	0.9547	1.007	-
Central $\left\{ \begin{array}{l} ^{239}Pu \\ ^{235}U \end{array} \right.$	$\left\{ \begin{array}{l} 551.8 \\ 423.8 \end{array} \right.$	$\left\{ \begin{array}{l} 511.8 \\ 389.6 \end{array} \right.$	$\left\{ \begin{array}{l} 0.927 \\ 0.919 \end{array} \right.$	$\left\{ \begin{array}{l} 445 \pm 5 \\ 339 \pm 4 \end{array} \right.$
Worths, $\left\{ \begin{array}{l} ^{238}U \\ ^{23}Na \end{array} \right.$	$\left\{ \begin{array}{l} -31.87 \\ -11.60 \end{array} \right.$	$\left\{ \begin{array}{l} -27.44 \\ -13.52 \end{array} \right.$	$\left\{ \begin{array}{l} 0.861 \\ 1.166 \end{array} \right.$	$\left\{ \begin{array}{l} -24.8 \pm 0.5 \\ -6.3 \pm 0.3 \end{array} \right.$
I_h/kg	$\left\{ \begin{array}{l} ^{10}B \\ -10081 \end{array} \right.$	$\left\{ \begin{array}{l} -13.52 \\ -9407 \end{array} \right.$	$\left\{ \begin{array}{l} 0.933 \\ 0.933 \end{array} \right.$	$\left\{ \begin{array}{l} -8926 \pm 60 \\ - \end{array} \right.$
Δk_{het}	0.0188 ^b	0.0188 ^b	-	-

*Two-dimensional (R-Z) 27-broad-group diffusion theory calculation.

^aTaken from Ref. 3.

^bObtained as difference between 2-D calculations using heterogeneous and homogeneous cross sections. The effects of anisotropic neutron diffusion (streaming) have not been included.

Fig. 1. Calculated neutron spectra for ZPR-6 Assembly 7.

modifications; viz, a smaller change in the Pu-fueled Assembly 7 than in the U-fueled Assembly 6A and no change in the smaller (harder spectra) Pu-C-fueled Assembly 48. Calculated values for β_{eff} increased (and the reactivity conversion factors decreased) with the new delayed-neutron data. This tends to reduce the central worth C/E discrepancy in these three assemblies. Another principal cross-section modification in the latest evaluation was a large (10 to 12%) reduction in $\sigma_{\text{inel}}^{238}$ above 1 MeV. The calculated neutron spectra for ZPR-6 Assembly 7 are shown in Fig. 1 and illustrate the effects these cross-section changes have on the calculated spectra for these benchmark assemblies. Using Version IV data, the spectra tend to flatten slightly with less flux in the 100- to 300-keV peaks.

In general, improved agreement between measured and calculated integral parameters has been obtained with the new ENDF/B evaluation. The overprediction of $^{238}\text{C}/^{49}\text{f}$ is generally reduced (~5 to 15%); and little change occurs in the calculated eigenvalues which remain ~1.0 to 1.5% δk low.

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Experiments and Calculational Methods for Evaluating Interaction in Fissile Material

History of Fissile Array Measurements in the U.S. and Comments on the Current Experimental Program, *C. L. Schuske (AI-Colo), invited*

The initial impetus for experimental array measurements was, as one would expect, the processing, fabrication, and storage of highly enriched uranium and plutonium primarily as metal and aqueous solutions. A recent literature search¹ reveals that more than 90% of all array measurements were with these materials.

The first measurements with arrays of fissile metal² were performed at the Los Alamos Scientific Laboratory in 1947 and the first with fissile solutions³ at the Oak Ridge Critical Experiments Facility in 1949. Since then, there have been many other significant experiments at several U.S. laboratories including Rocky Flats, Battelle Pacific-Northwest Laboratory, and the Lawrence Livermore Laboratory.

Prior to 1963, Monte Carlo calculational methods had been applied to a variety of physics problems but, because of high computing costs and the lack of adequate cross-section data, they had not been extensively applied to criticality problems in the U.S. To the author's knowledge, the first known application of Monte Carlo calculations for designing experimental arrays was done by the LRL critical assemblies group in 1963. In spite of the lack of computational models and methods to help guide the early experimentalist, he did manage to perform experiments in a manner that gave the data considerable utility. Much of this pre-1960 data continues to provide guidance in the handling and storage of highly enriched uranium and plutonium. In addition, these early data gave rise to several interaction models which are still in use by some nuclear criticality specialists.

With the widespread acceptance of Monte Carlo methods to solve plant problems, the complexion of the experimental programs has changed. Prior to the early 1960's, experimental data alone provided the guidance for criticality criteria and, in most cases, the experiments were tailored to the direct solution of specific industrial problems. Today, however, with versatile Monte Carlo codes, fast machines, and improved cross-section data, the previous experimental approach would be wasteful of our resources. Hence, industrial problems are currently solved with the aid of computer codes, and the experimental data are needed principally to check calculational methods for accuracy and reliability.

Today's experimental data priorities have shifted from highly enriched fissile systems to those in the under 5% range which support the reactor fuels processing and fabrication industry.

To more specifically identify the needs of the nuclear industry, a meeting was recently attended at Rockwell International's Rocky Flats Plant by representatives of

fuel processors and fabricators, ERDA contractors, and nuclear regulatory personnel. This meeting led to a list of problem areas in which benchmark criticality measurements are needed. Table I (see following page) presents a list of the needs in the next 5 to 10 years and sets priorities as established by the potential users.

The Rocky Flats measurement program for the next several years includes the following:

1. A study of the effects of concrete and plastic enclosures around single and multiple vessel arrays containing enriched $\text{UO}_2(\text{NO}_3)_2$ solutions. The results of these measurements will be valuable in determining the adequacy with which current criticality codes evaluate the neutron reflector properties of concrete and plastic.

2. Series of measurements with low-enriched UO_2 (proposed 5 and 10% U enrichment) with H/U ranging from ~0.5 to 3. Projects 1 and 2 are priority 1 programs and are funded by the U.S. Nuclear Regulatory Commission. Copies of our progress reports can be obtained from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, Virginia, 22161.

3. A third program presently well under way involves arrays of 3-kg pure metal plutonium cylinders fully flooded and reflected with water. Up to 200 kg of Pu will be used in these arrays.

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The Monte Carlo Method for Array Criticality Calculations, *Deanne Dickinson (AI-Colo), invited*

Availability of suitable computer hardware and software has made possible the routine application of the Monte Carlo method to plant criticality safety studies. This method is especially useful in treating the interaction among different fissile units, a problem which can, at best, be treated approximately by one- or two-dimensional transport codes. Before Monte Carlo codes became practical, interaction problems were handled in a very conservative (and not always economical) manner by

TABLE I (paper 1)
List of Areas Where Benchmark Measurements Are Needed
(1, 2, 3 = Priority Ratings)

High Enriched Reactors	HTGR	LMFBR	Pu Recycle (Mixed Oxide)	Low Enriched Light Water
<p>1 Concrete reflected single units and arrays</p> <p>2 Concrete attenuator between units</p> <p>3 > 95% enrichment solution geometries</p> <p>4 Low density water in fuel arrays</p>	<p>1 Chemical processing and reclaim $N/U = 0-40$ $C/U = 0-100$</p> <p>2 Moist oxide and carbides $M/U^{235}U = 0-10$ $C/U^{235}U = 20-4000$</p> <p>3 $^{235}UO_2$ at low N/U ratio</p> <p>4 ^{235}U plus polymers (rings, sheets, etc.)</p> <p>5 Concrete reflected</p> <p>6 Pellet systems</p> <p>7 Arrays of $^{235}UO_2$ reflected by poly, concrete, lead</p> <p>8 Fabrication of fuel elements. Sintering of UO_2-impregnated beads or pellets. $N/U = 13-30$</p> <p>9 Carbon-coated U & Th particles for C/U ratio < 400</p> <p>10 ^{235}U-Th-C composites or fuel mixtures at Th/U ratios of 1, 4, and 80</p> <p>11 Criticality of HTGR fuel assemblies immersed in water</p> <p>12 Th-^{235}U mixtures in undermoderated range (slurry range); low N/U ratio</p> <p>13 Mixed oxides; $N/U = 0-40$ $C/U = 0-100$</p> <p>14 Interacting arrays of HTGR fuel elements</p> <p>15 Minimum critical concentrations for Th-^{235}U mixtures</p>	<p>1 Fuel elements dissolving in poison $[Gd(NO_3)_3]$ solutions</p> <p>2 $Pu-U$ mixtures of dilute solutions for Pu content in the 8-30 wt% range (define bounds at lower concentrations)</p> <p>3 $Pu-U$ nitrate solutions (8-30% Pu) composed of Pu with high ^{241}Pu content (~25 wt% ^{241}Pu) for delineating effects of ^{241}Pu on criticality</p> <p>4 Dry and damp ($N/Pu = U < 0.45$) sized oxides (8-30 wt% PuO_2 in natural U)</p> <p>5 Limiting critical enrichment of Pu in dry and damp ($N/Pu = U < 0.45$) mixed oxides</p> <p>6 Critical data on thicknesses of infinite slabs for aqueous solutions of $Pu-U$ with Pu content in the range 8-30 wt%; $^{241}Pu = 25$ wt%</p> <p>7 Dry mixed oxide geometries</p> <p>8 $Pu-U$ carbides, and $Pu-U$ nitrides at concentrations above 200 g $Pu-U/l$</p>	<p>1 Undermoderated mixed oxide array at reduced interstitial H_2O density</p> <p>2 Undermoderated sized oxide powder</p> <p>3 Poisoned fuel bundles & pellet arrays</p> <p>4 3-8% PuO_2 in natural U; $N/Pu = 0.5-1.5$ wt% H_2O</p> <p>5 10% ^{241}Pu, 25% ^{241}Pu systems</p> <p>6 Concrete reflection and isolation</p> <p>7 Aqueous solutions of U (3-8 wt%) and natural U; $N/Pu = 90-1500$</p> <p>8 Minimum Pu enrichment in natural U</p>	<p>1 Undermoderated UO_2 powder (3.0-5.0 g/ml) with 5000-200,000 ppm H_2O</p> <p>2 Poisoned arrays of fuel bundles or pellets (e.g. boral sheet)</p> <p>3 Undermoderated UO_2 rod lattices</p> <p>4 Er clad UO_2 fuel rods or bundles at reduced interstitial H_2O densities; $H_2O < 0.7$ g/cm³</p> <p>5 Effect of Al, Er, & SS cladding alloying on criticality</p> <p>6 Soluble poisons $[Gd(NO_3)_3]$ effects on pellet dissolution</p> <p>7 Rod dissolution</p> <p>8 Critical mass measurements on UO_2 pellets in water and low enriched uranium compounds (ADU)</p> <p>9 Concrete reflection or unit isolation vs H_2O effects</p> <p>10 V_2 measurements on low enriched low N/U systems; minimum critical N/U ratio for ~2% UO_2; minimum enrichment for criticality for dry UO_2</p> <p>11 Concentrated solutions with Raschig rings (standard permits only up to 400 g U/l)</p> <p>12 Reflected ^{235}U low density alloys</p>

several methods.¹ Monte Carlo codes can also use more detailed cross-section data without a significant increase in computing time, but this additional capability is usually more important for reactor calculations than for safety studies.

For the application to array calculations, the most important part of a Monte Carlo code is the geometry section. The general-purpose geometry routines used in current codes can be grouped into four categories: (a) nested-region; (b) combinatorial; (c) hole-routine (fictitious collisions); and (d) quadric surface. In addition, there are many special geometry routines that handle a limited geometry description efficiently. For the criticality specialist, an important geometry feature is an

array option, which permits the user to describe a unit once and use the unit in various positions in an array. Monte Carlo codes may be used to study the effect of rearranging units within an array or of placing moderating or shielding material between array units.

Although current Monte Carlo codes have very powerful geometry routines, there are still many practical and important systems that cannot be described without approximation. Two examples are a tank randomly filled with Raschig rings and a hexagonal array of fuel rod bundles.

The fact that Monte Carlo codes such as KENO IV² or MORSE³ can be applied easily to complex plant processing

and storage problems may cause the user to forget that the method is not free from complications. Some of the difficulties that may occur when using Monte Carlo are discussed in Refs. 4 and 5. It is a disturbing fact that a calculation that has been correctly described and run with appropriate cross-section data may give erroneous results with no apparent warning signals. For criticality safety calculations, it may be desirable to bias the initial keff's on the high side by intentionally starting most of the neutrons in the most reactive location in the system. If the system is sufficiently complicated so there is no obvious choice for the initial source, it may be necessary to repeat the calculation several times with different sources. If very different initial distributions converge to the same final results, one can have increased confidence in the answer.

The user should routinely examine a number of quantities in addition to keff, especially for a complex problem. The use of additional computer storage to accumulate fission densities and the fission source (Green's function) matrix is justified by the value of the information obtained, especially since these extra computations require very little time.

Finally, a computer code provides only a value of keff, not a criticality safety analysis. The safety engineer must identify possibly hazardous situations, such as spills, flooding, or double batching, and use the computer code to evaluate them. Use of Monte Carlo calculations together with experimental data and calculational models serves to reduce the number of computer calculations and to provide a cross check of results.

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Evaluation of Criticality Safety Models Used in Interaction Analysis, Douglas C. Hunt (AI-Colo), invited

As the nuclear industry has evolved in this country, the need for methods to evaluate the criticality of discrete batches or arrays of the various fissile material forms has grown with it. This need has given rise to a number of models, developed at several laboratories and process facilities, to correlate and extend the limited amount of available experimental array criticality data. Generally, these models were geared to the needs of the particular laboratory or production facility and have found limited acceptance outside their place of origin. The author has reviewed¹ the techniques which, to his knowledge, are used to any extent by criticality specialists in this country and has not attempted to determine the relative popularity of the methods.

Several other reviews²⁻⁵ of one or more of the methods have been made. This paper is unique in that all the methods are critically compared and then all are applied to the same series of array problems.

Two general classes of methods are considered. The first is semiempirical models in which equations that describe the data on either critical or safe arrays in terms of some selected set of array parameters are written. Most of these methods (as, for example, the density analog technique) are directly based on comparing arrays to single units. In addition to the density analog, these models include the surface density and equilateral hyperbola techniques. The second is array unit interaction models, which are based on equations written to describe the neutron balance for each unit. The solution

TABLE I
Results of Applying the Calculational Methods to the Example Arrays

Array Model	KENO	NB _N ²	Albedo	Interaction Potential ^a	Weighted Interaction ^a	Density Analog	Surface Density		Equilateral Hyperbola	Solid Angle
							s/v	Fraction Critical		
Array Type	Multiplication Factors					Array Permitted?				
Small briquette	0.605 ± 0.006	0.877	0.78	0.81	0.60	yes	no	no	---	yes
Large briquette	0.805 ± 0.008	0.847	---	---	---	yes	no	no	---	---
Small ingot	0.820 ± 0.007	>>1.0	1.05	1.19	1.07	no	no	no	---	no
Large ingot	0.854 ± 0.008	0.915	---	---	---	no	no	no	---	---
Small PuO ₂	0.550 ± 0.006	1.019	0.84	0.81	0.59	yes	yes	---	---	no
Large PuO ₂	0.800 ± 0.008	0.915	---	---	---	yes	yes	---	---	---
Small UO ₂	0.689 ± 0.008	0.995	1.12	0.89	0.55	---	no	---	---	yes
Large UO ₂	0.838 ± 0.008	0.942	---	---	---	---	no	---	---	---
Pulse columns	0.806 ± 0.008	0.959	0.77	0.98	0.81	---	no	no [yes] ^b	yes	yes
Large ion column	0.946 ± 0.008	1.055	---	---	---	---	no	no	---	---
Small ion column	0.904 ± 0.009	>>1.0	0.98 ^c	1.09	0.96	---	no	no	---	no
Mixed	0.860 ± 0.008	>1.0	---	---	---	---	---	---	---	---
Small low enrichment	0.747 ± 0.006	<1.0	<1.0	>1.0	~1.0	---	---	no	---	no
Large low enrichment	0.910 ± 0.006	<1.0	---	---	---	---	---	no	---	---

^aDetermined using reflected array albedo values from albedo method calculations.

^bNot permitted with Paxton⁶ surface density limits but allowed with Gutman⁷ limits.

^cColumn walls assumed to be a void region.

of these equations then typically relates the array reactivity, the neutron exchange probability between units, and the reactivity of the individual units. These methods include the albedo method and the Oak Ridge solid-angle method.

The methods are compared by applying each of them to several specific storage and process operation problems. The procedure is to select an array and compute its multiplication factor (k_{eff}) as accurately as possible using Monte Carlo methods. This reference value of k_{eff} is compared to the results of predictions by the several models.

Each problem is selected both because it represents a class of practical situations of interest to the criticality specialist and because it illustrates some features, either positive or negative, of one or more of the methods. Where possible, the problems are constructed so most of the models apply and a general cross comparison may be made. Because some models give multiplication factor values (k_{eff}) and others give safe parameters, the only possible comparison may be a statement that one method says an array is safe and another that the array k_{eff} has a certain value. A summary of the results of these comparisons appears in Table I.

A few general conclusions are that the interaction methods are more practical and reliable for arrays consisting of a small number (say <64) of units, while the semiempirical techniques are most applicable to arrays with a large number of low-reactivity units. The albedo method, with suitable input parameters, was found to be the preferred interaction method. The s/v surface density technique, used in areas where it has been developed, was the most useful semiempirical model.

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The Role of Criticality Models in ANSI Standards for Nuclear Criticality Safety, J. T. Thomas (ORNL), invited

The most useful information in the prevention of criticality is knowledge of the conditions required to produce criticality for fissile materials. Experiment and validated, detailed neutronic calculations are the best source of useful basic information. Translation of these data into terms familiar to engineers and operators is

often by means of models. In this context, model implies a system of postulates, data, and inferences presented as a mathematical description of a state of affairs. In nuclear criticality safety it is a prescription proposed as worthy of imitating criticality.

Fortunately, there are now criteria for acceptance of a model given in the ANSI Standard¹ N16.9. The principal requirement for validation is the correlation of results from the proposed model with the results of criticality experiments, or with the results of another validated method, to establish a bias for the model. The bias serves to normalize the model over the applicable range of its parameters so criticality will be predicted within the evaluated uncertainty of the bias. Following validation, the parameters are adjusted, or limited such that applications of the model produce specifications for systems known to result in subcriticality. The procedure contributes to bases for standards in the field of nuclear criticality safety.

Examples of two methods used in nuclear criticality safety evaluations in the area of neutron interaction among subcritical components of fissile materials are the solid-angle and surface density techniques. As described in the "Nuclear Safety Guide," TID-7016, rev. (1961), the solid-angle method is a two-parameter model utilizing the computed total solid angle subtended at one unit by others, given in steradians, and the maximum of the evaluated k_{eff} 's of the units in the system. Since inception of the method, these parameters were, and remain, deliberately and arbitrarily limited to produce prescriptions of systems far removed from criticality. The method is deficient because of its inability to predict criticality, the absence of specific criteria for the evaluation of the unit k_{eff} 's, and a poorly defined area of applicability.

There are several surface density techniques,^{2,3} some of which meet the criteria for a model with appropriately defined areas of applicability. One of interest parallels the solid-angle method in its proposed application by specifying conservative but unknown subcriticality. Only two criticality data points appear necessary; that for the geometries of the infinite slab and of the unit considered for storage. The situation is aptly presented in Fig. 1 where the ratio of surface densities, that of the storage array to the infinite slab for the same reflector condition, is shown as a function of the ratio of storage-unit mass to the critical mass in the same geometry, expressed as a fraction. The systems presented are described in the

Fig. 1. Surface density representation of reflected infinite planar arrays of U(93.2) metals and solutions.

figure. The problem centers on how large an area, i.e., limits for the ordinate and abscissa, can be utilized to define storage systems known to be subcritical.

Note that the inconsistent treatment of the metal cube data suggests the critical mass value should not be the unreflected cube but that for the condition with associated reflector. The solution data reported by Odegarden and Stevenson⁴ appear to associate well with the data for U(93) metal spheres. The sphere data also represent the 44 different fissile materials given in the ANSI Standard⁵ N16.5. It is clear that an area of applicability can be defined. However, the effect on the criticality of these systems of introducing interstitial low-density hydrogenous moderation needs to be examined before firm limits are proposed.

1. ANSI Standard N16.9, "Validation of Computational Methods for Nuclear Criticality Safety" (1975).
2. J. T. THOMAS, *Trans. Am. Nucl. Soc.*, 22, 299 (1975).
3. H. C. PAXTON, "Criticality Control in Operations with Fissile Materials," LA-3366 (rev.), Los Alamos Scientific Laboratory (1972).
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5. ANSI Standard N16.5, "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials" (1975).

Effect of Boron and Gadolinium on the Criticality of Plutonium-Uranium Systems, R. C. Lloyd, E. D. Clayton (Battelle-Northwest)

Experiments were performed to substantiate the effectiveness of soluble nuclear poisons on the criticality of plutonium-uranium nitrate solutions and to determine their effectiveness for criticality control on a heterogeneous assembly of mixed-oxide fuel pins immersed in plutonium-uranium nitrate solutions containing the soluble poisons.

Calculations indicate the most effective use of soluble poisons may involve mixtures of several different elements, such as a mixture of boron and gadolinium, since one nuclide is more effective than another in absorbing neutrons from different portions of the neutron spectrum as the fuel concentration, or H/X ratio, is varied. As these are the first data known to be reported on boron-gadolinium mixtures, the data provide the only validation points for calculations of the mixed absorber effects.

Heterogeneous System: The lattice contained 301 fuel pins in a 55.5-cm-i.d. cylindrical vessel with 0.079-cm wall thickness. Pu+U nitrate solution was pumped into the vessel, using the critical approach method to determine the critical height. The Pu+U solution contained about 80 g Pu/liter. The ²⁴⁰Pu content of the Pu was 6.3%, and the ²³⁵U content of the U was 0.66%. The Pu made up about 30% of the total Pu+U by weight. The solution

TABLE I
Effect of Boron and Gadolinium on Criticality of Heterogeneous Lattice Assembly

Date	Exp No.	Critical Height (cm)	Pu Conc (g/liter)	U Conc (g/liter)	Acid Molarity	Total NO ₃ (g/liter)	Specific Gravity	Gd (g/liter)	B (g/liter)
5-7-76	141R	21.20	84.5	182.1	1.75	296	1.438	0.13	0.27
5-9-75	142	25.37	84.0	183.2	1.88	302	1.442	0.235	0.6
5-13-75	143	30.49	82.7	180.6	2.01	308	1.444	0.309	0.9
5-30-75	148	43.86	81.2	180.0	2.18	321	1.447	0.424	1.35
6-3-75	149	51.97	81.0	180.4	2.09	316	1.451	0.519	1.5
7-17-75	152	55.18	81.0	180.3	2.21	318	1.452	0.537	1.548
7-18-75	153	65.42	80.5	180.5	2.24	321	1.454	0.541	1.662
Fuel Pin Dimensions						Fuel Per Pin			
Fuel column Cladding (316-SS)		o.d. (cm)		Length (cm)		PuO ₂ -U (nat)O ₂ : 138.4 ± 1.3 g			
		0.495		69.22		Pu: 30.75 ± 0.03 g			
		0.584		72.90		U: 91.16 ± 1.03 g O: 16.49 ± 0.17 g			
Fuel Enrichment						Fuel Density			
25.2 wt% Pu						10.35 ± 0.09 g/cm ³ (93.34 ± 0.79% theoretical)			
Isotopic Composition of Pu in Pins						Lattice			
²³⁸ Pu: 0.04 ± 0.01 at.% ²³⁹ Pu: 86.19 ± 0.06 at.% ²⁴⁰ Pu: 11.88 ± 0.06 at.% ²⁴¹ Pu: 1.73 ± 0.01 at.% ²⁴² Pu: 0.16 ± 0.01 at.%						Triangular spacing = 3.048 cm			

TABLE II
Effect of Boron and Gadolinium on Criticality of Homogeneous Systems
(U + Pu Solutions in 61-cm-diam Water-Reflected Cylinder)

Date	Exp No.	Critical Height (cm)	Pu Conc (g/liter)	U Conc (g/liter)	Acid Molarity	Total NO ₃ (g/liter)	Specific Gravity	Gd (g/liter)	B (g/liter)
4-7-75	137R	18.13	85.0	182.5	1.55	320	1.433	0.04	0.0
4-9-75	138	18.65	84.9	182.2	1.61	311	1.434	0.04	0.1
4-11-75	139R	19.68	84.8	182.6	1.67	290	1.433	0.04	0.3
4-17-75	140	21.13	84.5	182.1	1.75	296	1.438	0.128	0.27
5-20-75	145	32.16	82.8	180.6	1.95	309	1.443	0.293	0.9
5-22-75	146	43.03	82.2	179.5	2.12	319	1.446	0.388	1.2
5-26-75	147	52.12	81.2	180.0	2.18	321	1.447	0.424	1.35
5-16-75	150	67.83	81.0	180.4	2.09	316	1.451	0.519	1.5
7-11-75	151	75.44	81.0	180.3	2.21	318	1.452	0.537	1.54

contained a combination of B and Gd in the ratio ~3-to-1 by weight. The data from these experiments are summarized in Table I.

Homogeneous System: The experiments were performed on a water-reflected, stainless-steel cylindrical vessel of 61.03-cm i.d. and wall thickness 0.079 cm. Critical experiment data on the homogeneous Pu+U solutions containing various concentrations of B and Gd are given in Table II. The variation in critical height with added quantity of neutron poison can be seen from the value given in the table. Several of the Pu+U solutions are the same solutions as used in the heterogeneous system.

The results of theory-experiment comparisons utilizing ENDF/B cross sections and the KENO Monte Carlo code will be presented and discussed.

Critical Experiments with Low-Moderated Homogeneous Mixtures of Plutonium and Uranium Oxides Containing 8 and 15 wt% Plutonium, S. R. Bierman, E. D. Clayton (Battelle-Northwest)

Data from the latest in a series of experiments to define the criticality parameters of undermoderated, homogeneous oxide mixtures of plutonium and uranium are presented in this paper. Data from preceding experiments have been reported in Ref. 1-3. These latest measurements were carried out with fuel mixtures containing about 8 and 15 wt% Pu in the Pu+U and having H/(Pu+U) atomic ratios of about 7 and 3, respectively. As in the past, with this high a concentration of oxide, the experiments were performed with solid compacts of PuO₂-UO₂-polystyrene 5.09 cm square and having thicknesses of about 5.09 and 1.3 cm. Each compact was clad with MM&M #471 tape. A complete description is given

in Table I for each of the two fuels, the cladding material, and the Plexiglas reflector used in the experiments. In both fuel mixtures the plutonium contained about 11.5 wt% ²⁴⁰Pu and the uranium was depleted to about 0.2 wt% ²³⁵U.

Neutron multiplication measurements were made on fully reflected parallelepipeds of each fuel. For the 8 wt% Pu-enriched fuel mixture, the geometry ranged from the near cubic to a relatively thin slab. However, because of the prohibitively large volume required for criticality with the 15 wt% Pu fuel, the experiments were limited to a single near-cubic assembly. The critical sizes obtained from these measurements are shown in Table II. Also shown in Table II are computed k_{eff} values for each of these critical assemblies. The calculations were performed with the averaged cross sections that had been processed from ENDF/B-III with the FLANGE and ETOG codes.

With two exceptions, the data covered in this paper, along with the data previously published,¹⁻³ provide sufficient experimental data for validating calculations on undermoderated LMFBR homogeneous plutonium-uranium mixtures containing up to 30 wt% Pu. Criticality data are still needed on dry or moist oxide mixtures and on mixtures having 25 to 40 wt% ²⁴⁰Pu in the plutonium.

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2. S. R. BIERMAN and E. D. CLAYTON, "Critical Experiments with Mixed Oxides of Pu and U Containing 8 and 30 wt% Plutonium," *Trans. Am. Nucl. Soc.*, 15, 307 (1972).
3. R. C. LLOYD, S. R. BIERMAN, and E. D. CLAYTON, "Criticality of Plutonium-Uranium Mixtures Containing 5 to 8 wt% Plutonium," *Nucl. Sci. Eng.*, 55, 51 (1974).

Description of Experimental Fuel and Reflector

	15.0 wt% Pu FUEL 2.86 H: (Pu + U), ATOMIC			8.1 wt% Pu FUEL 7.3 H: (Pu + U), ATOMIC		
FUEL COMPOSITION, 10^{24} atoms/cm ³						
²⁴¹ Am	2.765×10^{-5}			3.132×10^{-6}		
²³⁸ Pu	5.737×10^{-7}			3.809×10^{-7}		
²³⁹ Pu	1.118×10^{-3}			3.490×10^{-4}		
²⁴⁰ Pu	1.478×10^{-4}			4.678×10^{-5}		
²⁴¹ Pu	2.221×10^{-5}			7.745×10^{-6}		
²⁴² Pu	2.370×10^{-6}			1.116×10^{-6}		
²³⁵ U	1.365×10^{-5}			9.116×10^{-6}		
²³⁸ U	7.549×10^{-3}			4.606×10^{-3}		
O	1.840×10^{-2}			1.178×10^{-2}		
C	2.653×10^{-2}			3.567×10^{-2}		
H	2.534×10^{-2}			3.680×10^{-2}		
CLADDING COMPOSITION, 10^{24} atoms/cm ³						
H	4.489×10^{-2}			4.489×10^{-2}		
C	3.111×10^{-2}			3.111×10^{-2}		
Cl	7.240×10^{-3}			7.240×10^{-3}		
REFLECTOR COMPOSITION, 10^{24} atoms/cm ³						
H	5.666×10^{-2}			5.666×10^{-2}		
C	3.510×10^{-2}			3.510×10^{-2}		
O	1.428×10^{-2}			1.428×10^{-2}		
PuO ₂ PARTICLE SIZE, μ m						
95%	< 20			< 20		
50%	< 5			< 5		
5%	< 1			< 1		
UO ₂ PARTICLE SIZE, μ m						
95%	< 18			< 18		
50%	< 5			< 5		
5%	< 2			< 2		
POLYSTYRENE PARTICLE SIZE, μ m						
95%	< 225			< 225		
50%	< 150			< 150		
5%	< 50			< 50		
FUEL COMPACT SIZES, cm	UNCLAD	CLAD	STACKED ^a	UNCLAD	CLAD	STACKED ^a
Length	5.090 ± 0.005	5.114 ± 0.005	5.129 ± 0.013	5.090 ± 0.005	5.114 ± 0.005	5.138 ± 0.089
Width	5.090 ± 0.005	5.114 ± 0.005	5.129 ± 0.013	5.090 ± 0.005	5.114 ± 0.005	5.138 ± 0.089
Heights	5.082 ± 0.066	5.170 ± 0.067	5.173 ± 0.037	5.081 ± 0.003	5.168 ± 0.003	5.170 ± 0.060
	1.265 ± 0.041	1.353 ± 0.044	1.395 ± 0.056	1.274 ± 0.008	1.361 ± 0.009	1.381 ± 0.064

^a Average dimension of space occupied by compacts.

Critical Assembly Configurations and Calculated k_{eff} Values

CRITICAL NUMBER OF FUEL COMPACTS						
REFLECTOR	LENGTH (a)	WIDTH (a)	HEIGHT		CORRECTED HEIGHT (d)	KENO k_{eff} (e)
			(b)	(c)		
8.1 WT% Pu, 7.3 H: (Pu + U) PuO ₂ -UO ₂ -POLYSTYRENE FUEL COMPACTS						
Plexiglas	10	9	9	1.135 ± 0.001	9.274 ± 0.003	1.025 ± 0.005
Plexiglas	10	11	7	3.239 ± 0.001	7.780 ± 0.002	1.027 ± 0.004
Plexiglas	12	12	6	2.526 ± 0.003	6.609 ± 0.007	1.021 ± 0.006
Plexiglas	14	13	5	3.733 ± 0.005	5.919 ± 0.008	1.021 ± 0.003
15.0 WT% Pu, 2.86 H: (Pu + U) PuO ₂ -UO ₂ -POLYSTYRENE FUEL COMPACTS						
Plexiglas	10	10	9	6.651 ± 0.016	10.749 ± 0.031	0.994 ± 0.002

- a) Number of compacts in the length and width dimensions. Each dimension is 5.09 cm per compact.
- b) Number of full sized compacts in the height dimension. Each compact is 5.081 cm high for the 8.1 wt % Pu fuel and 5.082 cm high for the 15 wt % Pu fuel.
- c) Number of thin sized compacts in the height dimension. Each compact is 1.274 cm high for the 8.1 wt % Pu fuel and 1.265 cm high for the 15 wt % Pu fuel.
- d) Critical height in terms of full sized compacts corrected for measured reactivity worth of smaller compacts.
- e) 18 energy group EGGNIT averaged cross-sections using FLANGE-ETOG processed ENDF/B-III data.

Reactor Theory and Applications: I

Calculations for Highly Compressed Microspheres of Fissile Transuranium Isotopes, A. D. Krumbein (Israel AEC)

In a previous study,¹ the possibility of obtaining a fission microexplosion was examined for the fissile elements ^{233}U , ^{235}U , and ^{239}Pu . In view of the generally discouraging results of that study, it was suggested that it might be interesting to examine the higher transuranium fissile isotopes which have large values² of ν_{eff} . Even though these isotopes are much more expensive to produce than the usual fissile elements, if their critical masses are small enough it might prove worthwhile to extract them from spent reactor fuel or even to breed them.³

Initially, the use of the spontaneous neutron emitter ^{252}Cf was considered, but examination of the values of ν_{eff} for this isotope shows them to be much lower than those for ^{239}Pu . In addition, no advantage can be taken of the spontaneous fission since, in the time required for the microexplosion ($\sim 10^{-9}$ sec), only about 2×10^3 neutrons/g are emitted. However, ^{239}Cf might possibly be useful in a highly compressed configuration as a neutron trigger.⁴

An examination of the group cross-section sets for the heavy actinides, compiled by Benjamin,⁵ revealed that the most promising candidates for microfission are ^{245}Cm and ^{249}Cf . Californium-251, because of its very low high-

energy fission cross sections, appears to be interesting only in a moderated configuration. Accordingly, the 54-group higher energy cross sections, as well as the top 20 THERMOS group cross sections given by Benjamin, were reduced by a rather simple averaging process to the 16-group Hansen-Roach cross-section structure employed in the previous calculations.¹ These cross sections were then used in the SNG neutron transport code¹ to calculate critical radii at the maximum theoretical density, i.e., 240 times a normal density of 20 g/cm³, as well as for the three-region varying density configuration¹ deduced from the computations of Brueckner and Nuckolls. Several configurations were also run as a function of time using the SNT code¹ to obtain neutron multiplications. The results are given in Table I. Included in this table are the previous results obtained for ^{239}Pu , as well as a calculation for ^{241}Pu using the 26-group Bondarenko cross-section set.

The results show that a smaller critical radius is obtained for both ^{245}Cm and ^{249}Cf than for ^{239}Pu , but not small enough to have a significant effect on any practical microfission process. The advantage becomes even smaller when one goes to the more realistic varying density configuration. The results also show that the addition of a deuterium blanket does indeed have a strong effect on the critical radius of ^{251}Cf . However, as the time-dependent computations indicate, the neutron multiplication is very low because of the long transit times of the moderated and reflected neutrons, compared to the disassembly times of the pellet.

TABLE I
Criticality and Neutron Multiplication Results for Highly Compressed Fissile Pellets

Pellet Composition	Radius (cm)		K _{eff}	Neutron Multiplication	Time for Neutron Multiplication (sec)
	Max Theoretical Density	Layered Densities			
^{239}Pu	0.020 0.075	0.25	1.0000 1.48	4.6×10^{28}	1×10^{-9}
^{241}Pu	0.023		1.0000		
^{245}Cm	0.011 0.015 0.021	0.14	1.0000 1.14 1.30	4.0×10^{25} 1.0×10^{44}	1×10^{-9} 1×10^{-9}
^{249}Cf	0.013	0.17	1.0000		
^{251}Cf	0.35		1.0000		
$^{239}\text{Pu} + \text{D}^a$	0.0018 (0.005) 0.007 (0.003)	0.07 (0.075)	1.0000 1.43	3.0	2×10^{-10}
$^{245}\text{Cm} + \text{D}^a$	0.00095 (0.005) 0.0035 (0.003)		1.0000 1.56	5.5	2×10^{-10}
$^{249}\text{Cf} + \text{D}^a$	0.0010 (0.005) 0.0002 (0.015)		1.0000 1.0000		
$^{251}\text{Cf} + \text{D}^a$	0.0006 (0.005) 0.0005 (0.015)		1.0000 1.44	2.3	2×10^{-10}

^aThe number in parentheses is the thickness of the deuterium shell in centimeters.

Table I also gives the neutron multiplications obtainable for pure fissionable microspheres at the maximum theoretical density. Here it is seen that a pellet of ^{245}Cm must have a radius of only about a quarter of that required for ^{239}Pu to achieve a comparable neutron multiplication. Additional calculations were run, however, which take into account the depletion of the fissionable material with time. These show that, because of depletion, neutron multiplication reaches a maximum of only 5×10^9 (for the ^{239}Pu case shown in the table) at about 3×10^{-10} sec, and then steadily decreases. For the ^{245}Cm pellet of 0.021-cm radius, a maximum of about 3×10^9 is obtained at about 2.4×10^{-10} sec.

These results for the transuranium isotopes tend to confirm our previous conclusions that the fission-fusion approach to microfission explosions is the one to be pursued.

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Measurement and Calculation of Reactor Parameters: I

Testing of ENDF/B-IV with GCFR Design Codes Through Analysis of Critical Experiments, A. L. Hess, R. A. Moore, D. R. Mathews (GA)

General Atomic (GA) participates in the CSEWG program for testing ENDF/B-IV: (a) as a cooperative effort to improve the nuclear data standards for reactor design; (b) to check in-house codes against integral experiments; and (c) to compare our techniques with methods employed by other fast reactor groups. Here we report on analyses for the CSEWG benchmark assemblies¹ ZPR3-48, ZPR6-6A, ZPR6-7, and ZPPR-2 and also on a comparable study for the initial GCFR critical assembly.² The capability of the design codes with large void channels, rather than sodium, was a prime concern.

Neutronics methods at GA, reviewed by Merrill,³ include the spectrum code GGC5 for generating multi-group cross sections. Heterogeneity effects in the ZPR cells are treated in the resonance section of GGC5 and by S16 transport calculations in slab geometry. Generally, the system is less rigorous than MC²-2/SDX, but more precise than the shielding-factor formalisms for cross sections.

Table I lists specifications of the analyses and gives the results as calculated-to-experimental ratios (C/E

values). For ZPR3-48, studies were conducted of the variations produced by different models, group structure, and heterogeneity treatments. The GCFR Phase-1 analysis includes the effect of streaming in the voids by using directional diffusion coefficients; for this case, the applicability of the transport correction is uncertain.

For the ZPR3-48, ZPR6-6A, and ZPPR-2 cases, the Table I k-values agree well with results given in the CSEWG test report⁴ for the ETOX-1DX shielding-factor method. However, the GA k-values are generally higher, 0.5 to 1.6%, than ANL results⁴ obtained using MC²-2/SDX. Central worths and reaction ratios from our calculations are not significantly different from those of the other test participants. Calculations for the Phase-1 GCFR assembly compare with measurements as well as those for the sodium assemblies.

Additional ZPR3-48 analysis was done to define other potential data tests. Figure 1 compares calculated and measured ²³⁸U fission rate profiles; the agreement implies adequate ²³⁸U removal cross sections at high energies in ENDF/B-IV.

1. "Cross Section Evaluation Working Group Benchmark Specifications," BNL 19302 (ENDF-202) (Nov. 1974).
2. S. K. BHATTACHARYYA et al., "The Gas-Cooled Fast Reactor Benchmark Critical Assembly," *Trans. Am. Nucl. Soc.*, 22, 696 (1975).
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Fig. 1. Comparison of radial ²³⁸U fission rate profile measured in ZPR 3-48 with GA calculation using ENDF/B-IV.

TABLE I

Comparisons of Measurements in Fast Critical Assemblies with Calculations by GA Methods using ENDF/B-IV Cross Sections

Assembly Cell Treatment Reactor Model Groups/Method	ZPR3-48 Homogeneous Sphere 29/P ₁ -S ₄	ZPR3-48 Heterogeneous 2-D Cylinder 10/diffusion	ZPR6-6A Homogeneous Sphere 27/P ₁ -S ₄	ZPR6-7 Homogeneous Sphere 27/P ₁ -S ₄	ZPR-2 Homogeneous 2-D Cylinder 10/diffusion	GCFR-Ph1 Heterogeneous 2-D Cylinder 10/diffusion
Calculated K	0.9876	0.9930	0.9903	0.9840	0.9748	0.9982
Δk for Heterogeneity	0.0140	--	0.0073	0.0166	0.0175	--
Transport correction	--	0.0073	--	--	0.0024	0.0077
Corrected k (C/E)	1.0016	1.0003	0.9976	1.0006	0.9997	1.0059
<u>C/E for Central Ratios^a</u>						
²³³ U f/ ²³⁵ U f	--	--	--	--	--	0.990
²³⁸ U f/ ²³⁵ U f	1.033	1.045	0.922	0.917	1.054	0.988
²³⁹ Pu f/ ²³⁵ U f	0.985	0.988	--	0.965	0.982	1.006
²⁴⁰ Pu f/ ²³⁵ U f	1.009	1.018	--	--	1.067	1.025
²³⁸ UC f/ ²³⁵ U f	1.038	1.038	1.036	1.048	--	1.017
²³² Th f/ ²³⁵ U f	--	--	--	--	--	0.953
<u>C/E for Central Worths</u>						
CH ₂	--	--	--	--	--	-1.38
10B	1.00	1.03	0.93	1.11	1.03	1.17
C	6.12	6.60	0.21	1.68	1.46	1.71
Na	1.76	2.81	-2.59	1.08	1.06	--
Fe	1.30	1.34	--	1.20	1.27	--
²³⁵ U	1.17	1.21	1.10	1.14	1.26	1.26
²³⁸ U	1.10	1.15	1.17	1.04	--	1.21
²³⁹ U	1.18	1.20	1.07	1.18	1.14	1.20
²⁴⁰ Pu	1.07	1.05	--	--	--	1.26
Factor Used, 1h/%k	920	920	436	944	947	958.6

^a Calculated ratios corrected for cell heterogeneity

Physics Problems Associated with LWR Fuel Storage Systems

ENDF/B-IV Thermal Data Testing: Methods, Results, and Recommendations, F. J. McCrosson (duPont-SRL)

The Evaluated Nuclear Data File/Version B (ENDF/B)¹ provides a computer-oriented reference set of evaluated neutron cross sections for thermal and fast reactor applications. New versions of ENDF/B have been released at about a frequency of one every two years to improve its applicability, and its scope has been enlarged to include shielding and dosimetry applications. The latest version, released in 1974 and denoted ENDF/B-IV, provides reevaluations of the ²³⁵,²³⁸U cross sections to improve predictions for thermal systems. Development and testing of ENDF/B are performed by the Cross Section Evaluation Working Group (CSEWG), which is comprised of scientists from universities and laboratories throughout the U.S. and Canada. The file is released through the National Neutron Cross Section Center at Brookhaven National Laboratory.

The CSEWG Data Testing Subcommittee has analyzed 17 benchmark experiments to test the merit of ENDF/B-IV for thermal reactor calculations.² The experiments were made at room temperature and were free of fission products. Representative values of k_{eff} computed with ENDF/B-IV for the thermal benchmarks are shown in Table I. Comparisons between measured and calculated activation parameters for the lattices have also been made. The specifications for the CSEWG thermal benchmarks are given in Ref. 3.

ENDF/B-IV predictions of criticality and the measured activation parameters of the uranium experiments are significantly improved over those of ENDF/B-III (Ref. 4). These improvements are due (a) to revisions of the ²³⁵U thermal cross sections,⁵ which raised predictions of k_{eff} for the uranyl (93 wt% ²³⁵U) nitrate spheres by 0.2%, and (b) revisions of the ²³⁸U resonance cross sections,⁶ which raised k_{eff} for the less-moderated lattices by about 1% and reduced ρ^{23} , the ratio of epithermal-to-thermal ²³⁸U captures, by nearly 5%. In spite of these improvements, there is a residual underprediction of k_{eff} for the lattices of about 1%. The parameter, ρ^{23} , is 3 to 5% too high, which indicates the observed underprediction of k_{eff} is traceable, at least in part, to an overprediction of epithermal ²³⁸U capture. The 1 to 2% overpredictions of k_{eff} for the plutonium nitrate spheres are very nearly the same as those obtained from ENDF/B-III (Ref. 4).

Calculational methods differed widely among the eight ENDF/B-IV thermal data testing participants. Reported values of k_{eff} for the lattices differed by as much as 1% among the laboratories, i.e., as much as the representative values in Table I differ from unity. To isolate the origins of these differences and thereby provide a truer assessment of the cross sections themselves, the CSEWG thermal data testing included comparisons of the processed multigroup cross sections. For Benchmark TRX-1, the comparison included reaction rates by nuclide for three epithermal groups and one thermal group, and the calculated slowing-down sources. The comparisons revealed significant differences among the processed multigroup cross sections, but the most significant differences in the few-group reaction rates were due to variations in resonance self-shielding treatments.

For ENDF/B-V, the CSEWG Data Testing Subcommittee should define the most rigorous methods for calculating each of the benchmark experiments. The Subcommittee should then use these methods to evaluate the merit of the cross sections rather than accept a consensus obtained by averaging the results of numerous calculational techniques. Processed multigroup cross sections and intermediate results of the calculations should be reviewed by the Subcommittee. Estimates of the calculational uncertainties should be sought and documented. Results from frequently used but less accurate methods should be reported to provide a measure of the sensitivity of the predictions to variations in calculational methods.

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3. "Cross Section Evaluation Working Group Benchmark Specifications," BNL 19302 (ENDF-202) (Nov. 1974).
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TABLE I
Criticality (ENDF/B-IV)

Benchmark	Description	k_{eff}
ORNL	Unreflected spheres of uranyl nitrate solution	
-1	H/ ²³⁵ U = 1378; R = 34.595 cm	1.0004
-2	H/ ²³⁵ U = 1177; R = 34.595 cm	1.0002
-3	H/ ²³⁵ U = 1033; R = 34.595 cm	0.9970
-4	H/ ²³⁵ U = 971; R = 34.595 cm	0.9981
-10	H/ ²³⁵ U = 1835; R = 61.011 cm	0.9961
TRX	H ₂ O-moderated U lattices	
-1	Mod/Fuel = 2.35	0.9887
-2	Mod/Fuel = 4.02	0.9942
-3	Mod/Fuel = 1.00	0.9950
-4	Mod/Fuel = 8.11	0.9968
MIT	D ₂ O-moderated U lattices	
-1	Mod/Fuel = 20.74	0.9878
-2	Mod/Fuel = 25.88	0.9880
-3	Mod/Fuel = 34.59	0.9906
PNL	Unreflected spheres of Pu nitrate solution	
-1	H/ ²³⁹ Pu = 698; R = 19.509 cm	1.0249
-2	H/ ²³⁹ Pu = 124; R = 19.509 cm	1.0250
-3	H/ ²³⁹ Pu = 1204; R = 22.70 cm	1.0026
-4	H/ ²³⁹ Pu = 911; R = 22.70 cm	1.0110
-5	H/ ²³⁹ Pu = 578; R = 20.1265 cm	1.0185

Review of EPRI Programs in Cross-Section Development, Odelli Ozer (EPRI)

The objective of EPRI programs in the area of nuclear cross sections has been the development of a "National Standard Data Base" for power reactors.

Since the ENDF/B system has been found to satisfy most of the requirements of such a standard, EPRI programs have been coordinated with those of CSEWG in an attempt to improve the performance of this library in primarily thermal reactor applications.

The overall effort can be considered as being directed into four areas: uses of data in analysis, review of integral experiments, new evaluations, and differential measurements.

Considerable attention has been given to the area of analysis. The most accurate methods available for the calculation of thermal lattice cells have been implemented in-house and at the Neutron Cross Section Center (BNL). The purpose of selecting these methods has been to separate effects of nuclear data uncertainties from those caused by approximations in the analytic models. Additional information on the performance of faster but less accurate methods commonly used as design tools has also been compiled for comparison purposes. To facilitate the participation of the utility industry in CSEWG data testing activities, ENDF/B-IV files have been processed for input into codes such as LASER, LEOPARD, HAMMER, CINDER, etc., and are presently being processed for use in EPRI's primary fuel cycle analysis code package, ARMP.

Integral experiments selected for analysis initially consisted of the set specified by CSEWG as "thermal benchmarks." This set is limited to homogeneous critical systems and to clean uranium metal lattices moderated by H_2O and D_2O . The analysis was next extended to clean and poisoned UO_2 lattices in H_2O , mixed (U-Pu) oxide lattices in H_2O , ^{235}U -thorium lattices in graphite, and other experiments of interest to EPRI programs relating to plutonium recycle, HTGR temperature coefficients, thorium utilization in LWRs, etc. A new set of mixed-oxide lattice experiments moderated by clean or poisoned H_2O and using a series of fuel-to-moderator ratios has been supported at the PRCF facility at Battelle Northwest Laboratory to provide an additional reference for testing Pu-recycle methods and cross sections. A program of reviewing the interpretation of derived experimental quantities such as ρ^{28} , δ^{28} , etc., has resulted in significant correlations to the CSEWG set.

Information on the performance of the ENDF/B-IV library in the analysis of the above benchmark experiments is being used in determining areas where improvements or new evaluations are needed. The attempt to arrive at a self-consistent set of 2200 m/sec cross sections for the fissile nuclides (^{235}U , ^{238}U , ^{239}Pu , ^{241}Pu) initiated by CSEWG for ENDF/B-IV is being continued under EPRI support with the Monte Carlo analysis of a few key irradiation experiments, the development of a nonlinear least-squares-fitting code based on the Adler-Adler formalism, and the reevaluation of the shapes of the cross sections of the above nuclides in the thermal range. New evaluations have also been supported for zirconium, hafnium, gadolinium, ^{232}Th , graphite (thermal scattering kernel), and a series of heavy actinide nuclides leading to the formation of ^{238}Pu and ^{252}Cf . The persistent discrepancy between measured and calculated capture rates in ^{235}U has been the subject of a two-day workshop-seminar during which various solutions to this problem have been proposed.

In the area of differential cross-section measurements, EPRI's program has been limited to the support of a single ^{238}U capture self-indication experiment recommended by the ^{238}U capture seminar. This experiment will provide information on the self-shielding of the lowest few ^{238}U resonances for a range of temperatures extending from ~ 0 to ~ 1000 K.

Future EPRI efforts will be directed toward testing of the next version of the ENDF/B library prior to its release and determination of the sensitivity of fuel cycle calculations to cross-section uncertainties.

A Review of Physics Analyses of Indian Point Unit No. 2 (PWR) Fuel Assembly Storage Racks, S. N. Purohit, G. Canavan, B. Chao, M. L. Lee (Con Ed-NY)

The General Electric announcement, in August 1974, of the inoperability of its Midwest Fuel Recovery Plant at Morris, Illinois, created a problem for the affected utilities to make provision for the storage of spent fuel that would be accumulating at their reactor sites, as well as for the storage of partially burned fuel assemblies that would be discharged from the reactor core in the event of emergency. This paper describes the physics studies, both nuclear criticality and decay heat, to permit expansion of the spent-fuel storage capacity at Indian Point Unit No. 2.

The following two objectives governed the design of the Indian Point Unit No. 2 expanded fuel storage capacity:

1. K_{∞} , the infinite multiplication factor, for a stored array of assemblies containing fresh 3.5 wt% ^{235}U fuel would be ≤ 0.90 . This criterion was adhered to to meet the current technical specification limit for the IP2 reactor.

2. Maximize the number of fuel assemblies that could be stored in the spent-fuel pool by reducing the center-to-center spacing to the extent permitted by subcriticality criterion, fabrication, and cooling considerations.

The increase in k_{∞} caused by the reduction of center-to-center spacing was compensated by incorporating boron-stainless-steel plates in the rack design.

Nuclear criticality analyses were carried out with the core analysis methodology currently being employed in the core-physics analyses of three Indian Point PWRs (Ref. 1). This methodology involved generation of fast- and thermal-group cross sections using GAM II and THERMOS codes, respectively; and the criticality analyses of stored fuel assemblies using DOT (transport theory) and CITATION (diffusion theory) codes. Some of the k_{∞} results obtained with the DOT code are shown in Table I to demonstrate the sensitivity of k_{∞} with the variation of center-to-center spacing and boron concentration.

Decay-heat calculations were undertaken to ensure that the spent-fuel pool cooling system would accommodate

TABLE I
 K_{∞} , Infinite Multiplication Factors
 for IP2 Fuel Storage Rack
 (3.5 wt% ^{235}U)

Center-to-Center Spacing (in.)	K_{∞} (1.0 wt% Boron 68°F)
13.875	0.874
13.275	0.891
12.000	0.950
Natural Boron Content (wt%)	K_{∞} (at 13.875-in. Spacing and 68°F)
1.0	0.874
0.9	0.878
0.75	0.885
No neutron absorber plates	0.950

the increased heat load and that the design basis maximum pool temperature reported in the IP2 FSAR would not be exceeded.

The decay-heat analysis involved the use of the ANS decay-heat standard.² Conservatism was employed in the analysis to account for the uncertainty in the reported data on decay heat.

Finally, it is our assessment that the uncertainties in k_{∞} calculations of current physics designs can be reduced by the availability of standard benchmarks for LWR fuel storage systems. This would also result in the reduction of economic penalties corresponding to the uncertainties in physics designs.

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2. American Nuclear Society Proposed Standard (ANS 5.1, N18.6), "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors" (Oct. 1971; rev. Oct. 1973).

A High-Density Spent-Fuel Storage Rack Design for Boiling Water Reactors, M. M. Allison, D. J. Denver, B. C. Slifer (YAEC)

Delays in the availability of reprocessing have presented many nuclear power plant operators with a shortage of on-site storage space for spent-fuel assemblies. This paper describes the design, physics, and safety aspects of a high-density replacement rack planned for installation at Vermont Yankee and Pilgrim Nuclear Generating Station. The design reduces nominal assembly pitch to ~7 in., thereby increasing the available storage space by a factor of 3 to 4.

To permit high-density storage, a survey of possible poison materials indicated that the most readily available material was Boral, a sandwich composed of a 0.084-in. core of 35 wt% natural B₄C in an aluminum matrix clad with 0.045 in. of aluminum. In the design, each storage location is isolated from its neighbors on all sides by a single plate of this neutron-absorbing material which is approximately 6 in. wide and extends beyond the active fuel length at both top and bottom.

The total weight of stored fuel and racks is limited by the load-carrying capability of the pool structure. The required minimum weight design was accomplished by a complete modal analysis of racks and supporting structure using the applicable seismic response spectra. Operational and handling loads have also been considered in the design. All installed equipment has been designed for minimum general and galvanic corrosion.

Extensive criticality calculations have been done for this rack to ensure that it meets the applicable criteria¹ under all conditions. Calculations were performed using a four-energy-group, xy representation of the racks in the PDQ-07 (Ref. 2) program. Cross sections for the PDQ-07 model were obtained from LEOPARD (Ref. 3) super cell calculations, which also yielded fast and thermal spectra for use in RODWORTH (Ref. 4), a blackness theory code which was modified to account for boron particle self-shielding. The resulting cross sections were then used in a PDQ-07 calculation which explicitly modeled the system geometry. The results have shown that the rack will have a K_{eff} below 0.83 under normal operating conditions and a K_{eff} below 0.95 including uncertainties and off-nominal design conditions. The accuracy of the PDQ-07 model has been checked in two ways:

1. The model was used to calculate a number of unpoisoned criticals.⁵
2. The PDQ-07 analysis of the Vermont Yankee spent-fuel rack in the nominal condition was compared with a similar analysis performed using the Monte Carlo program KENO-II (Ref. 6) with a 16-group Knight-modified Hansen-Roach cross-section set.⁷

The heat load on the pool for the modified installation is within the capability of the installed cooling components. Conservative selection of the critical design condition including assessment of in-reactor fuel operating histories and decay times after shutdown has shown that the heat load for the modified arrangement is equal to or smaller than that used in the original design.

The Boral curtains, in addition to serving as a neutron absorber, act also as coolant flow isolators preventing cross flow of the water between adjacent assemblies. The effect is such that each of the bundles becomes isolated and sits, therefore, in its own thermal chimney. Relap-4 (Ref. 8) models previously reported⁸ were used to ensure that local boiling would not take place with the current design.

Installation procedures have a large influence on the design and economics of the planned modification. Therefore, these considerations have been incorporated in the choice of rack materials and geometric arrangement.

In its review procedures, NRC has shown particular interest in the criticality, cooling, and seismic analysis in addition to the design and quality assurance specifications and codes.

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Diffusion Theory Analyses for Spent-Fuel Storage, R. T. Harris, D. S. Frentzos, W. R. Jones (NUSCO)

New emphasis has been placed on the analysis of criticality of spent-fuel storage pools in light of the lack of fuel reprocessing facilities. Fuel rack designs with decreased assembly center-to-center spacing have been proposed to increase at-reactor plant storage capability. This paper describes the results of criticality analyses performed using diffusion theory applied to a wide range of fuel rack designs. "Blackness theory" was used to treat strong absorbers. Comparisons between diffusion/blackness theory analyses and transport solutions will be presented.

The purpose of these criticality analyses was to provide a preliminary basis to determine maximum fuel pool capacity, and evaluate effective conceptual rack designs. A number of rack design types were analyzed with criticality as a function of spacing, including those with (a) no neutron absorbers, (b) boron carbide pins and plates, (c) Boral plates, (d) borated steel plates, and (e) stainless-steel plates. Plate-type designs include both single and double plates between adjacent assemblies; the latter are referred to as flux trap designs. LWR fuel assemblies including Westinghouse 15 × 15 stainless cladding, Combustion Engineering 14 × 14, General Electric 7 × 7, and Westinghouse 17 × 17 Zircaloy cladding were considered, reflecting this utility's nuclear plants.

Diffusion theory analyses utilized four-group PDQ-7 with a two-dimensional (2-D) xy geometry representing an infinite repeating array of fuel assemblies. Most group cross sections were generated with the CEPAC program, except those for strong absorbers which were based on blackness theory calculations. Structural materials, unless designed for neutron absorption, and dissolved boron in PWR pools, were neglected.

Fig. 1. Connecticut Yankee fuel pool with fresh 4.0 wt% SS-clad fuel.

Because of the wide range of designs considered, only representative results are presented in this summary. Figure 1 presents K_{eff} versus spacing for Connecticut Yankee fuel for the rack designs indicated. For one design, a comparison with an NUS Corporation Monte Carlo calculation is shown. Limited comparison with 2-D discrete ordinate transport theory calculations now in progress will be presented.

A useful representation of the results is the assembly neutron leakage coupling factor defined as (K_{eff}/K_{∞}) (assembly) versus spacing. If substantial spectral differences do not exist, e.g., different enrichments, this representation expands the applicability of results.

These analyses have resulted in the following conclusions:

1. Diffusion theory is an effective scoping tool for preliminary evaluation of proposed storage rack designs.
2. Flux trap design was found to be very effective.
3. Limited diffusion and transport theory comparisons indicate differences of $\sim 1\%$ Δk exist.
4. K_{eff}/K_{∞} assembly is a useful method for representing data for applications to slightly different fuel designs.

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An Instrumented Pool—A Possible Way to Increase Storage Capacity, T. Gozani (SAI-Cal)

The delays in the startups of reprocessing facilities have created considerable uncertainty as to the adequacy of the capacity of the spent-fuel storage in the nuclear power plant. The available storage pool capacity has been constrained by the requirement to apply unirradiated fuel (with the highest envisioned enrichment for the plant) critical mass limits to fuel storage configurations.

Certain increase in the storage capacity results from the use of more accurate and reliable methods for criticality calculations,¹ while still complying with the above-mentioned constraint. A more substantial increase in capacity (e.g., a factor of 2 or more) can result from combining modifications in the racks with the interposition of poison sheets (e.g., Boral) among the fuel assemblies.¹ An intermediate approach is to build the racks themselves from a good structural material that is also a moderate thermal-neutron absorber like stainless steel. These methods of increasing spent-fuel storage capacity may prove to be effective, when licensed and implemented. They, however, may also prove to be rather expensive.

The main reason for the fresh-fuel criticality constraint is the lack of accurate knowledge of the burnup of individual spent-fuel assemblies. Another reason for the extreme conservatism employed in the pool design arises from the lack of direct current knowledge of the subcriticality of the pool.

The capacity of the storage pool can be increased substantially without compromising its safety and at a reasonably low cost if the fuel irradiation effects, namely burnup and fission product, on storage criticality parameters are incorporated. This can be achieved by direct nondestructive measurements of these irradiation effects in each spent-fuel assembly; namely,

1. measurement of the residual fissile content (²³⁵U, ²³⁹Pu, and ²⁴¹Pu)
2. measurement of the neutron emission rate
3. measurement of the gross fission product via measurements of gamma emission rates
4. continuous monitoring of the subcriticality of the loaded pool.

The first three measurements make it possible to classify the measured fuel assembly in terms of its contribution to the source multiplication. The fourth measurement indicates the current criticality status of the pool. The key elements of the proposed approach are the direct nondestructive measurement of the residual fissile content, which was recently studied,² and the continuous monitoring of the subcriticality.

Several methods for subcriticality measurements^{3,4} may be considered. However, the high neutron background and the variability of the active core size in the storage pool dictate the use of the multidetector source multiplication technique, where the neutron emission from the fuel serves as the source. The modified source multiplication⁵ can serve as the method for the reactivity monitoring. The accuracy of the measurement will be rather poor when the system is very subcritical. In these instances it is important only that the instrument will indicate the trend correctly; namely, that the reactivity had indeed increased when a fuel assembly was added. As more fuel is added the reactivity increases and the accuracy of the subcriticality determination is greatly improved.

The information supplied by the fissile detector, the fission-product detector, the neutron detector, and the subcriticality monitor can be very effective in increasing the loading capacity of the present pools. It allows the storage of assemblies with a measured residual fissile content in appropriate (possibly predetermined) locations and in a sequence that will maximize the storage capacity while minimizing the reactivity of the subcritical system. The latter is constantly being monitored and it provides the main safety feature of the pool. The essential detector systems for fissile material and for subcriticality determination can be redundantly built and interlocked into the pool fuel handling facility, thereby enhancing the reliability and safety of the operation. Incorporation of the measuring and interlocking systems into the existing pools will require only moderate changes in the fuel racks. Obviously, the instrumented pool concept can be added to the other approaches of augmenting storage pool capacity. It will provide better knowledge of the pool status, furnish information on the utility's overall inventory of fissile material, and possibly allow for the reuse of fuel with unusually low burnup.

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Radiation Applications, Sources, and Detection

Criticality Studies of a Neutron Multiplier,

R. C. Lloyd, S. R. Bierman, E. D. Clayton, B. M. Durst (BNWL)

There has been considerable need for neutron irradiation units with high neutron flux for use in research, industry, and education.^{1,2} Sources of various types have been used for this purpose for many years, but have lacked the intensity and volume, with level flux, for larger sample irradiation. While the research reactor can produce the desirable neutron flux, the construction and operation is rather costly. A neutron multiplier has been described using a subcritical lattice of enriched uranium rods with a central cylindrical flux trap. A ^{252}Cf source is used to provide the initial neutrons to drive the unit. This lattice unit was loaded at the Critical Mass Laboratory to study aspects of criticality including k_{eff} and effects of temperature, voids, and added material.

The experimental system consisted of a lattice assembly, reflected and moderated with water, in a cylindrical tank having a diameter of 102 cm. Measurements were made at a lattice spacing of 1.52 cm. A schematic diagram of the assembly is shown in Fig. 1. The canned fuel rods were a U + Al alloy having a ^{235}U enrichment of 93 wt%; they were 1.02 cm in diameter and 60.33 cm in length.

The fuel rods were safely loaded, using the critical approach method; this also provided the extrapolation for the critical number of rods. At selected fuel loadings, pulsed-neutron source measurements were made to determine reactivity and prompt-neutron decay rates. The effect of voids and various materials placed in the central region and outside the loaded lattice was measured in terms of the number of rods required for criticality.

Table I shows the results of the critical approach measurements made at various temperatures with the

Fig. 1. Neutron multiplier lattice.

1.52-cm lattice. The critical number of fuel rods at 21°C, with a 5.1-cm central void and a 37-rod equivalent flux trap, was 271.4. The same system with the 5.1-cm-diam central tube half filled with water was 276.3. When either system was heated, the maximum decrease in critical number of rods, heating from 21°C, was about 2.6, while the system having the 5.1-cm-diam void leveled off at a somewhat lower temperature.

Measurements were made to determine the variation in critical number of rods with the size of central void. The fuel-rod loading was at a minimum with the flux trap slightly less than 30% voided.

A number of measurements were made to determine the effects of various materials in the central void region. A fuel rod of $\frac{1}{2}$ length (7.41 g ^{235}U), positioned in this region was equivalent, in terms of reactivity, to adding 4.4 fuel rods to the outside lattice region. On the other hand, filling the 1.52-cm void with 6 g/litre boric acid solution gave a large reduction in reactivity, a reactivity loss equivalent to the removal of some 29 of the outer fuel rods from the assembly.

TABLE I
Criticality of Neutron Multiplier 1.52 cm Lattice
(effect of temperature change)

Exp. No.	Temperature (°C)	N_c (Rods)	Max. Temp. Effect (Rods)	Condition
007	21	271.42	2.62	5.1-cm-diam Center Void 37-Rod Equivalent Flux Trap
010	32	269.08		
011	50	268.80		
012	71	269.17		
015	23	276.25	2.61	5.1-cm-diam Center Void- $\frac{1}{2}$ -filled with H_2O 37-Rod Equivalent Flux Trap
017	47	273.73		
018	70	273.64		

Neutron flux measurements showed a relative thermal-neutron source multiplication of 20 in the flux trap with the multiplier at a k_{eff} of 0.99. The multiplication of the system will vary depending on the value of k_{eff} , but a further advantage of the multiplier is the enhancement of the flux profile axially and radially; i.e., in addition to the multiplication, the flux is spread out over a larger volume.

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Moderation Control for Purposes of Nuclear Criticality Safety

Development of Standards for Moderation Control: A Case History, *D. M. Dawson (GE-San Jose)*

Moderation control as a technique for primary criticality control has been utilized in the nuclear industry for many years. It is a particularly useful technique in applications involving uranium enrichments of a maximum of 5 to 6% ^{235}U since the critical masses at these enrichments approach infinity as the amount of moderation approaches zero.

Moderation control, then, can be defined as

"the limitation of the amount of moderator or its distribution or both to less than optimum moderation or limitation of the total mass of moderating material to less than that required for criticality by design and control of processes and design of facilities for the purpose of nuclear criticality safety."

Moderation control enables the use of very large process "lots" to improve efficiency for large-batch BWR fuel fabrication. An example familiar to many of you is the shipment of several tons of enriched UF_6 in very large containers. The principle of moderation control is also applicable to higher uranium enrichment where critical masses can be made very large, though finite.

Although the current standards effort on moderation control includes the use of concentration control (limiting the amount of fissile material in solution), this session will be devoted to the principle of "keeping your powder dry."

I will briefly review the standards effort, the contents of the current draft of the Standard, and will appeal to your best intentions for help in generating data needed to complete the Standard.

Historical

In early 1968, the United States American Standards Institute suggested several areas of potential standardization to ANS-8, including moderation control. Jack McLendon asked me to chair a working group which was to be called "Criticality Safety Limits For Special Application Involving Parametric Controls" and to survey the industry to evaluate interest in the area of moderation control. The survey evoked only one negative response to the question of need, and therefore a scope was produced and a working group formed. The scope included moderation control of systems containing any fissile material.

The original Working Group included C. E. Newlon, Union Carbide; D. F. Cronin, United Nuclear; N. Ketzlach, Atomic International; H. Morton, Nuclear Fuel Services, and myself.

We met for the first time in Seattle in June 1969. There was limited interest in the first draft produced as a result of that meeting and it was decided to restrict the scope to UO_2 . A second draft was distributed in November 1969. Following a short meeting of the Working Group in San Francisco in December of 1969, the effort was abandoned for about one year due to the Work Group members' priorities. In October of 1970 a third draft was prepared and distributed to 16 persons. Comments were received from three of these. In October 1971, Francis Alcorn of Babcock and Wilcox replaced H. Morton. Francis was instrumental in the preparation of another draft with the scope restricted to 5% or less ^{235}U . As a result of comments on this draft, the scope was changed to include concentration control.

One must keep in mind that very little calculative data and essentially no critical data were available to us and a significant weakness of the standard was that it incorporated no data. The scope of the October 1971 draft was changed to incorporate data in an appendix. The appendix was not included with the draft due to the lack of data mentioned above. As expected, comments on this point were critical. An October 1972 draft incorporated all other comments and requested that ANS-8 approve the draft for transmittal to N-14 without data. It was proposed that data could be added when available by approval of ANS-8. This approach would allow promulgation of the moderation control principles but require the user to provide his own data. This request was denied.

The next draft was produced in June 1974 and included data in the form of "single parameter limits" based on a paper by Fred Welfare of Babcock and Wilcox which was presented at the 1974 ANS Winter Meeting. Very little response (4 out of 22) was received, because the data were still incomplete and it was evident that the Standard wasn't ready for approval.

At a September 1974 Working Group Meeting, it was decided to reorganize and to rewrite the data section and to utilize extrapolated values to complete the data tables. It was recognized that data could be generated through extensive calculative effort and attempts were made to get industry support. Results were not encouraging.

An October 1974 draft with extrapolated data was submitted to ANS-8 after encouragement from ANS-8 Chairman, Jack McLendon. Results of the letter ballot were 5 negative ballots out of a total of 12 returns. These negative ballots were due to lack of data verification and questions relating to validity of the data.

At the June 1975 ANS-8 Meeting, the results were reported and an action plan was developed at Jack's request. Based on lack of data and lack of interest in developing data at this time, the ANS-8.8 efforts on moderation control were tabled for 2 to 3 years.

Present Status

As I have attempted to emphasize, the major problem is the lack of verified data, either calculative or experimental, in the low moderation ranges for UO_2 . Originally the Working Group planned that the principles of moderation control, which are discussed in Section 4 of the Standard, were to be emphasized with the user of the Standard supplying his own data. However, one member of ANS-8 and later the majority of that group pressed for limits being a part of the Standard and although adequately verified and documented data are unavailable it is difficult to disagree with that view.

In late 1975, ANS-8 was requested by ERDA and EPRI to provide them with a definition of research and development needs related to our Standards efforts.

Our input to them was the following:

Verified calculational results of experiments are needed to provide values of k_{∞} , migration areas (associated with k_{∞} values greater than one), and conservative extrapolation distances (associated with k_{∞} values greater than one) for low enriched (≤ 5.0 wt% ^{235}U) uranium dioxide (UO_2) with very low moisture contents (0.5 to 5 wt% H_2O). These calculations or experiments should be performed at different tap densities (2.0 through 6.0 UO_2/cm^3). The results are to be used to establish subcritical limits if k_{∞} is less than one and to establish subcritical physical parameters using one-group calculations if k_{∞} is greater than one.

Programs are being developed by the NRC to generate experimental data related to this definition of need.

The present state of affairs can be attributed to a lack of widespread need for this Standard. Although the Working Group received repeated encouragement to continue due to need, very little interest by way of direct support has been forthcoming. It is my opinion that until the NRC exerts pressure (Regulatory Guide, etc.) there will continue to be an apathetic approach to this subject, although many organizations are using this method of criticality control.

Moderation Control of 5% ^{235}U Enriched UF_6 , C. E. Newlon, H. R. Dyer (UCC-OR)

In the moderation-control method for 5% ^{235}U enriched UF_6 , criticality safety is assured by control of two parameters—the ^{235}U enrichment and the hydrogen moderation. Topical subjects to be reviewed in method development include: UF_6 purity control in liquefaction operations; the influence of container size on internal and external moderation effects; decoupling properties of the cylinder wall in water-moderated UF_6 arrays; special neutron multiplication experiments with large UF_6 assemblies; derivation of moderation control limits; large cylinder cleaning procedures; and protective packaging. The administrative requirements for method application are also discussed briefly.

Moderation Control in a Large-Capacity Process Unit, H. W. Webb, M. A. Glor (B&W)

Introduction

A routine step in the production of UO_2 pellets for use in commercial reactor fuel is blending of the UO_2 feed material to assure homogeneity of the powder itself and thorough mixing of any additives required for process purposes. Nuclear safety of the blending operation can be

assured by mass, geometric, or moderation controls. In the production environment, the use of mass and, in some cases, geometric controls is limited due to throughput requirements. This paper describes the nuclear safety considerations for application of a large-volume, moderation-controlled blending unit to the production process. While the imposition of stringent design and administrative controls is necessary, large-volume powder blending has several production- and health-physics-related advantages.

Nuclear Criticality Safety Demonstration

The nuclear safety of the large powder blender is achieved by limiting the weight percent moisture equivalent in the material to 1.42 ($\text{H}/\text{U} = 0.45$). This is based on calculations of k_{eff} as a function of H/U ratio and mass of powder. The calculations were made with the one-dimensional (1-D) discrete ordinate transport code, ANISN (Ref. 2), using the modified Hansen-Roach³ 16-group cross-section set. The input assumed 4% enriched UO_2 at a density of 5 g/cm^3 homogeneously mixed with water in a fully water-reflected sphere. The H/U ratios used were 0.6, 0.9, and 1.2; and the mass of powder ranged from ~600 to 70,000 kg. The results of these calculations are shown in Fig. 1. Selected values read from Fig. 1 are replotted in Fig. 2 to determine the effective multiplication of both a normal-size lot and a double-batch lot for various H/U ratios. From this figure, it can be seen that a normal batch at an extrapolated H/U ratio of 0.45 has an effective multiplication factor of <0.80 , which is an acceptable value. An accident, which would double the mass in the blender, would result in a multiplication of <0.85 , which is also an acceptable value. By adopting 0.45 as the upper limit on H/U , a significant safety margin has been incorporated in that the critical mass would exceed 100 MT, which is more than 30 normal batches, and, therefore, would exceed the blender capacity. Additional conservatism comes from the assumptions of spherical geometry, full reflection, and no neutron absorptions due to blender components. The assumed UO_2 density of 5 g/cm^3 is also conservative in that the powder density does not exceed 3 g/cm^3 .

The addition of selected hydrogenous additives to a batch of UO_2 in the blender is also safe for the aforementioned reasons if the H/U does not exceed 0.45 and the material (powder lubricant) is fed in at a controlled rate so it is well mixed. An accident assuming the addition of

Fig. 1. Effective multiplication vs moderation and mass.

Fig. 2. Effective multiplication vs moderation and mass.

a second lot of lubricant instantaneously—such that it is only partially blended—has also been evaluated with ANISN and found to result in a maximum effective multiplication of 0.89. As the blending continues, the effective multiplication drops below 0.8. The ANISN input assumed a multiregional spherical system, fully water reflected. All the regions contained 4% enriched UO₂ at a density of 5 g/cm³ with the H/U ratio in each region decreasing from the central region outward, thus simulating various degrees of mixing.

The use of ANISN calculations for this type of system has been benchmarked as reported in Ref. 1 and has shown errors of about $\pm 1\%$.

Nuclear Criticality Safety Controls

Operational safety of the unit is provided by a combination of physical and administrative controls. Physical protective measures include baffling or double containment of all moderator lines in the area, installation of a glove box over the blender loading port, and an interlock system to feed powder additives at a controlled rate and only when the blender is operating. UO₂ and lubricant additions to the blender are limited by administrative controls to assure that both virgin and process recycle material satisfy H/U limits. Typical administrative controls consist of sampling and hydrogen analysis of UO₂ prior to addition to the blender (where the material has been exposed to a "nonmoderation-controlled" environment) and dual approvals prior to any additions to the unit. In addition, composition and available quantities of powder lubricants are strictly limited.

Summary

The nuclear criticality safety of the large-volume powder blender under normal and credible accident conditions has been demonstrated using ANISN and the Hansen-Roach cross-section set. The use of a large moderation-controlled process unit has been shown during operation to be safe and to provide certain process advantages. Under certain conditions, the advantages of such a system outweigh the penalties that accrue from the need for operating within strict physical and administrative limitations.

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2. W. W. ENGLE, Jr., "A User's Manual for ANISN," K-1693, Oak Ridge Gaseous Diffusion Plant (Mar. 30, 1967).
3. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAM-2543, Los Alamos Scientific Laboratory (1961).

Regulatory Perspective of Moderation Control—Low-Enriched Uranium Operations, R. L. Stevenson (NRC)

The objective of this paper is to identify those features of nuclear criticality safety analyses for moderation-controlled, low-enriched uranium systems that receive particular attention in licensing reviews. For certain moderation-controlled operations of long standing, the specific techniques in analysis and justification are well documented and innovation is therefore unnecessary. These practices are:

1. UF₆ storage in OR-30B-type cylinders
2. UF₆ vaporization from large cylinders
3. dry powder storage in large, close-packed arrays
4. pellet-tray depth limits based on close packing
5. dry storage of finished fuel assemblies.

(The analysis and spacing of the fuel assembly storage usually provides for subcriticality under a range of accidental moderation conditions so it may be a misnomer to refer to such storage as a moderation-controlled operation.)

The foregoing moderation-control practices are not completely dependent on engineering or administrative controls to exclude moderator from the fissile material. Certain newer practices have associated technical and administrative problems because the newer practices rely more fully on the exclusion of moderator as the basis for criticality safety. The newer practices are used in the UO powder blending operation and in certain proprietary process steps. The technical problems in establishing the safety of the newer practices relate to:

1. the integrity of the water barrier under normal and accident conditions
2. limiting the sources of water in the process area
3. control of the moderator content of the incoming fissile material and control of the moderator added to the fissile material
4. assurance of uniform distribution of available moderator throughout the fissile material
5. the hygroscopicity of the fissile material
6. control of fissile material density if assumed less than theoretical
7. the calculation of correct mass limits for systems of low moderation.

Maintenance of the integrity of the water barrier will probably call for more control and maintenance of auxiliary systems than otherwise. The techniques for limiting the sources of water in the process area may include, in addition to controls on routinely added moderator, controls on coolant and hydraulic systems and on firefighting.

In blending large masses of dry material, consideration must be given to possible overbatching or maldistribution of added moderator. The hygroscopicity of the fissile material must be considered in setting allowable moderator additions and in evaluating possible accident effects. If fissile material densities are assumed to be less than theoretical on the basis of process history, corresponding controls on relevant process variables should be instituted. There may be theoretical reasons for accepting average fissile densities below theoretical but particle densities are probably near theoretical.¹ Density-moderator relationships are therefore expected to fall on the low density end of the theoretical fissile density-moderator relationship. Assumptions to the contrary require experimental justification. The experience of the ANS-8 work groups developing standards for low-enriched uranium and for mixed oxide (ANS-8.8 and ANS-8.12, respectively) has demonstrated the current technical difficulty of calculating consistent limits for systems of low moderation.

The administrative problems associated with moderation control relate to:

1. training of operators
2. emergency preplanning and training
3. maintenance of water barriers
4. analytical and process controls on moderation
5. maintenance of batch controls and spacing between units.

The listed administrative control items are relevant to almost all operations with fissile material but have greater importance while being more difficult to implement when moderation control is the primary control.

1. J. C. CLAYTON and S. ARONSON, "Some Preparative Methods and Physical Characteristics of Uranium Dioxide Powders," *J. Chem. Eng. Data*, 6, 1 (Jan. 1961).

Data and Analyses for Nuclear Criticality Safety Guidance

Some Investigative Results of the Solid-Angle Method Used in Nuclear Criticality Safety, *J. S. Tang (ORNL)*

The solid-angle method has been used for nearly 20 years^{1,2} to specify safe spacings for subcritical components of fissile materials. This method is based on the concept that neutron interaction is directly proportional to the solid angle subtended among fissile units and can be related to the neutron multiplication factor of an individual unit.

This paper describes the application of the solid-angle method to single- and multiple-array systems. The Monte Carlo criticality program, KENO,³ with Hansen-Roach 16-group neutron cross-section data sets,⁴ was used for the k_{eff} calculations.

For a single array, neutron interaction was investigated for planar arrays of cylindrical units of highly enriched UO_2F_2 and for $UO_2(NO_3)_2$ solutions and for water- and concrete-reflected cubic arrays of spherical U(93.2) metal units. Figure 1a contains results of unit interaction of the three fissile materials. These data show that the "safe-curve" for unit interaction recommended by TID-7016 (Ref. 5) cannot be applied generally to reflected cubic arrays of large numbers of units.

For multiple-array systems, cubic arrays of 216 air-spaced units of U(93.2) metal spheres centered in cubic cells were examined. Three distinct cubic arrays were considered, each composed of 10-, 15-, and 20-kg units. The parameters of each array were selected so the array would be critical if it were totally reflected by water. Nine identical arrays were arranged in a square pattern at the center of a square room enclosed by 406-mm-thick concrete. The total average fractional solid angle Ω_f subtended at the center array was determined by the approximate method described in TID-7016 (Ref. 5). Depending on its location in the room, a single array will have different values of k_{eff} due to the different contributions of neutron reflection from the concrete enclosure. The array k_{eff} is maximum when the array is at a corner of the room and minimum when it is located at the center of the room.

Neutron interaction curves for the nine-cubic-array systems are presented in Fig. 1b. For comparison, the interaction between two square planar arrays of $UO_2(NO_3)_2$ cylinders is also shown. Although the interaction curves for the nine-array system were obtained for U(93.2) metal, they are applicable to 46 different fissile materials.⁶ Due to the generic nature of reflected arrays,⁷ the uranium metal units in the cubic arrays may be replaced by equivalent units of other fissile materials

Fig. 1a. Interaction curves for single-array systems.

Fig. 1b. Interaction curves for multiple-array systems.

without changing the array multiplication factor. This was demonstrated for the uranium oxide material, $U(30)O_2$ at an $H/U = 0.4$.

The results of this investigation suggest that:

1. The safety curve for unit interaction given in TID-7016 (Ref. 5) is applicable to planar arrays but not necessarily to reflected cubic arrays.
2. The solid-angle method does not have general applicability and that specific conditions, such as reflecting environment and array geometry, for using the method should be investigated.

1. H. F. HENRY, J. R. KNIGHT, and C. E. NEWLON, "General Application of a Theory of Neutron Interaction," K-1309 (Nov. 15, 1956).
2. J. A. POND, "Critical Geometries for Bare Cylinders," GAT-189 (July 1956).
3. L. M. PETRIE and N. F. CROSS, "KENO IV—An Improved Monte Carlo Criticality Program," ORNL-4938 (Nov. 1975).
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5. *Nuclear Safety Guide*, Subcommittee 8 of the ASA Sectional Committee N6, Project 8 of the ANS Standards Committee, TID-7016, USAEC, rev. 1, p. 35 (1961).
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Fission and Explosive Energy Releases of PuO_2 and PuO_2 - UO_2 Assemblies, G. E. Hansen, C. C. Byers, J. J. Koelling (LASL)

For the purpose of determining off-site effects of criticality accidents it has been normal practice to postulate accidents in various configurations and media within a plant. For any specific proposed accident it is the fission energy estimate that ultimately determines the dose at the boundary since all other factors, i.e., fission-product yields, decay rates, dilution effect, breathing rates, plateout, etc., involved in the dose estimate are relatively well understood.

The fission and explosive energy release from critically accidents involving liquid and metal assemblies has been well established from both accidents and experimentally induced excursion data; however, very little work has been performed to date in dry or near-dry powder assemblies. With the possible advent of plutonium recycle it has become increasingly apparent that the upper limits of energy release for PuO_2 and PuO_2 - UO_2 assemblies similar to that found in nitrate to oxide conversion plants and mixed-oxide fuel fabrication plants should be established.

This study focused on PuO_2 assemblies with 0 to 10% (by weight) H_2O content and an oxide density of 5.0 g/cm^3 .

Density effect verification was performed through a separate set of calculations over the range from 1.0 to 5.0 g/cm³. Reactivity insertion rates (α) were varied from <1 to 300\$/s. The reactivity insertion rates are very conservative to what would be found under normal mass transfer conditions in any of the plants investigated. The effect of water and concrete on the bare oxide assemblies was also considered. The critical mass of PuO₂ (at 5 g/cm³) with various concentrations of water is shown in Fig. 1. The total number of fissions resulting from various reactivity insertion rates are shown in Fig. 2. For the case of 0 wt% water, air was introduced in the void area to assist in disassembly.

The fission and explosive energy releases were determined with a hydrodynamic-neutronic code, PAD.¹

Fig. 1. Critical mass of PuO₂ × H₂O.

Fig. 2. Total fissions vs reactivity insertion rate.

Neutron multiplication and period calculations were performed with a neutron transport code, DTF-IV.² The Hansen-Roach cross sections were utilized in both the DTF-IV and PAD codes.

Because of the considerable range in the expected isotopic composition of plutonium in the plants investigated,^{3,4} a model composition was chosen which approximates light-water reactor (LWR) fuel and yields conservative results for the various calculations performed. Spherical geometry was chosen for ease of modeling and its inherent conservatism for such calculations. All energy release analysis was performed for initial fission spikes only; multiple spikes were not considered.

Results:

1. The strong neutron source ($\sim 10^2$ n/g · s for our model) provides the means to initiate the fission chain reaction quickly, thus preventing the assembly from achieving a mass much greater than would be required for delayed criticality.

2. For maximum estimated material transfer rates achievable in conversion and fabrication facilities, it was determined that, at best, a maximum ramp rate of a few cents per second was possible; hence, the system was incapable of sustaining a superprompt critical excursion under these "normal" flow conditions.

3. Water is the basic disassembly material even in the case of extremely small amounts ($\sim 0.1\%$ by weight). The water content and subsequent vapor pressures resulting from the water ultimately determine the fission yield during an excursion for a given ramp insertion rate. Without water, the disassembly mechanism is the pressure resulting from the vaporization of the fuel and the air held in the void spaces.

1. W. R. STRATTON, D. M. PETERSON, and L. B. ENGLE, "The Pajarito Dynamics Code with Application to Reactor Experiments," *Trans. Am. Nucl. Soc.*, 15, 819 (1972).
2. K. D. LATHROP, "DTF-IV, A Fortran-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering," LA-3373 (July 1965).
3. License Application, Westinghouse Nuclear Fuels Division, Recycle Fuels Plant, Anderson, South Carolina, July 1973, Docket No. 70-1432.
4. Preliminary Safety Analysis Report, Allied-Gulf Nuclear Services, Barnwell Nuclear Fuel Plant, Addendum #7, Plutonium Product Facility, July 1974, Docket No. 50-332.

Critical Experiments Measuring the Reactivity Worths of Materials Commonly Encountered as Fixed-Neutron Poisons, S. R. Bierman, B. M. Durst, E. D. Clayton (BNWL)

Shipping casks, storage facilities, and process vessels involved in handling fissile materials contain neutron-absorbing materials—either for their neutron-absorbing properties or for structural reasons. These types of systems can be fully utilized, safely, only if the neutron absorbers are properly accounted for in the criticality analyses. The results from measurements on two such materials, copper and copper containing 1 wt% cadmium, were reported on previously.¹ Since that time, the reactivity worths of several other materials, commonly considered for use as fixed neutron poisons, have been

Fig. 1. Effect of poison plates on critical height.

measured. The results of these measurements are presented in this paper and provide a set of clean, well-defined, poisoned critical assemblies that can be used to check calculational techniques and cross-section data.

Two different fuels were utilized in these measurements to provide data on the relative worths of the following materials in systems of differing neutron energy spectrum:

- 304-L steel
- 304-L steel containing 1 wt% boron
- 304-L steel containing 1.5 wt% boron
- Uranium depleted to 0.2 wt% ^{235}U
- Boral
- Cadmium
- Lead

Both sets of fuel were homogeneous mixtures of $\text{PuO}_2\text{-UO}_2\text{-polystyrene}$ having Pu concentrations typical of the LMFBR fuels. One set had an atomic H:Pu+U ratio of 30 and contained 15 wt% PuO_2 in the oxide. The second set had an atomic H:Pu+U ratio of 3 and contained 30 wt% PuO_2 in the oxide. (A complete description of each fuel can be found in Ref. 1.) The critical assemblies consisted of rectangular parallelepipeds, fully reflected with 6 in. of Plexiglas (a methacrylate plastic), and containing a single poison-plate region of varying thicknesses. The poison-plate regions were horizontal in each assembly and had essentially the same cross-section dimensions (40.72×40.72 cm) as the assembly.

Typical data results from the measurements are presented in Fig. 1, where the relative effect that each material had on the critical height of the 30 H:Pu+U fueled assembly is shown as a function of poison-plate thickness.

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Measurements of Surface Multiplication Using a Fission Chamber, R. E. Wilson (Allied Chem), E. D. Clayton (BNWL)

In the nuclear criticality field a significant need exists for a parameter that can be both measured in subcritical systems and calculated analytically. The comparison between these could be used to check cross-section sets, calculational models, and neutronics theory. Of the analytic parameters currently in use as an index of criticality of systems not at or near the critical state, surface multiplication (M_s) appears to be the one with the best potential of being measurable.¹ M_s can be defined as the number of neutrons that leave an assembly of nuclear materials due to a single incoming neutron. Preliminary measurements of M_s were reported^{2,3} for far subcritical and unreflected cubes of plutonium-polystyrene using indium foils as the neutron detector. These measurements had large uncertainties but demonstrated the feasibility of measuring M_s .

In the experiments reported here, a 22-cm^2 flat-plate thorium fission chamber was developed and constructed for use as a neutron detector. A number of gasket sizes were used to define the active surface of the detector for measurements with various size cubes. This detector would cover the entire face of each side of the cubic assemblies measured. The external fission spectra neutron source used was $20\text{ }\mu\text{g}$ of ^{252}Cf . The detector was used to measure the leakage neutrons from four sides (the cube face toward the external neutron source (front),

Fig. 1. Inverse surface multiplication of cubes.

the opposite face (back) and the two side faces of each assembly size]. The top and bottom neutron leakage was assumed the same as the side values. The effect of reflected neutrons from the floor was accounted for with a separate calculation. The incoming neutrons from the neutron source were also measured. The experimental count rates were weighted for differing spectra and angle of incident to detector for incoming and outgoing neutrons to obtain source and leakage neutron currents. These weighting factors were found by calculations.

The measurements were made on cubes of plutonium-uranium-polystyrene. Sources of measurement error include the positioning errors of the source, gasket, and

cube; variations in the detector system and counter gas flow; and counting statistics. Three independent sets of measurements were made on each of four cube sizes. The results are shown in Fig. 1 as the inverse surface multiplication is plotted against cube length. The solid lines are from calculations using the GEM Monte Carlo Code.⁴ The expected statistical variation on the calculated values of M_S is $<1\%$. The broken line is an extension of GEM results to critical cube size. Note that for all cube sizes, the average is well within the expected error of the GEM calculated values. This agreement supports confidence in the weighting factors used to find M_S from the count-rate data.

It is noted that measurements were on assemblies containing from 0.1 to 7% of a critical mass. It is concluded that surface multiplication can be measured with accuracy and precision and as both an experimental and calculational parameter is a useful and valid index of criticality for subcritical assemblies.

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2. R. E. WILSON, R. J. HALL, and E. D. CLAYTON, *Nucl. Sci. Eng.*, **39**, 119 (1970).
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4. P. J. HEMMINGS, "The GEM Code," AHSB(S)R 105, United Kingdom Atomic Energy Authority (1967).

Criticality Data and Experience Applicable to Nuclear Criticality Safety Guidance

Effective Neutron Isolation Thickness Values Obtained from Reflector Savings Data, B. W. Howes, S. R. Bierman, E. D. Clayton (BNWL)

The accurate determination of minimum material thicknesses required for isolating multiplying media from neutron interaction is a problem of considerable interest in the design of shipping containers and storage vaults. The net amount of storage space required for nuclear materials can be effectively reduced and criticality controls improved through the proper use of neutron-absorbing and -moderating materials near the fuel boundary. A series of critical approach experiments was conducted at the Battelle-Northwest Critical Mass Laboratory utilizing a split-table machine¹ to measure effective isolation thickness values for several materials. Analytical methods were developed for investigating the effects of fuel geometry and material compositions on the experimental results. As a result of the data analysis, a simplified experimental technique was developed for determining isolation thickness values from the reflector savings data. Also, it was determined that many of the currently acceptable isolation thickness values are low by at least a factor of 2.

Slab interaction calculations were performed utilizing the albedo reflection options of the diffusion theory code, HFN,² and the discrete ordinates code, DTF-IV.³ Both the HFN and DTF-IV calculations were made using

18-energy-group GAMTEC⁴-averaged cross sections that had been processed from ENDF/B-III with the FLANGE⁵ and ETOG⁶ codes. Critical dimensions of two identical fuel regions separated by varying thicknesses of isolator material were obtained by performing two-region, one-dimensional calculations with an albedo of one at the centerline of the isolator region. Corresponding calculations were made of end-reflected systems by utilizing mirror reflection about the centerline of the fuel region. It was found that the test material thicknesses required for full reflection of the experimental fuel systems were essentially one-half that required for isolating two critical systems of identical cross-sectional dimensions. These results appear to be physically realistic. In addition, for the 15-cm Plexiglas end-reflected experimental system, excellent agreement was obtained between theory and experiment. A k_{eff} of 1.003 was obtained with the P_1 neutron transport theory calculations using four orders of quadrature, and a k_{eff} of 1.011 was obtained with diffusion theory calculations.

Good agreement was also obtained between experimental reflector savings data⁷ and diffusion theory calculations as a function of Plexiglas end-reflector thicknesses if appropriate transverse neutron leakage corrections and constant interval spacings between calculational points were utilized in the reflector region as well as in the fuel region. All transverse buckling data were computed by the GAMTEC code for input to the HFN code. It was found that the reflector savings data obtained from HFN calculations converged on an asymptotic value

TABLE I
Experimental and Analytical Neutron Isolation Thickness Values*
Obtained for Mixed PuO₂-UO₂ Polystyrene Fuel Systems

Material	Experimental Results, cm		Diffusion Theory Calculations, cm	
	Previous Results (8)	This Work	Experimental System	Infinite Slab System
Concrete	25.0 ± 2.0	56 ± 4	55	104
Lead	26.0 ± 2.0	56 ± 4		
Borated Concrete	17.5 ± 0.5	54 ± 4	53	113
Boroplaster		32 ± 2	30	52
Stainless Steel (304-L)		56 ± 4	55	115
Boron Steel (304-L)		50 ± 10	48	91
Plexiglas			31	45

*Neutron isolation thickness is defined here as that amount of material required to limit the reactivity effect of a separated fuel system to <0.1% $\Delta k/k$.

in a manner that could be fit to an exponential function in the region of interest. Since the calculated values are in good agreement with the experiments, these results suggest that it is possible to extrapolate limited experimental information to obtain effective neutron isolation thickness values.

Table I presents the experimental results obtained from an earlier series of measurements⁸ performed at BNW using a slightly smaller fuel assembly. These data are compared with the current experimental results and analytical data. The factor of 2 difference in experimental results is due primarily to a different interpretation of the experimental data. The earlier results are in good relative agreement with the present work if the isolation thickness value is taken to be twice that thickness for which the fuel core critical length approaches an asymptotic value. In terms of reactivity effects, for the small experimental fuel systems, a factor of 2 increase in isolator thickness would result in a 1% decrease in k_{eff} . However, for two infinite fuel slabs, the additional factor of 2 increase in the isolation thickness value is worth $\sim 10\% \Delta k/k$.

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Subcriticality Design Considerations for LMFBR Spent Fuel Shipping Casks, J. S. Philbin, S. A. Dupree (Sandia Labs)

During storage or transport it is imperative that assemblages of fissile material not go critical under either normal or credible accident circumstances. Criticality studies have, therefore, been included as an impor-

tant safety-data base in the liquid-metal fast breeder reactor (LMFBR) spent fuel shipping cask technology development program at Sandia Laboratories.

The purpose of the present study was to examine the effects of geometry, coolant, and loading changes (along with shield/structural material selection) on the neutron self-multiplication of two types of LMFBR fuel subassemblies placed in a limited range of configurations pertinent to shipping cask application. Fuel subassemblies typical of the Clinch River Breeder Reactor (CRBR) have formed the basis for most of the studies. However, because it is apparent that a full-scale commercial breeder reactor will utilize larger subassemblies of lower ^{239}Pu enrichment than will CRBR, calculations were also performed on hypothetical subassemblies larger than those of CRBR.

A three-dimensional (3-D) Monte Carlo criticality program, KENO IV,¹ and the KENO 16-group Hansen-Roach cross-section library were used to obtain the results reported here.

A hypothetical cask design for transporting seven CRBR subassemblies has been developed to serve as a reference. This reference design is essentially a scaled-up version of a B_4C cask design discussed in Ref. 2. In this cask, up to seven subassemblies and coolant are located in a central region in a "spaced" hexagonal array (i.e., not a close-packed array). A structural region surrounds the fuel/coolant section, followed by concentric radial regions of: depleted uranium (10 cm thick), boron carbide (18 cm), and outer steel structure (5 cm). The cask has an outside diameter of 1.22 m.

Figure 1 shows the relationship between k_{eff} and the total mass of heavy metal loaded into the reference cask for sodium and a hydrogenous coolant (Dowtherm A). For the two coolants, there is an appreciable difference between the masses required to produce any given value of k_{eff} . Of course, reasonable amounts of neutron absorbers can be introduced in the central cask structure to reduce self-multiplication, and hence increase the payload capacity, of a Dowtherm-cooled cask.

Fig. 1. k_{eff} vs cask heavy-metal inventory of CRBR fuel. Beginning of equilibrium cycle isotopic mix. ^{239}Pu = 30.2% of all heavy atoms.

Changes in the neutron self-multiplication resulting from rearranging a given number of subassemblies within the cask, e.g., by changing the distance (or pitch) between subassemblies within a reasonable range, were found to be smaller than the change resulting from the addition or removal of a subassembly. For example, in a spaced-hexagonal array of seven hypothetical commercial LMFBR subassemblies,^a the negative slopes of " k_{eff} vs center-to-center spacings between subassemblies" calculated for various coolants were: 0.017/cm (Na), 0.013/cm (gas or void), and 0.011/cm (Dowtherm). These slopes are reasonably constant for center-to-center spacings between closest packed and 18 cm for Na and gas coolants, and above 16 cm for Dowtherm coolant. The corresponding changes in k_{eff} per subassembly are 0.038 (Na) and 0.18 (Dowtherm) at a pitch of 16.36 cm.

These and other correlations examined³ are expected to be useful in making decisions related to shipping cask designs.

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^aThe hexagonal subassemblies (13.5 cm across the flats) were modeled as equivalent cylinders with diameters of 14.16 cm for the spaced hexagonal arrays and as a cylinder equivalent to seven hexagonal subassemblies for the closest packed cases.

Criticality Safe Tank Design Analysis, A. D. Summers, R. E. Wilson (Allied Chem)

In the reprocessing of highly enriched uranium at the Idaho Chemical Processing Plant (ICPP) corrosive concentrated uranium solutions are generated from process upsets, custom processing, and sample residues. A study was performed to determine an alternative critically safe storage system to the one in present use for these solutions.

Presently, large-diameter tanks with fixed boron stainless-steel plates are being used.¹ Acidic process solutions and highly corrosive decontaminating solutions capable of corroding the boron stainless-steel plates are

used in the vessels. Because only a small portion of the poison plates is accessible to allow periodic verification of plate thickness, these tanks are to be replaced with vessels of a more desirable critically safe design.

The major design constraints are volume, criticality safety (including neutron poison verification), decontamination, corrosion resistance, mixing and sparging capability, and process cell size.

Design concepts without neutron poison evaluated were (a) banks of small-diameter tanks, (b) an annular tank, and (c) a slab tank. A bank of small-diameter tanks, although shown safe by criticality calculations, would be too difficult to sparge and obtain a homogeneously mixed sample. Calculations showed that annular tanks without neutron poison would not fit safely in the process cell because of neutron interaction between tanks.

Slab tanks, unless thick walls or tie rods are used, have the inherent potential for wall deformation (bowing) which can significantly affect criticality safety. Tie rods and connections are susceptible to corrosion.

Design concepts with neutron poison evaluated were (a) an annular tank, (b) a tank containing a removable cartridge of poison plates, and (c) a tank containing removable neutron poison tubes. In the annular tank, a center region contains solution and each alternating annulus contains neutron poison. This, in effect, isolates the solution regions. Although shown to be critically safe, this design presents difficulty in neutron poison verification. Tanks with a single annulus with accessible neutron poison around the outside would be acceptable. The tank design containing a removable cartridge of neutron poison plates was eliminated because of the difficulty of decontaminating and verifying plate thickness. Although superior to the present design utilizing fixed neutron poison plates, it is still considered unsatisfactory.

The poison tube concept consists of a large-diameter tank containing individual small-diameter stainless-steel tubes incorporating boron poison. Drilled plates could be used to locate and support the poison rods in the tank. Calculations showed this to be a favorable critically safe design and the individual neutron poisoned tubes could easily be removed, decontaminated, and inspected for neutron poison verification. Also, the tank itself could easily be decontaminated and adequately sparged.

The conclusions of this preliminary study are that of the design concepts analyzed, within the specified criteria, the large-diameter tank containing neutron poison tubes and the single annulus tank with accessible neutron poison provide the best overall critically safe designs for ICPP.

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Spent Fuel Storage

B&W-Designed Close-Spaced Fuel Storage Racks, J. S. Olszewski, R. H. Clark (B&W), invited

In response to the existing industry concern over fuel storage capacity, B&W has redesigned their fuel storage racks to give closer spaced storage capabilities. The B&W fuel storage racks are intended for temporary underwater storage of new fuel assemblies and long-term storage of irradiated fuel assemblies. In this redesign our objective was to offer an optimum balance between maximum storage and minimum cost, with an adequate safety margin. Our goal was a "standard" design independent of site conditions, fuel assembly design, and building. Design conditions were:

1. B&W Mark "B" and Mark "C" fuel assemblies
2. UO_2 at 3.5 wt% enriched or MO_2 at 4.5 wt% enriched
3. $k_{\text{eff}} \leq 0.95$.

To achieve good geometric representation of the fuel assemblies and storage racks, KENO Monte Carlo computer codes were used. The UO_2 studies used KENO II with a 16-group neutron structure and the standard Hansen-Roach library. The MO_2 studies used KENO IV with a 123-group structure. The KENO II and KENO IV calculations were benchmarked against unpublished ORNL critical experiment data and Saxton/WREC critical experiments, respectively. Parametric studies were made of k_{eff} versus can spacing, thickness, and boron concentration. Some results of these studies are given in Fig. 1.

Fig. 1. Calculated k_{eff} vs assembly spacing for selected design conditions.

The data are for the more restrictive of the UO_2 and MO_2 cases.

Systematic errors caused by biases in the calculational techniques were identified by comparing (benchmarking) calculated values with experimental values. For UO_2 calculations systematic errors were found to be conservative by about 2% in k_{eff} , but no credit was taken for

Fig. 1.

this bias. Calculations for MO_2 were found to be nonconservative and were adjusted upward by 0.0026 in k_{eff} to assure conservatism. Random errors caused by statistical errors in Monte Carlo calculations, and material and fabrication tolerances were combined by the square root of the sum of the squares before being combined with applicable systematic errors.

The individual fuel cells are constructed by butting two "C" channels together to form a box assembly (see Fig. 2). A 10-gauge plate is positioned along the butt joint in the interior of the box cell and intermittently plug welded to the box cell. This welded plate assures cell mechanical integrity, increased nuclear fuel isolation, and provides square interior corners which protect the spacer grid corners during fuel handling and seismic events. A "funnel" top and a $\frac{3}{16}$ -in. clearance between the fuel assembly and the can assures compatibility with the fuel handling equipment even with deformed assemblies. Individual cells are welded to supporting box beams at the top and bottom to form standard 6×8 modules. Type 304 stainless steel is used throughout to assure a 40-yr life in boric acid water.

Static and seismic loads are transmitted by bolts to floor embedments and/or to the pool walls by horizontal compressive beams. During a safe shutdown earthquake (SSE), stored fuel integrity is maintained; following an SSE, fuel is removable. Following a $\frac{1}{2}$ SSE, stored fuel is reusable.

The cells are elevated above the pool floor so pool water flows by natural circulation beneath the cells, through an opening in the bottom of each cell, and up through the can and fuel assembly. This natural circulation provides sufficient cooling to preclude surface boiling during normal pool temperatures, and bulk boiling during extreme upset conditions.

We conclude that with a lattice spacing of 13 in. and a 10-gauge can (0.134 in.) of unborated stainless steel, all criticality safety and seismic requirements are met. This basic design is a conservative approach to the fuel storage problem which utilizes the best current technology, and offers an optimum balance of the design variables; the design is described in the B&W Standard Plant Safety Analysis Report (B-SAR-205) as being standard for future B&W plants. The same basic design may also be used to retrofit operating plants and increase storage capability from the conventional $1\frac{1}{2}$ cores (1-yr discharge) to over $3\frac{1}{2}$ cores (8-yr discharge).

For some plants where variables such as fuel enrichment, structural steel, and fuel assemblies are different, we have offered closer spaced racks using unborated stainless-steel cans. As shown by Fig. 1, cell spacing may be reduced considerably by using poisoned rack material; however, at this time B&W does not consider boric acid steel to be economically attractive nor other materials such as Boral to be metallurgically acceptable.

Fig. 2. Fuel storage cell structures.

Measurement and Calculation of Reactor Parameters—II

Analysis of BNL 1% ^{235}U Exponential Experiments with ENDF/B-IV Data, J. J. Weiss, B. K. Malaviya (RPI)

The continuing interest in developing the data base and calculational methods for the design analysis and evaluation of light-water reactors has led us to analyze a series of light-water exponential experiments¹ performed at

Brookhaven National Laboratory (BNL). The objectives of this work are (a) to evaluate the cross-section data for ^{235}U and ^{238}U in ENDF/B-IV, and (b) to test the validity of specific analytical models used in nuclear design.

The HAMMER² computer code, which utilizes integral transport theory, was chosen as the computational method; resonance self-shielding is accounted for by the Nordheim approximation in this version of the code. The data base used in the analysis involved a 30-group thermal cross-section set generated from ENDF/B-IV tape and the FLANGE³ code, and a 54-group epithermal cross-section set in MUFT⁴ format processed through the BNL version of ETOG.^{5,6}

BNL exponential experiments involving four different ^{235}U enrichments were examined (Table I). The lattices studied involved four different rod sizes, moderator-to-fuel volume ratios ranging from 1.0 to 4.0, and several different lattice pitches for each enrichment. The cladding material in all lattices was aluminum. Trace amounts of ^{234}U and ^{236}U present in the lattices were accounted for in the analysis, also using the ENDF/B-IV data. The measured bucklings were determined by either flux shape analysis or variable loading analysis techniques,⁷ depending on the data available or evaluated.

The calculated values of k_{eff} , using ENDF/B-IV data and the HAMMER code system, are shown in Table I. Overall, the experimental results are predicted well by the data and calculational method employed. In all cases studied, calculated values of k_{eff} decrease as the moderator-to-fuel volume ratio increases (for a constant rod size) or as the lattice pitch increases. A typical result is shown in Fig. 1. Thus, it is seen that the use of ^{235}U and ^{238}U cross-section data from ENDF/B-IV leads to reasonable results, but systematic error trends in predicting LWR benchmark experiments.

Fig. 1. Typical variation of calculated k_{eff} with lattice pitch.

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TABLE I

1.016% ²³⁵ U				
V_M/V_F	Diam. (in.)	Pitch (in.)	Measured B_M^2 (m ⁻²)	Calculated k_{eff}
1.5	0.250	0.4190	12.29	1.0037
2.0	0.250	0.4516	19.95	.9989
3.0	0.250	0.5105	24.348	.9940
4.0	0.250	0.5633	20.34	.9936
1.0	0.387	0.5674	3.23	1.0233
1.5	0.387	0.6244	19.70	1.0105
2.0	0.387	0.6767	29.02	.9988
3.0	0.387	0.7706	31.39	.9916
4.0	0.387	0.8542	25.68	.9857
1.0	0.600	0.8537	9.90	1.0311
1.5	0.600	0.9444	29.63	1.0064
2.0	0.600	1.0273	36.07	.9972
3.0	0.600	1.1754	33.15	.9898
4.0	0.600	1.3070	20.96	.9860
1.027% ²³⁵ U				
1.334	0.750	1.1293	28.90	1.0159
1.584	0.750	1.183	34.70	1.0077
1.834	0.750	1.2371	37.50	1.0027
2.334	0.750	1.336	36.73	.9979
2.834	0.750	1.4285	32.88	.9919
3.834	0.750	1.5977	18.60	.9856
1.143% ²³⁵ U				
1.5	0.250	0.4190	21.16	1.0070
2.0	0.250	0.4516	31.80	.9956
3.0	0.250	0.5105	39.05	.9870
4.0	0.250	0.5633	36.05	.9868
1.0	0.387	0.5674	12.03	1.0257
1.5	0.387	0.6244	31.21	1.0082
2.0	0.387	0.6767	42.26	.9947
3.0	0.387	0.7706	46.18	.9873
4.0	0.387	0.8542	40.50	.9841
1.0	0.600	0.8537		1.0275
1.5	0.600	0.9444	40.23	1.0093
2.0	0.600	1.0273	48.22	.9984
3.0	0.600	1.1754	47.12	.9882
4.0	0.600	1.3070	36.03	.9840
1.299% ²³⁵ U				
1.0	0.387	0.5674	20.98	1.0280
1.5	0.387	0.6244	40.51	1.0135
2.0	0.387	0.6767	52.19	1.0011
3.0	0.387	0.7706	59.25	.9892
4.0	0.387	0.8542	54.69	.9857
1.0	0.600	0.8537	32.11	1.0265
1.5	0.600	0.9444	51.87	1.0097
2.0	0.600	1.0273	61.08	.9982
3.0	0.600	1.1754	60.99	.9889
4.0	0.600	1.3070	50.38	.9853

Evaluated Nuclear Data

ENDF/B-IV Eigenvalue Calculations for UO₂ and Mixed-Oxide Critical Lattice Experiments, Odelli Ozer (EPRI), Sidney Fiarman (Stanford Univ)

A series of water-moderated critical lattice experiments has recently been carried out at Battelle Northwest Laboratories under EPRI support.¹ The purpose of these experiments has been to provide benchmark information for testing nuclear cross-section data and current methods of analysis used in LWR fuel cycle calculations.

The experiments consisted of six lattices fueled with UO₂-2.35% ²³⁵U, and six lattices fueled with UO₂-2 wt% PuO₂ (8% ²⁴⁰Pu). The fuel was a vibrationally compacted powder with an effective PuO₂ grain diameter of 25 μm. Each series of six experiments included three different lattice pitches with either clean or borated H₂O as moderator, providing moderator-to-fuel volume ratios for optimally moderated as well as under- and over-moderated cases.

The procedure followed in the present eigenvalue calculations was entirely based on ENDF/B-IV cross sections processed into multigroup form by the ETOG-III (Ref. 2) and FLANGE-II (Ref. 3) codes. The epithermal hydrogen cross sections were adjusted to account for proton binding effects and upscattering into the low-energy ²⁴⁰Pu resonance region.⁴ All cross sections were further reduced into a four-group form by the BNL version of the Hammer code⁵ using a B₁ approximation to the spectrum in each lattice. Bucklings put into the Hammer calculations were obtained from least-squares fits to the measured axial and radial power distributions. However, due to the scatter in the measured data, a few of the estimated radial bucklings contained an uncertainty of several percent. In the resonance range (3.35 keV to 0.625 eV), the Hammer absorption rates in ²³⁵U and ²³⁸U have been modified by the introduction of calculated correction factors. In the case of the UO₂ lattices, these factors were obtained by matching zero-buckling Hammer absorption rates with those obtained from the continuous-energy Monte Carlo REPC code.⁶

The four-group constants obtained in the above manner have been used as input into a two-region one-dimensional discrete ordinates transport calculation using the ANISN code.⁷ These calculations, which used an explicit radial representation of the core and reflector regions, were carried out to remove the strong dependence of the eigenvalues calculated with Hammer on the estimated radial bucklings. The axial leakage was accounted for by the addition of a DB₂ term to the absorption cross sections.

The calculated values are shown on Table I. Except for the very tight and undermoderated lattices for which an improved leakage treatment may be required, the calculated eigenvalues are in excellent agreement with the measurements.

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TABLE I
Calculated Eigenvalues for the Battelle UO₂ and Mixed-Oxide Critical Lattices

Lattice i.d.	Critical Number of Rods	Volume Ratio V_m/V_f	Boron Concentration (wppmB)	Axial Buckling (m ⁻²)	Radial Buckling (m ⁻²)	Hammer- k_{eff}	Hammer- k_{eff} (Res. Corrected)	ANISN- k_{eff}
UO ₂ (2.35 wt% ²³⁵ U)								
U-L175	708	1.196	0.9±0.2	8.471	60.649	.9737	.9801	.9891
U-L119	1201	"	463.8±1	8.587	40.807	.9908	1.0013	1.0041
U-L66	383	2.408	0.5±0.2	9.351	74.784	.9931	.9968	1.0010
U-L73	1201	"	568±1	9.166	29.861	.9988	1.0032	1.0021
U-L162	342	3.687	0.9±0.2	8.967	64.551	1.0014	1.0036	1.0001
U-L143	885	"	285.8±1	9.007	30.119	1.0004	1.0018	1.0009
UO ₂ -PuO ₂ (2 wt% PuO ₂)								
U-L266	469	1.195	1.7±0.1	9.091	62.186	.9935		.9817 ^a
U-L250	761	"	680.9±2	9.381	49.934	.9843		.9908 ^a
U-L189	195	2.527	0.9±0.2	9.322	101.678	.9894		1.0008 ^a
U-L212	761	"	1090.4±2	9.487	35.390	.9976		.9979 ^a
U-L282	160	3.641	1.6±0.1	9.842	94.396	1.0109		1.0057 ^a
U-L232	689	"	767.2±2	9.480	31.342	.9963		.9973 ^a

^aThe mixed-oxide calculations do not include the Monte Carlo Resonance Correction.

Safeguards: Measurements, Accountability Systems Analysis

Operation of a Nuclear Test Gauge at Low Multiplications, *N. P. Baumann (DuPont-SRL)*

Analytic studies and field tests were made to demonstrate that the nuclear test gauge¹ (NTG) at the Savannah River Plant could be operated at very low multiplications with improved criticality safety, throughput, precision, resolution, and convenience of operation.

The NTG is a subcritical multiplying facility with H₂O moderator and 2.54-cm-diam fuel slugs of 5 wt% ²³⁵U in aluminum alloy at a 4.285-cm triangular pitch. The core is 61 cm long with a normal diameter of 27 cm. Components are tested by passing them through an axial test port 11.63 cm in diameter and comparing ion-chamber response from changes in neutron source multiplication to changes from known standards.

For the low-k tests, the NTG was modified by (a) reducing the reference k_{eff} , (b) upgrading the data acquisition system, (c) changing the mode of component assay from incremental to continuous, and (d) replacing the extended neutron source with a compact ²⁵²Cf source.

k_{eff} was reduced from 0.96 to 0.84 by reducing ²³⁵U inventory from 3.7 to 2.3 kg, primarily for criticality safety. Computations supported by experimental tests indicate that, at the resultant low reactivity, all hypothesized test-port misloadings (including multiple events such as flooding of the test port when unauthorized high-concentration fuel is inserted) leave the facility at least 5% subcritical. The reduced ²³⁵U content of the core has operational advantages as well. The effect of lower neutron multiplication is to reduce significantly the magnitudes and time constants of errors due to non-equilibration of delayed neutrons. This effect is shown in Fig. 1 for a typical SRP component (worth = 2% Δk) and a typical precision (0.1% of Δk). The error due to non-equilibrated delayed neutrons assumes no corrective algorithm. The data acquisition error is based on the system described below and a neutron source strength of $1.6 \times 10^8 \text{ s}^{-1}$.

The upgraded data acquisition system included four ¹⁰B-lined current ionization chambers, a pulse-generating current integrator,² and a minicomputer-based data storage and processing system. Ion-chamber current data are acquired continuously, minimizing errors due to fluctuations in neutron flux. Sensitivity tests using pile oscillator techniques indicate that the system is capable of detecting $\Delta \rho_a$ differences of 0.01 cm² in a time interval of several minutes.

Assay by continuous uniform motion of components has several advantages over the incremental mode. It provides a detailed profile as well as an integral assay of fissionable isotopes for safeguards purposes. It also minimizes errors due to delayed neutrons, especially those due to transport of precursors in fissionable components.

The compact ²⁵²Cf neutron source can be inserted without disrupting the basic core lattice pattern. In addition, the compact source and the lower multiplication combine to reduce the axial resolution (FWHM) to just under 20 cm, about half the former value.

Typical data obtained with the modified facility are shown in Fig. 2 for a wide range of ²³⁵U concentrations in ²³⁵U-Al alloy tubes ~9.0 cm in diameter. The index defined in Fig. 2 is independent of source strength and is

closely proportional to Δk , the reactivity worth of the component being assayed. The uncertainty of the index is ± 0.00015 , which translates to $\pm 0.13 \text{ g/m}$ at low concentrations, 0.33 g/m (0.12% of total) in the midrange, and 0.50 g/m (0.10% of total) at the upper end. Uncertainty of concentrations is much less than the scatter that is due primarily to small nonuniformities within the tubes.

The results demonstrate that the low-k method of operation of an NTG is safer and more convenient than the former (high-k) method.

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2. Model 1000A Current Integrator, Brookhaven Instruments Corporation, Austin, TX.

Fig. 1. Response of a multiplying system to a step change in reactivity.

Fig. 2. Measured response of a low-k NTG to ²³⁵U content of aluminum-alloy tubes.

Monte Carlo Methods—I

Applications of the Probability Table Method to Practical Problems, *E. F. Plechaty, D. E. Cullen (LLL), L. Levitt (BNL), invited*

Background

The probability table method (PTM)¹ was developed at Atomics International to account for resonance self-shielding effects in the unresolved resonance region without having to use cross-section ladders directly in Monte Carlo calculations. In the PTM, cross-section ladders are used to define the probability of successive cross-section ranges.² By using the narrow resonance approximation in the unresolved resonance region in Monte Carlo calculations, the probability tables may be independently sampled following each collision; the cross section after collision is uncorrelated to the cross section before collision. The method was implemented in the VIM³ code and thoroughly tested. It was shown to represent the unresolved resonance region as well as the data permitted while achieving computational efficiency in severely limited space.

At Livermore, the objective of using PTM was to use a continuous energy Monte Carlo code⁴ in which the cross sections are given as multigroup constants and yet to account for self-shielding within each group. To achieve this objective, the probability table method was extended to deterministic methods, and the relationship between PTM and the linear Boltzmann equation was demonstrated.^{5,6} The method was applied to all energy ranges⁶ (not just the unresolved) and to photon transport by analogy between Bonderenko, Planckian, and Rosseland averages.⁷ It was demonstrated that by exactly conserving important moments of the evaluated cross section, probability tables can be derived directly from self-shielded multigroup libraries⁸; the number of cross-section bands⁶ required is usually only two or three, and at most, four bands.⁹ Compared to the use of ladders, this represents a large reduction in complexity and computer costs. The narrow resonance approximation⁶ normally used in the probability table method was eliminated; the method was extended to consider intermediate and wide resonances.⁸

Applications of the Probability Table Method

The ALICE Monte Carlo Neutron Transport Code uses the probability table method for all 175 energy groups which span the energy range from thermal to 20 MeV. The code is used for criticality as well as shielding problems.

Criticality Problem

The TART,¹⁰ ALICE, and MCN¹¹ codes were used to determine the critical radius of a sphere of UH₃, where the ²³⁵U enrichment varied from 20 to 100%. All codes derived their cross-section information from the same evaluated library.¹² The TART and ALICE codes use continuous energy neutrons and the same 175-group cross-section structure, while MCN is continuous in both neutron energy and cross-section representation. The results¹³ of these calculations are in Fig. 1. The ALICE and TART codes were used to calculate 41 fast¹⁴ and 4 slow experimental critical assemblies.^{15,16} Forty of the fast assemblies have *K*_{eff} between 0.984 and 1.014 for

ALICE and TART. The remaining nickel-reflected assembly gave a *K*_{eff} of 1.004 for ALICE and 1.018 for TART. For the slow assemblies,¹⁷ ALICE has *K*_{eff} in the range 0.843 to 0.901, while ALICE has 0.957 to 0.996.

Shielding Problem

The shielding use of the PTM is illustrated by calculating the transmission through a 30-cm iron plate. The source was normal to the slab and distributed as 1.0/E from 20.0 MeV to 1 keV. Both the total and uncollided flux were calculated. The results of both Monte Carlo and theoretical calculations are given in Table I. The uncollided flux calculated by ALICE is 7% less than the theoretical calculation, whereas the TART (175 group) is 90% less. The differences in the total flux between ALICE and MCN are negligible, while TART (175 group) is 16% less.

For the problems investigated, the ALICE code can calculate both shielding and criticality problems as accurately as the pointwise code MCN.

Fig. 1. Critical radius of UH₃ sphere ($\rho = 11.0 \text{ g/cm}^3$) vs ²³⁵U/²³⁸U enrichment.

TABLE I

Calculated Flux/(Source Neutron)

Method	Uncollided Flux	Total Flux
Theoretical	0.00478	
MCN		0.157
ALICE (175 group)	0.00443 ± 0.00006	0.158 ± 0.001
TART (2020 group)	0.00430 ± 0.00016	0.156 ± 0.001
TART (175 group)	0.00049 ± 0.00005	0.1305 ± 0.0013

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Training for Purposes of Nuclear Criticality Safety

A Training Program for Supervisors and Senior Staff Personnel, *David R. Smith (LASL), invited*

At the Los Alamos Scientific Laboratory the need has been recognized for a program to provide an improved appreciation of criticality safety fundamentals to supervisors of fissile material processes. Many of the senior people who have been with the Laboratory for many years are now retiring. These people had the opportunity to develop their familiarity with criticality safety considerations as operations expanded from the relatively small scale which was characteristic of the early days. Their successors are denied this opportunity, and some feel the need for an educational program having more depth than is appropriate for every employee.

Supervisors and senior staff members should have enough understanding of nuclear criticality that they feel comfortable answering simple questions from new employees, and provide some of the safety orientation needed by such employees. In an attempt to provide this understanding, we have developed a course consisting of six hours of classroom lectures and six hours of laboratory work. Classes of from 12 to 16 participants are preferred.

The first two-hour lecture session starts with a discussion of the fission process, the production of neutrons during fission, and the role of neutrons in inducing fission. Cross sections are introduced as measures of reaction probabilities, including capture-to-fission ratios. This leads into a discussion of fast fissioning systems, followed by neutron slowing down, moderator effects, and the criticality of solutions. Mixtures of finely divided fissile material, such as foil, wire, or powders, in moderating media are discussed as pseudo-solutions.

The introduction of neutron lifetimes and delayed neutrons attempts to prepare the student for a discussion of the kinetics of near-critical systems. This, however, has been found to be a difficult concept to explain.

The first lecture concludes with a presentation on neutron leakage, shape factors, and reflector effects.

A tabulation of minimum critical values, including mass, volume, dimension, concentration, and enrichment, starts the second lecture, which continues with the influence of density and dilution of fissile material. The balance of the session is devoted to interaction between subcritical units, the criticality of arrays, array moderation and reflection effects, and the effective isolation of units by neutron-absorbing materials.

The third lecture session covers accident experience and safety practices. Several criticality accidents are examined in sufficient detail to take advantage of this means of illustrating basic considerations, and to provide a perspective on accident consequences. The legal framework of criticality safety is outlined by a review of policies as presented in the Code of Federal Regulations, the ERDA Manual Chapter, and Laboratory directives.

The lectures conclude with a listing of standards, guides, handbooks, and data sources relevant to criticality safety.

Laboratory work starts with control room and laboratory familiarization and an introduction to safety practices. Handstacking then begins on a simple assembly consisting of alternating $\frac{1}{8}$ -in. Plexiglas plates and 3-mil highly enriched uranium foil. This assembly is built up to a multiplication of about ten in accordance with an approved experimental plan, with additional reactivity additions being made under remote operation. This approach to critical is terminated at a multiplication of about 100. Laboratory work concludes with operation of Jezebel, an unreflected plutonium critical assembly. Positive periods of about 20 s are demonstrated.

The unique opportunity to participate in this laboratory work gives students a valuable perspective. Reactions of the approximately 200 people who have participated in this program have generally been very favorable.

Classes have been offered about every 6 to 8 weeks and it is planned to continue these classes as necessary to meet future demands.

Criticality Safety Training of Operators in a Large ERDA Production Facility, *W. T. Mee (UCC-OR), invited*

Nuclear criticality safety training programs contribute to the safety and efficiency of large production facilities processing fissile materials, especially if operating personnel are trained on the HOWs, WHATs, and WHYs to perform their productive tasks. It is firmly believed that the informed operator will perform more satisfactorily and conscientiously if told the reasons for controls and limits in handling enriched uranium.

Criticality safety training programs are a vital part of the policy of Union Carbide Corporation—Nuclear Division (UCC-ND) production facilities operated for ERDA under contract. The criticality safety training program in UCC-ND Oak Ridge production facilities has utilized a number of audio/visual aids to supplement lectures on the physics and practical explanations of nuclear criticality safety. These lectures have been used to train as many as 4000 operations, maintenance, service, and technical personnel in a year. Usually, half of this number receive training annually.

Years ago, the operator was given only the criticality limit and control for his specific job, with no explanation. It became apparent that such limited information was an unnatural barrier to the education of the inquisitive operator. Therefore, a formalized training program was initiated in the mid-1950's to explain the reasons for these limits. The need was further supported by the fact that operations personnel have direct access to fissile materials, requiring some dependence on administrative controls.

Training programs were developed for the operators and other crafts. The term "operator" includes the chemical, inspection, assembly, and machining personnel who have direct contact with, and control of, the fissile material in their respective jobs. The "other crafts"—the janitor, cleaner, maintenance man, and utility servicemen—also have opportunities to be in contact with fissile material.

The practices within UCC-ND production facilities are to explain to both groups the DOs and DON'Ts of their respective jobs. In the case of the operators—those directly involved in processing enriched uranium—criticality safety limits and controls are fully explained.

Training programs for other crafts—those not directly processing enriched uranium—are designed to define the primary DON'Ts they are expected to follow. Such crafts as maintenance and utility personnel are informed of how they must coordinate their work tasks with the personnel who are responsible for handling the enriched uranium.

Audio/visual aids have been used generously for the operator and other crafts in training. The ideas for these and other aids were developed within the plants, and, as a theme, were directed toward general information on the reason for criticality safety. In each aid, both descriptive and illustrative explanations were used to inform personnel of the neutronics of fissile material. The first of these was a pocket brochure entitled "Principles of Nuclear Safety," printed originally in April 1959. The brochure describes the difference between radioactive decay and the fission process, and how they may be controlled for nuclear criticality safety. The brochure is used in lectures explaining it and giving details of the audience's responsibility in the plant.

The second aid was an exhibit employing liquid and wood models of critical configurations of enriched uranium in solution and solid metal form. The illustrations consisted of critical mass and geometry configurations determined by critical experiments. Use of the exhibit allowed personnel to see actual critical sizes of the materials they were accustomed to handling in smaller quantities on a daily basis.

The third aid was a film entitled "Safe Handling of Enriched Uranium." The film was made using scenes of enriched uranium processing in the Oak Ridge Y-12 Plant, and animated descriptions of the parameters affecting criticality. Again, it was an opportunity to inform personnel of the conditions the nuclear criticality safety engineer must evaluate to establish safe controls. Also, it showed these controls in practice.

Another approach that has been used consisted of describing actual criticality safety violations with respect to how they happened, the degree of criticality safety infraction, and what additional actions would have been necessary for criticality to exist. Still another approach involved testing of personnel after descriptive lectures, with true/false and multiple-choice questions followed by answering sessions to ensure all personnel understood the correct answer.

As in any effective program, management support is mandatory. UCC-ND management has always supported the criticality safety program. This support has encouraged the development of the described visual aids. Time is allotted for personnel to participate in these training sessions on a routine basis. These sessions offer sufficient time to discuss pertinent criticality safety rules and controls with the participants.

UCC-ND procedures specify that an employee involved in processing enriched uranium will receive nuclear

safety training at least annually, or whenever transferred from one major job to another within the uranium processing areas. Training is documented, using a form completed by the operator and supervisor and signed by each. The completed record is retained in the personnel files.

Criticality safety training programs continue to be a vital contributing factor in the safe operation of large fissile material production facilities. Past programs have been productive, and the dividends received are best illustrated by the criticality safety records. On-the-job observations made by criticality safety specialists indicate operations personnel appreciate efforts made to explain the WHYs of criticality safety. This appreciation shows up in their work habits and attitudes.

Nuclear Criticality Safety Training in the Private Sector, *F. M. Alcorn (B&W), invited*

Training in nuclear criticality safety is and always has been a part of the private sector. The purpose of this paper is to summarize a survey of training in the private sector made recently by the author and to reflect the author's experience on training. The private sector is defined for purposes of this paper to be nuclear fuel processors licensed by the NRC as opposed to ERDA contractors and national labs; power reactor facilities and universities are not included. The author surveyed eight installations in the private sector; specific reference will not be made to any particular installation. Most of the installations surveyed are facilities concerned only with low-enriched uranium fuel (< 5 wt%); although, high enrichment and plutonium facilities were contacted and are represented in the reported findings.

It should be noted that there are no private sector installations that regard their training in nuclear criticality safety to be primary (even secondary) control to maintaining criticality safety. The most important factor to maintaining safety is through good design which maintains safety even in the face of any credible accident or operator error. Secondary to design are operating procedures and rules that enhance safety and add to the built-in safety margins. All facilities recognized that the various types of training, health physics, industrial safety, accountability and safeguards, complemented each other and were all important.

Two private sector installations have a training program for their nuclear criticality safety engineer(s) or specialist(s). Most installations must look to either hiring an engineer experienced in nuclear criticality safety evaluations or to tutoring by an in-place specialist together with on-the-job training. At some installations specific nuclear criticality expertise is supplemented by reactor physics personnel or by outside consultants. Except for University of New Mexico short courses, there appears to be no good formal academic training available to nuclear criticality safety specialists and potential specialists.

About half of the installations have training for plant managers. Where management training was given, it usually was in the form of management review of the safety program. Managers are free to attend operator training sessions and quite often do. There appeared to be no lack of management support where there was no official management training.

All installations have nuclear criticality safety training for operators and their foremen. All but one has special classes for new operators; all give periodic training to their operators and foremen. Four of the eight

installations give retraining classes on a biennial basis; three provide annual retraining; and one retrained twice a year. Planning and execution of training and retraining appear uniformly to be the responsibility of the nuclear criticality safety function of the installation. One facility utilized the help of a professional training department to prepare its presentations. Typical classes for new employees and for retraining run from 20 min to a little over an hour. Training for nuclear criticality safety is normally integrated with general safety training; however, at most installations the nuclear criticality safety training is a unique program executed separately from other safety training. Training for plant engineers who must design and lay out the equipment generally follows that given to operators and their foremen; the level is usually the same.

Typical programs deal with criticality and the parameters related to criticality, the effects of a criticality incident, the importance of following procedures and rules on safety, and response required by a criticality alarm. None of the installations uses models; all use the Y-12 film "Safe Handling of Enriched Uranium," all make use of overhead projectors or slides as visual aids, and four have developed an in-house brochure. A retraining program nearly always emphasizes immediate past performance and practice in the plant and new procedures and rules. Three installations identified informal monthly meetings held by the various plant groups, which were led by the foremen and dealt with a number of safety and production issues. Criticality safety was discussed several times a year and the nuclear criticality specialist was usually invited to those meetings dealing with criticality safety.

Training programs are documented in only half the installations. Apparently the requirement for nuclear criticality safety is found in all but one of the eight SNM licenses. In all installations attendance by the operators and their foremen is mandatory and attendance is taken. Compliance inspectors usually monitor the facility to ensure it does carry through with its training program; however, the level of interest in training programs by compliance inspectors varies from region to region.

How effective are the various nuclear criticality safety training programs? Three installations have formalized an attempt to measure the effectiveness of their program (via tests and oral questioning of operators on the floor). Most felt that the effectiveness could be "felt" by performance of plant personnel and lack of NRC citations. The nuclear industry's safety record should speak for itself; there has been only one criticality incident in the private sector, and that occurred 13 years ago.

University Activities in Criticality Safety Education, *Ronald A. Knief (U of NM), invited*

Universities have two potential roles in nuclear criticality safety education: (a) continuing education programs for practicing professionals, and (b) regular on-campus and special off-campus course coverage for degree students. Both roles are addressed.

In May 1973, the University of New Mexico (with the help of the ANS Trinity Section) sponsored the first Nuclear Criticality Safety Short Course to meet continuing education needs. Nearly 100 participants—expert to novice professionals in the field—shared the educational experiences of the one-week program. This short course, as well as the programs that followed, was held at the University's D. H. Lawrence Ranch near Taos, New Mexico. The informal atmosphere and remoteness of

the facility have been found to be extremely conducive to interaction between the faculty and participants.

Additional short course offerings from 1975 to 1977 were directed completely toward those with two or fewer years of experience in the nuclear criticality safety field. Faculties for the programs were selected and drawn from industry, national laboratories, government, and universities. The short courses' educational objectives were carried out through formal lectures, work/discussion sessions, and informal interactions between faculty and participants.

Two more-advanced programs have also been sponsored by UNM. In 1976 the Advanced Seminar in Nuclear Criticality Safety brought together thirty of the field's acknowledged experts from the U.S., Canada, Great Britain, France, and Japan. In May 1977 the Advanced Short Course in Nuclear Criticality Safety provided practical instruction and work sessions for those with intermediate tenure in the field.

These continuing education programs have been attended by over 200 individuals from around the world; approximately 20 of these people have served in faculty capacities through the support of their respective organizations.

Historically, criticality safety courses have appeared in regular university curricula only where an individual faculty member had an active association with the field. Consideration through special seminars and colloquia was somewhat more common.

Prospects for substantial emphasis on criticality safety, through the efforts of the Nuclear Fuel Cycle Education Committee (NFCEC),¹ now exist. The Committee, which was originally organized by Prof. Dean Eckhoff of Kansas State University and consists of university and industry representatives, has as its goal the development of high-quality teaching modules. The author and Prof. Bob Seale of the University of Arizona have responsibility for the module devoted specifically to nuclear criticality safety. These modules are to be made available to the universities and industries as soon as they are completed.

The major off-campus programs for university students have been conducted in parallel with the Nuclear Criticality Safety Short Courses. During planning for the 1975 course it was recognized that graduate students could benefit greatly from exposure to the program. Thus, a pilot group of four students from UNM and two from Arizona attended the short course and subsequently participated in a follow-on week of experiments at the Critical Facilities of the Los Alamos Scientific Laboratories.

Success of the pilot program encouraged expansion to a national graduate-student constituency. Modest grants from ERDA to cover travel and housing costs plus a willingness by UNM to waive the short-course registration fee made possible the programs in 1976 (Ref. 2) and 1977. In all, 19 students from 11 different universities have been active participants in the two-week programs of short course, computer application, and experiment activities.

Though relatively few students will eventually be employed in the criticality safety field, those exposed to this program will have the valuable experience of seeing nearly all of the engineering and administrative aspects of one important part of the nuclear fuel cycle. This is sure to provide useful insight into any other area of nuclear engineering in which they may choose to work.

University activities in nuclear criticality safety are expanding. The University of New Mexico is planning to continue and extend the short course, seminar, and graduate student programs. The work of the NFCEC provides a strong likelihood that criticality safety will take a more prominent place in nuclear fuel cycle courses. The continuing strong support of nuclear criticality safety professionals as well as that of their management will assure the success of these educational activities.

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A Graduate Student's View of Nuclear Criticality Safety Education, *Mark. D. Hoover (U of NM), invited*

Graduate students have become a familiar part of the nuclear criticality safety workshops which are organized and conducted by the Department of Chemical and Nuclear Engineering of the University of New Mexico. After the first workshop in May 1973, it was recognized that any future courses could provide a unique opportunity for graduate student education. Graduate student participants have been included in courses ever since. I had the good fortune to attend the second workshop in May 1975, as part of the first group of graduate student participants. There were 6 graduate students at that course, 4 from UNM and 2 from the University of Arizona. The scenic and remote setting of UNM's D. H. Lawrence Ranch in Taos, New Mexico, provided an ideal atmosphere for both work and relaxation. We spent the week participating in discussions and problem sessions and attending formal lectures. We were able to interact on both a technical and a personal level with criticality safety professionals from industry, national laboratories, government, and other universities. After the week in Taos, we spent an additional week at the Critical Facility of the Los Alamos Scientific Laboratory participating in mini-tours and experiments. These two weeks provided a massive exposure to criticality safety. My wife claims I even dreamed criticality safety after returning home. She alleges I sat up in the middle of the night, threw my pillow out of bed, and shouted, "It went critical!"

The graduate student program was considered a success by the students, the short-course faculty, the participants, the LASL staff members who conducted the second week activities, and members of the ANS Criticality Safety Division. The program was expanded in 1976 to attract graduate students from all over the nation. A modest grant from ERDA to cover travel and housing costs plus a willingness on the part of UNM to waive other costs associated with the short course made this expanded program possible. A total of 10 students from 8 different schools from across the nation attended in 1976. The program again included both the week at Taos and the week at LASL. I participated that year as a junior member of the short-course faculty conducting work sessions on hand-calculation techniques. The 1977 short course had a total of 9 students from 6 different schools and included work at UNM on computer techniques during the second week as well as the experiments and tours at LASL. In addition to the basic short course, an advanced criticality safety course was offered in May 1977. I was able to attend this as a graduate student participant. It

was a very worthwhile experience for me and a definite pleasure to see familiar faces from the two previous years.

Perhaps only a few of the graduate students who have participated in the short courses will actually go on to become criticality safety specialists. It's safe to say, however, that at some time in our professional careers each one of us will feel the impact of criticality safety considerations or see the application of some basic skill or attitude associated with criticality safety. Some of the insights we have gained are a historical appreciation of the development of a mature engineering discipline; an understanding of the fundamental physical theories which are the basis of criticality safety calculations; an awareness of pitfalls to avoid based on the advice of experienced veterans; the development of good engineering judgment and experimental techniques; the cultivation of effective written and oral communication; and an appreciation of the integral part that criticality safety plays in the administration, management, and operation of the nuclear industry.

Some of the other gains to graduate students are in the form of unique personal experiences. It is probably not possible to put a value on being able to say, "I had breakfast with Hugh Paxton and Dave Smith."

Evaluations completed by the graduate students indicate we are all grateful to the faculty and participants of the short courses, to the staff of the Los Alamos Criticality Facility, and especially to the director of the courses, Dr. Ronald Knief of UNM, for having made us a part of the program. U.S. ERDA support is also gratefully acknowledged and it is hoped the graduate student program will continue to be a viable part of the criticality safety short courses.

The Application of Fault Tree Analysis to Nuclear Criticality Safety, *John W. Morfitt (EG&G Idaho), invited*

Fault tree analysis was successfully developed and utilized by the aerospace industries during the 1960's. During the early 1970's, W. G. Johnson (former General Manager of the National Safety Council) developed a more specialized version of the fault tree known as MORT (Management Oversight and Risk Tree) as an evaluative tool for safety professionals and their management. Its effectiveness has been verified in actual field trials. Its use is now mandatory within ERDA for the more severe classes of accidents.

MORT can be equally useful and even more valuable, however, as an analytical tool to reduce or eliminate potential sources of accidents. Such uses have been frequent, but not widespread; moreover, their application to criticality safety problems, with certain exceptions, has been neglected. Yet criticality safety requires the kind of thorough, highly disciplined, and carefully documented analysis that can be greatly strengthened by the use of MORT techniques. In addition, the author believes that training criticality safety professionals in the use of MORT early in their careers would not only speed up their ability to use their formal training effectively, but would induce, at an early stage in their career, those mental attitudes and disciplines prerequisite to effective performance.

Among other concepts, MORT is based on the following:

1. Safety is simply one special aspect of the reliable control of work.

2. Accidents are one member of the broad family of errors and malfunctions; therefore (a) much of the literature on error control can be adapted for safety improvement, (b) it is more important to analyze the "near miss" than it is to analyze actual accidents.

3. The task of the investigator is to find out not who is wrong, but what is wrong. Virtually all accidents have not one, but several antecedent causes.

4. Training personnel in the use of a fault tree is one of the best ways to guarantee their analyses will be thorough, competent, and objective. Although a fault tree is merely "structured common sense," it [like the genealogy chart (another type of tree)] brings to light relationships that might otherwise be overlooked.

5. The typical manager has no clear idea of the risks he/she is actually assuming in conduct of the operations. An important feature of a MORT fault tree analysis is that it surfaces these risks, forcing a series of choices on management: (a) to eliminate certain risks (or reduce their probability); or (b) to accept the remaining risks as known and permanent characteristics of the operation.

6. Any operation resulting in an accident not previously identified with a specified operational risk implies, therefore, not only a human and/or equipment failure at the operating level, but a concomitant failure at the management level.

7. It is possible to construct a fault tree of such generality, breadth, and controlled redundancy that it can be used effectively—with minor adjustments—in every investigation of actual or potential accidents. The new professional does not need to be trained how to construct fault trees, only how to use one particular, readily available version that has been constructed, reshaped, and checked by literally hundreds of people. Even the least experienced trainee immediately inherits this compact, pervasive technology.

Training in MORT, like training in any other worthwhile discipline, is not a trivial exercise. The time required to bring a potential user of MORT to a minimum level of competence is approximately 20 to 30 hours of class work plus a roughly equal amount of homework assignments. However, based on a single 3½-hour introduction to MORT that I gave at the University of New Mexico under Dr. Ronald Knief's direction (Advanced Course on Nuclear Criticality Safety), I believe the students received sufficient insight into the process to encourage at least some of them to try self-study or to consider additional formal instruction. As a single experiment, the results of which are not yet known, the superficial results were encouraging enough to merit its repetition.

Criticality Safety Assessment, R. C. Lloyd, E. D. Clayton, R. D. Carter (BNWL)

An analysis of data on criticality safety violations, in terms of criticality safety philosophy and the human and mechanical factors involved, will permit judgments that may help reduce the number of occurrences of future violations. These data may further be utilized in a fault tree analysis where cause factors can be assigned frequency values. When the higher frequency causes of violations are known, corrective action can be taken to alleviate them in the most efficient way.

A study of 100 criticality safety specification (CSS) violations reported during 1967 to 1976 was made. These violations were classified by a severity index; the method

of disclosure and cause of violation were further delineated.

The divisions of the severity category were chosen with zero as no possibility of criticality and five as critical, with one through four denoting conditions between these extremes.

About half the violations were detected and reported by plant personnel and half by auditors. Figure 1 shows a graph of the percent of violations in each severity index by detection group, plotted against severity index. The graph clearly shows that plant personnel reported a lower percentage of low severity index violations and a much higher percentage of the higher severity index than did the auditors.

Although the trend has been toward more frequent audits by larger auditing groups, these curves indicate little benefit is to be expected by further emphasis in this area. Rather, if further improvements are to be made, the emphasis might well be placed on the training and encouragement of plant operation personnel (who appear to be doing a credible job) to report violations.

The causes behind the violations in order of frequency are (a) CSS clarity, (b) lack of care, (c) insufficient training, and (d) mechanical breakdown. Mechanical breakdown (being only 2%) indicates that equipment, where failure would cause a violation, is being adequately maintained and is the cause of few violations. The 2% value can be considered normal, as it is about the same as equipment failure assigned in other areas such as auto accidents or general plant injuries.

Since CSS clarity, lack of care, and insufficient training were essentially equal contributors to the cause of violations, efforts to improve these areas should be shared equally. From analyses of violation reports and interviews of individuals involved with the CSS, the following suggestions for improvements are considered.

CSS clarity is very important for reduction of violations. The CSS representative and the operations engineer should coordinate their work when developing CSS to ensure an understandable CSS for the individuals doing

Fig. 1. Comparison of severity index by detection group.

the actual physical work. The limits should be written in the shortest and clearest way. Once written, the CSS should not be changed unless there are changes in the process, etc. The practice of periodically revising the CSS, involving unnecessary and insignificant changes, trips up the operation technicians. This does not mean that review as to applicability, on a periodic basis, would not be worthwhile. Some of the violations occurred because the facility involved was in violation of a CSS as issued or revised; i.e., the normal operation was not covered properly by the CSS. This is an indication of lack of interchange at the operations level.

The violations caused by lack of care appear to be the result of rushing, too many things involved, or in

improper attention to details. An obvious area of difficulty concerns the communications exchange at shift change, an improvement in this area would be worthwhile. The second area involves adequate planning so there is no need to rush to complete the job before shift end.

Insufficient training is normally involved for new employees and employees who have recently changed jobs. Discussion of CSS should be a regular part of the general safety training program. All new CSS and revised CSS should be read and understood by those involved in the facility work; this especially applies to employees who are new to a work function.

These data will be incorporated with fault tree methodology to further study criticality safety.

Data and Analysis for Nuclear Criticality Safety Guidance

Validation of Criticality Safety Broad-Group Library Using Uranium Systems, *N. F. Cross, R. M. Westfall, K. R. Turnbull, P. B. Fox (ORNL)*

The performance of a broad-group neutron cross-section library being developed¹⁻³ as a standard for criticality safety calculations has been evaluated using several critical uranium systems. The primary effort was directed toward the evaluation of clean and borated water-moderated, low-enriched lattices of uranium metal⁴ or uranium oxide^{5,6} rods. A set of uranium solution experiments^{7,8} over a wide range of H to ²³⁵U ratios was also evaluated.

The 19-group structure that was used for this study is given in Table I. The group boundaries of the first 15 groups correspond to those of the 16-group Hansen-Roach⁹ structure with an additional 4 thermal groups for treating the transition region between the resonance and thermal energy ranges.

Transport analyses of the critical systems were performed using XSDRNP,¹⁰ KENO IV,¹¹ KENO V,¹² and PX-KENO¹³ codes. PX-KENO is a point cross-section version of KENO which provides benchmark quality results using ENDF-IV data. Five different cross-section sets were used in the analyses presented in Table II. For the 218-group and broad-group ENDF/B-IV libraries and the 123-group GAM-THERMOS¹⁴ library, problem-dependent cross-section processing was performed using the Nordheim Resonance treatment of the NITAWL¹⁰ module in the AMPX¹⁰ system. The 123-group library is based primarily on GAM II and THERMOS data.¹⁴ The 16-group Hansen-Roach library as used here has been modified by applying a Bondarenko¹⁵ treatment in the resonance energy groups. This Bondarenko procedure allows for the effects of lattice heterogeneity in calculating resonance self-shielding.

The conclusions drawn from this study include:

1. For the metal rod lattices, both the 19-group ENDF/B-IV results and the Bondarenko-corrected 16-group Hansen-Roach results agree well with the experiments.
2. The presence of boron does not affect the accuracy of the calculations.
3. For low water-to-fuel volume ratios, the ENDF/B-IV data underestimate k_{eff} by about 1 to 1½% as compared to experiment, while the 123-group cross-section set agrees with the experiments. This discrepancy is in agreement with the findings of other investigators.¹⁶
4. From the viewpoint of the broad-group library validation for uranium systems, consistent agreement was found between the 218-group ENDF/B-IV, the 19-group ENDF/B-IV, and the point cross-section ENDF/B-IV data, indicating that the discrepancy between calculation and experiment in the low water-to-fuel volume ratio range is due to deficiencies in the ENDF/B-IV data and not in the method used to process the cross-section libraries.
5. The broad-group ENDF/B-IV, the 218-group ENDF/B-IV, the ENDF/B-IV point data, and the 16-group

TABLE I
Energy Bounds of 19-Group Structure

Group No.	Energy Boundaries (eV)
1	2E+7 to 3E+6
2	3E+6 to 1.4E+6
3	1.4E+6 to 9E+5
4	9E+5 to 4E+5
5	4E+5 to 1E+5
6	1E+5 to 1.7E+4
7	1.7E+4 to 3E+3
8	3E+3 to 550
9	550 to 100
10	100 to 30
11	30 to 10
12	10 to 3.05
13 ^a	3.05 to 1
14 ^a	1 to 0.4
15 ^a	0.4 to 0.1
16 ^a	0.1 to 0.05 ^b
17 ^a	0.05 to 0.03 ^b
18 ^a	0.03 to 0.01 ^b
19 ^a	0.01 to 1E-5 ^b

^aThermal groups

^bGroups below Hansen-Roach structure

Hansen-Roach cross sections calculate within 1% of critical for the uranium solution spheres, with the broad-group results being 0.1 to 0.3% higher than the fine-group results.

In conclusion, this study demonstrates that the 19-energy-group structure yields results consistent with the ENDF/B-IV many-group and point cross-section data. Preliminary studies indicate that the final structure of the standard broad-group cross-section set may need to be expanded to adequately treat plutonium systems.

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TABLE II

Calculated Results for Critical Uranium Metal and Uranium Oxide Lattices with Clean and Borated Water Moderators

Critical Experiment	Water/Fuel Volume Ratio	Pitch (in.)	Number of Rods	Calculation Method	Number of Groups	Calculated Results
WCAP ⁵ A water-moderated 23 × 23 array of 2.72% enriched UO ₂ rods	1.49	0.6	529	KENO V	218	0.9848 ± 0.0068
				KENO V	123	0.9974 ± 0.0088
				KENO V	19	0.9867 ± 0.0044
				KENO V	16	0.9957 ± 0.0046
				PX KENO	POINT	0.9869 ± 0.0063
EPRI ⁶ Clean-water-moderated lattice of 2.35% enriched UO ₂ rods	1.20	0.615	708	KENO V	218	0.9864 ± 0.0042
				KENO V	123	0.9977 ± 0.0037
				KENO V	19	0.9849 ± 0.0040
				KENO V	16	1.0105 ± 0.0039
				PX KENO	POINT	0.9900 ± 0.0060
	2.41	0.750	383	KENO V	218	0.9922 ± 0.0050
				KENO V	123	0.9989 ± 0.0041
				KENO V	19	0.9934 ± 0.0039
				KENO V	16	1.0055 ± 0.0044
	3.68	0.87	342	KENO V	218	0.9922 ± 0.0050
				KENO V	123	1.0052 ± 0.0045
				KENO V	19	0.9934 ± 0.0039
				PX KENO	POINT	0.9984 ± 0.0061
EPRI ⁶ Borated-water-moderated lattice of 2.35% enriched UO ₂ rods		0.615	1201	KENO V	19	0.9837 ± 0.0035
		0.75	1201	KENO V	19	0.9983 ± 0.0036
		0.87	885	KENO V	19	1.0007 ± 0.0034
4.98% enriched metal rod, water-moderated lattices ⁴ in a basic 21 × 21 array from which rods were removed	3.09	0.52	400	KENO IV	19	0.9878 ± 0.0050
				KENO IV	16	0.9929 ± 0.0056
	3.48	0.52	400	KENO IV	19	0.9952 ± 0.0050
				KENO IV	16	0.9909 ± 0.0049
	3.58	0.52	324	KENO IV	19	0.9965 ± 0.0050
				KENO IV	16	0.9930 ± 0.0056
	6.66	0.52	256	KENO IV	19	0.9967 ± 0.0050
				KENO IV	16	0.9916 ± 0.0047
	9.40	0.81	256	KENO IV	19	0.9913 ± 0.0053
				KENO IV	16	1.0082 ± 0.0048
	10.15	0.81	400	KENO IV	19	0.9908 ± 0.0044
	10.38	0.81	324	KENO IV	16	1.0077 ± 0.0045
				KENO IV	19	0.9962 ± 0.0051
				KENO IV	16	1.0000 ± 0.0050

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Application of Raschig Ring Standard to Low-Enriched Uranium Fuels, Norman Ketzlach (NRC)

The primary objective of the results of calculations reported here was to extend the bases for the Raschig Ring Standard¹ to permit inclusion of realistic rules for aqueous low-enriched uranium ($\leq 5\%$ ^{235}U) systems. The objective was attained by calculations, using the KENO Code with validated cross-section sets, made by F. G. Welfare, Babcock & Wilcox Company, and by the writer. Welfare utilized the Knight modified 16-group Hansen-Roach cross sections; the writer, the 123-group GAM-THERMOS cross-section library.

The variables investigated included uranium concentration, B_2O_3 content of the glass, thickness and density of UO_2 coating on the rings, volume fraction of the vessel occupied by the rings, ring dimensions, and ^{235}U enrichment.

Factors of conservatism in the calculations included the geometric model and the use of uranium-metal-water mixtures as a stand-in for the generally less-reactive chemical solutions or oxide-water mixtures.

The model used for nearly all the calculations was an infinite triangular lattice of closely packed borosilicate glass tubes (1.50-in. o.d.). The inside diameter of the tubes was adjusted to 1.286 in. to provide assemblies having a tube volume equal to 24% of the total volume. The remaining volume, both inside and outside the tubes, was filled with the specified uranium metal-water mixtures. The model used was conservative² compared to the actual conditions of Raschig ring use, which involve random loading of vessels. The thickness of the borosilicate glass tubes was the minimum possible since greater spacing between tubes would require thicker tubes for the same volume fraction of glass.

The reactivity calculations covered a range of uranium concentrations from 1.0 to 18.8 g/cm³ in the matrix solution. Welfare's calculations were made over the range of 1.0 to 5.4 g U/cm³ with the rings (containing 12.8% B_2O_3) occupying 24% of the total volume. The writer extended the calculations from 5.4 g/cm³ (equivalent to 270 g ^{235}U /litre permitted by the Standard when the volume fraction of the borosilicate-glass rings is 24% of the total). Good agreement ($\sim 0.1\%$) was obtained between the comparison set of calculations made at 5.4 g U/cm³. The results of the calculations are summarized in Fig. 1. The reactivities of the unpoisoned mixtures are also plotted. It is seen that the poisoning effect of the glass is sufficient to prevent criticality for all uranium concentrations.

Reducing the B_2O_3 content in the glass to 11.8% resulted in an increase of $\sim 3.3\%$ in the calculated reactivity compared to the value for 12.8% B_2O_3 .

Reactivity calculations were made for systems with 5.4 g U/cm³ in the matrix "solution" in which an oxide coating was formed both on the inside and outside walls of the rings. The UO_2 density of the coating was varied from 5.4 to 10.9 g UO_2 /cm³ and the content of the coating varied from 50 g contained ^{235}U /litre glass (upper limit in the standard) to three times that quantity. The maximum increase in reactivity was found to be $\sim 3\%$.

A reactivity calculation made for close-packed systems in which the volume of glass was increased to 32% and the uranium concentration increased to 8.0 g U/cm³ (equivalent to 400 g ^{235}U /litre) resulted in a reactivity increase of 0.8% over that of a system containing 24% glass by volume and a uranium concentration of 5.4 g/cm³.

Fig. 1. k_{∞} vs uranium concentration.

Increasing the ring o.d. to 1.60 in. from 1.50 in. (for the same ring thickness and occupied ring volume fraction) resulted in very small differences ($\sim 0.5\%$) in reactivity.

Several calculations were made for systems with 4% ^{235}U enriched uranium with the expected decrease in reactivity.

Available data^{3,4} support the use of the rings in caustic solutions limited to specified maximum conditions of concentration and temperature (e.g., 3N caustic and 30°C temperature).

From the above it can be concluded that (a) it is safe to extend the application of Table I in the present Standard¹ to $< 5\%$ ^{235}U enriched uranium-water systems at contained ^{235}U concentrations equal to the tabulated uranium concentrations (for highly enriched solutions), and that (b) the Raschig rings may have limited application in caustic solutions.

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Safety Calculations and Benchmarking of PWR Fuel Storage Pool Studies, William D. Bromley (B&W)

The Babcock & Wilcox Company has completed a study to ascertain how close spent PWR fuel assemblies can be

placed in a closed packed storage pool. An overview encompassing the design calculations and the final design was presented at the Summer ANS Meeting.¹ The purpose of this paper is to detail the methodology of the nuclear safety calculations and more importantly to discuss the benchmarking work that was necessary to validate the study.

Nuclear safety of the storage pool was assured through a series of parametric studies aimed at establishing the final design mode. All calculations assumed fresh fuel of either 3.5 wt% uranium oxide or 4.5 wt% mixed-oxide fuel in the B&W Mark 'C' canless assembly. Cases assuming uranium oxide fuel were calculated with KENO^{2,3} employing the 16-group Hansen-Roach cross-section set,⁴ while the mixed cases were run with 123 groups.⁵

The final design decided on is a fuel storage rack based on 13-in. center-to-center spacing. Throughout the storage pool each array location will contain a 0.134-in.-thick unborated stainless-steel can placed around the whole length of the assembly while two sides of the can will contain a 0.134-in.-thick stainless-steel doubler plate. A 13-in. center-to-center spacing gives B&W a 260% increase in PWR fuel pool storage capacity over the present 21-in. spacing. This translates to approximately 6 years, on the average, of future operational ability to the utility. Further savings can be achieved for lower initial enrichments. Our study has shown that a 12.5-in. spacing is adequate for lower enrichments.

Critical experiments were used to benchmark both the uranium oxide and mixed-oxide calculations. Several sets of criticals were looked at before choosing experiments that most closely matched the assembly's pin pitch and fuel types. The uranium oxide fuel system was benchmarked using unpublished data, received by personal communication, from five critical experiments done at the Oak Ridge National Laboratory and one B&W critical experiment.⁶ The variables investigated in modeling the uranium oxide experiments included:

1. number of fuel rods
2. use of depleted uranium metal block reflector
3. use of stainless-steel plate placed through the fuel pins
4. use of "Boral" plate (boron impregnated aluminum) placed through the fuel pins.

The mixed-oxide cases were benchmarked against the Saxton⁷ and Westinghouse Reactor Evaluation Center (WREC)⁸ critical experiments. The Saxton and WREC critical experiments considered the effects of varying the following factors:

1. pin pitch
2. number of fuel rods
3. array configuration
4. ²⁴⁰Pu content of the fuel rods.

Results of the benchmarking have shown that the code KENO with the 16-group Knight-modified Hansen-Roach cross-section set is adequate when calculating PWR fuel assemblies with 3 to 4 wt% uranium oxide fuel. For these cases K_{eff} is consistently overpredicted by an average of 2.6%. We have found, however, that the 16-group cross-section set should not be used with mixed-oxide fuel. When the Hansen-Roach cross-section set was used to calculate the mixed-oxide criticals, K_{eff} was overcalculated by 6 to 8%. With this large an overcalculation, the ability of 16 groups to calculate even a relative reactivity change in a mixed-oxide system is very much

in doubt. Our studies indicate that the 123-group cross-section set, available through the AMPX code package, is more appropriate with K_{eff} calculated to within 0.25% on the average.

The final K_{eff} calculated for a storage pool added a bias to the mixed-oxide cases. However, no credit or decrease in K_{eff} was taken for overpredicting the neutron multiplication in uranium oxide for two reasons:

1. Not enough criticals were benchmarked to obtain a statistically accurate magnitude for the bias; only enough to be confident that it was adequately conservative.
2. The critical experiments available to date do not exactly describe the geometry encountered in a fuel storage pool.

Final results indicate that with a lattice spacing of 13 in. and a 0.134-in.-thick stainless-steel can, all nuclear safety, thermal-hydraulic, and seismic design requirements are met. This basic design is a conservative approach to the fuel storage problem which utilizes current technology, and offers an optimum balance of the design variables at the minimum cost with an adequate safety margin. For some plants where variables such as fuel enrichment are different, closer spaced racks using non-poisoned stainless-steel cans can be utilized. Results of the benchmarking work show that the KENO code with the 16-group Hansen-Roach cross-section set is adequate when calculating K_{eff} of a PWR fuel assembly containing uranium oxide fuel but not adequate for mixed-oxide fuel. Using a 123-group cross-section set with KENO will adequately predict the K_{eff} of a mixed-oxide array.

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A Graphic Method for Comparing k_{∞} in Fuel Storage Pools, Edward Lantz (NRC)

Delays in availability of spent fuel reprocessing facilities have made it necessary for many licensees to file

applications with the U.S. Nuclear Regulatory Commission (NRC) for license amendments to authorize increases in storage capacity, which are generally obtained by reducing the spacing between storage locations; i.e., by going to higher density racks.

Using results of criticality calculations, which were submitted to the NRC by the licensees, a graphic method has been developed which appears to provide neutron multiplication factors, k_{∞} , in spent fuel pools within accuracy limits of about one-half of one percent.

Most of the analytical results filed by the licensees were for stainless-steel racks, which are to be used for storage of pressurized water reactor (PWR) fuel. The graphic method for these fuel assemblies is the subject of this paper.

The uranium dioxide fuel pellets in PWR fuel assemblies that were considered all had diameters between 0.82 and 0.97 cm. The fuel pins were all Zircaloy clad, and the metal-to-water ratios in the fuel assemblies were all about the same. Factors that varied significantly were: (a) amount of ^{235}U in the assemblies; (b) overall size of the assemblies; (c) thickness of the Type 304 stainless steel in the racks; and (d) thickness of water between the fuel assemblies in the storage lattice.

The change in k_{∞} in these lattices is essentially determined by the change in the thermal utilization, f , which to first order is proportional to the fuel region volume fraction, V_f ; i.e.,

$$k_{\infty} \propto f \propto \frac{\text{Fuel Absorptions}}{\text{Total Absorptions}} \propto V_f$$

Also, by using the fuel region volume fraction, both the size of the assemblies and the thickness of water between the assemblies is accounted for with one variable. The variation in the amount of ^{235}U is taken out by using a reactivity coefficient. We are then left with two variables: the thickness of the stainless steel and the fuel region volume fraction, V_f .

By considering storage racks which all had the same thickness of stainless steel and plotting the infinite neutron multiplication factor, k_{∞} , as a function of V_f , we found that, when the neutron energy spectrum in the

Fig. 1. Infinite neutron multiplication factor, k_{∞} , for 0.635-cm Type 304 stainless-steel storage rack for PWR fuel assemblies with ^{235}U .

$$\text{Density} = 0.088 \frac{\text{g } ^{235}\text{U}}{\text{cm}^3 \text{ of fuel region}}$$

Fig. 2. Infinite neutron multiplication factor, k_{∞} , for Type 304 stainless-steel (SS) storage racks for PWR fuel assemblies with ^{235}U .

$$\text{Density} = 0.088 \frac{\text{g } ^{235}\text{U}}{\text{cm}^3 \text{ of fuel region}}$$

water region where the stainless steel was located was calculated as well as it could be, k_{∞} was almost a linear function of the fuel region volume fraction of the storage lattice. The results of one such plot are shown in Fig. 1. This figure depicts the case for 0.635-cm-thick stainless steel. The points shown were obtained by licensees using four different analytical techniques. When uncertainties and inaccuracies are considered, it is seen that k_{∞} is almost a linear function of V_f .

Figure 2 was developed by repeating this procedure for three other thicknesses of stainless steel and putting all of the results on the same graph. With all of the lines on one graph, it was possible to check the relative position of the lines by using calculated reactivity coefficients for the thickness of the stainless steel. It was found that this coefficient was a function of both the thickness of the stainless steel and the fuel region volume fraction, which affects the neutron energy spectrum in the stainless-steel region. This is the reason the lines have different slopes.

Based on a comparison of the results from these figures with the analytical results filed to date by licensees, we conclude that this method will provide an appropriate k_{∞} for Type 304 stainless-steel storage racks for PWR fuel assemblies to an accuracy of about one-half of one percent (i.e., $\Delta k_{\infty} = \pm 0.005$).

Some Effects of Packaging Materials on Critical Arrays of Fissile Materials, J. T. Thomas, J. S. Tang (ORNL)

The Computer Sciences Division at the Oak Ridge National Laboratory has undertaken a program to evaluate the criticality criteria for transport packaging of fissionable materials as expressed in Title 10 of the Code of Federal Regulations Part 71. The study is focused presently on the definition of criticality for water-reflected arrays of loaded containers as a function of container volume, type of insulating material and its thickness, and the number of packages.

The calculational method employed in the study utilizes the KENO Monte Carlo Code¹ with the Hansen-Roach

neutron cross-section sets.² The method has been validated against critical experiments³ performed with the materials of interest and was also used to establish the basis for the ANSI Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, N16.5-1975. The data base represents criticality of reflected arrays of air-spaced units, has an analytic representation,^{4,5} and provides a suitable vehicle for interpretation of the relative effects on criticality of the packaging materials.

Initial calculations are of U(93.2) metal cylinders having a height-to-diameter ratio, h/d , of unity. The calculated critical arrays of air-spaced units are shown in Fig. 1. The introduction of Celotex as the insulating material placed at the cell boundaries results in a reduction of the mass needed to maintain criticality; i.e., reactivity of the arrays would increase were the mass not reduced. The results for three thicknesses of Celotex at a density of 225 kg/m^3 (15 lb/ft^3) are presented in the figure. The calculations were of 216 unit arrays having cubic cell volumes of 56.8, 113.6, and 208.2 litres (15, 30, and 55 gal, respectively).

The introduction of a 3.18-mm-thick inner steel container to these Celotex-moderated arrays results in a negative reactivity effect which about nullifies the positive reactivity caused by the Celotex.

The addition of the outer steel container causes a further decrease in array reactivity as indicated in the figure by the larger masses necessary to maintain criticality. Allowing the number of units to become in-

Fig. 1. Effect of packaging materials on the criticality of a 216 unit array reflected by water. Variation of cell volume and thickness of Celotex insulation is shown.

Fig. 2. Criticality of arrays of 113.6-litre containers with various Celotex insulator thicknesses.

definitely large, $k_{\infty} = 1$, establishes the three asymptotes shown for the three volumes and three Celotex thicknesses.

The variation of number of containers for a constant volume of 30 gal is shown in Fig. 2. As N increases, the mass necessary for criticality decreases, and the curves for the three thicknesses of Celotex asymptotically approach the $N = \infty$ values of mass. Also shown in the figure, for comparison, is the line depicting the data from the N16.5 Standard.

The reactivity response of a given array to perturbations is interpretable as the reactivity response of an individual unit in the array to its mass change. Thus, the reactivity associated with the total change in Celotex thickness examined, which may be interpreted as a 20% range of density variation, is on the order of 3%.

Definition of the magnitude of reactivity changes corresponding to perturbations implicit in the regulations, such as illustrated here, is necessary to examination of the criteria for criticality assessment in transportation.

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Criticality Calculations for Fast Reactor Fuel Storage Facilities, J. M. Otter, V. J. Pecora, E. R. Specht (AI)

Criticality calculations for sodium-filled fast reactor fuel storage facilities were made for various full loadings of driver assemblies to assure that criticality criteria would be met. Calculations with B_4C included in various ways were also made to provide guidance for fixed poison designs which enable more fuel assemblies to be stored in a given volume.

The space-energy coupled neutron transport calculations needed to evaluate criticality extend to domains that are not significant in fast reactor core calculations. Spatially, significant flux shaping across a fuel assembly cell occurs because of the large pitch-to-fuel-duct-diameter ratio. The latter also produces much higher axial leakage. Spectra are softer than typical LMFBRs because of the higher sodium volume fractions. Consequently, resolved resonance reactions in fuel isotopes are more important. In LMFBR cores, materials are normally represented in a homogeneous manner, for which a satisfactory formulation for effective resonance cross sections has been evolved. Proper formulations have not been derived for models where the fuel assembly is explicitly represented, such as fuel assembly cell or three-dimensional (3-D) models needed to account for cell flux shaping.

Both finite-difference RZ(DOT-III Code) and Monte Carlo 3-D (KENO-IV Code) calculations of criticality were made. Corrections to RZ calculations were made for homogenization of the fuel (axial leakage) and to 3-D calculations for basing effective resonance cross sections on the equivalent homogeneous mixture for isolated

TABLE I

Diffusion-Based Criticality Calculation Components and Uncertainties for the Upper Limit Configuration

	δk_{eff}	k_{eff}
Base case criticality (no B_4C)		0.966 ± 0.009
Worth of B_4C annulus	-0.035 ± 0.003	
Correction for geometry approximations	+0.005 ± 0.002	
Correction for homogenization of fuel in RZ system model	-0.006 ± 0.008	
Normalization to experiment	+0.005 ± 0.007	
Final criticality		0.935 ± 0.014

TABLE II

Monte Carlo-Based Criticality Calculation Components and Uncertainties for the Upper Limit Configuration

	δk_{eff}	k_{eff}
Base case criticality		0.906 ± 0.008
Correction for geometry approximations	+0.005 ± 0.003	
Correction for extra spatial flux shaping on resonance reaction rates	+0.017 ± 0.010	
Normalization to experiment	± 0.000 ± 0.009	
Final criticality		0.928 ± 0.016

lumps. Results were normalized to fast reactor experimental critical assemblies with similar spectra, since no benchmarks for fast reactor fuel storage arrays are available. Uncertainties were estimated.

Boron was found to be very effective in reducing criticality due to the relatively soft spectra. Replacing 1% of the stored fuel assemblies in the highest worth positions with naturally enriched B_4C rods reduced the criticality by about 3%. Borated steel storage tubes, small interstitial B_4C rods, and a B_4C annulus near the tank center were also found to be tractable.

Good agreement was obtained between finite-difference and Monte Carlo methods. Results for the upper limit criticality configuration (all fresh fuel) for the fuel storage facility of FFTF are given in Tables I and II. The facility contains 466 storage positions in 10 concentric rings with a 9-in. minimum spacing. A $\frac{1}{2}$ -in.-thick, 45-in.-o.d. B_4C annulus is included. The tank is filled with 400°F sodium. Full loadings of driver fuel assemblies with average discharge burnup (60,000 MWd/MTM) had criticalities about 0.09 δk below fresh fuel loadings.

Cell flux shaping and resonance heterogeneity effects were found to be important in criticality calculation for fast reactor fuel storage facilities. Techniques to account for these effects were included in finite-difference and Monte Carlo calculations of criticality. Very good agreement between methods was obtained. The effectiveness of the inclusion of boron in increasing facility capacity was demonstrated.

Critical Measurements of a Water-Reflected Enriched Uranium Sphere, Cleo C. Byers, Jerry J. Koelling, Gordon E. Hansen, David R. Smith (LASL), Howard R. Dyer (UCC-OR)

Experimental determination of the critical state of a fully reflected 97.67% enriched uranium metal sphere has been made to provide a benchmark value with which machine calculations may be compared. The sphere mass and density are 22,159.92 g and 18.794 ± 0.010 g/cm³, respectively. The total amount of impurities in the metal is <390 μ g/g. The major impurities are listed in Table I.

Figure 1 illustrates the experimental arrangement in which a 60-cm-diam fill tank was filled with water to a depth of approximately 70 cm. This tank was placed on

TABLE I
Chemical Analysis of Metal ($\mu\text{g/g}$)

P < 100	Mo < 50	Fe < 10
Ca < 10	Nb < 10	Sr < 20
Zn < 10	Ni < 15	W < 100
As < 10	Si < 25	Others < 30

the hydraulic lift of the critical assembly facilities Venus machine and was connected by a 5-cm-diam flexible line to a similar stationary experimental assembly tank.

The assembled uranium sphere was placed on a Plexiglas stand in this stationary tank containing two horizontal dry well ports approximately 10 cm below the bottom of the sphere to accommodate BF_3 neutron detectors.

Water could be transferred into and out of the assembly tank in a controlled manner by raising or lowering the hydraulic lift from the remote control console for this assembly. Additionally, the water could be drained rapidly from the assembly tank by activation of a scram valve in a 5-cm line in the bottom of the tank.

In the first series of measurements, a neutron source was positioned at the top of the tank directly above a beryllium mockup of the uranium assembly. In this way the water level could be calibrated on the ram position indicator with relation to the assembly and a series of unmultiplied counts taken to be used in the multiplication measurements during the approach to critical.

The uranium hemispheres as received proved to be too reactive and they were subsequently remachined to provide the final configuration.

Fig. 1. Experimental setup at the LASL critical assembly facility.

Fig. 2. Reactivity versus water height above sphere.

The sphere, mounted on the support stand, and the water were allowed to reach an equilibrium temperature of 16.3°C prior to the final measurement. Figure 2 shows the resulting reactivity versus water height above the sphere.

At the 16.5-cm water level, the observed 30.8 ± 0.2 cents above delayed critical corresponds to the reproduction number $k = 1 + 0.308 \beta_{\text{eff}} = 1.0020 \pm 0.0002$. By calculation, replacement of the support stand by water reduces k by 0.0018 ± 0.0004 , and extension of the water thickness to infinity increases k by 0.0001. By specifying the nominal density, we can transfer the geometry uncertainty to an equivalent reproduction number uncertainty of ± 0.0002 . Thus, for the specification

$$r_{\text{core}} = 6.5537 \text{ cm}$$

$$N(^{234}\text{U}) = 0.00053 \text{ atoms/b-cm}$$

$$N(^{235}\text{U}) = 0.04703 \text{ atoms/b-cm}$$

$$N(^{236}\text{U}) = 0.00010 \text{ atoms/b-cm}$$

$$N(^{238}\text{U}) = 0.00049 \text{ atoms/b-cm}$$

$$r_{\text{reflector}} = \infty$$

$$N(\text{H}) = 0.06679 \text{ atoms/b-cm}$$

$$N(\text{O}) = 0.03340 \text{ atoms/b-cm},$$

we obtain $k = 1.0003 \pm 0.0005$.

Pulsed Reactivity Measurements of Large ^{235}U -Al Castings in H_2O , D. J. Pellarin, J. L. Jarriel (DuPont-SRL)

The margin of error in reactivity calculations with KENO-IV (Ref. 1) was investigated over a wide range of subcritical reactivities for H_2O -moderated lattices of massive, highly enriched ^{235}U -Al alloy fuel forms. A previous paper² describes pulsed reactivity measurements and calculations for regularly spaced, rectangular, subcritical arrays of large alloy ($1.94 \text{ kg } ^{235}\text{U/ft}$) fuel castings in H_2O . In this paper, the measurements are extended to castings with $5.30 \text{ kg } ^{235}\text{U/ft}$, the pulsed experiments are more accurately analyzed by the αv^{-1}

method,³ and the measurements for both 7-assembly hexagonal and 2×3 square pitch are compared with KENO-IV calculations.

The fuel pieces used in the experiments were bare, annular uranium-aluminum alloy castings (14.796-cm i.d.; 22.019-cm o.d.). This alloy was 30.5 wt% uranium (76.97 wt% ^{235}U , 10.54 wt% ^{236}U , 11.49 wt% ^{238}U , and 1.00 wt% ^{234}U) with an alloy density of 3.545 g/cm³. The fuel assemblies were 105.0 cm high.

Pulsed reactivity measurements were made in the Subcritical Experiment Facility⁴ at the Savannah River Laboratory. A rigid aluminum support frame maintained accurate spacing within the arrays. The lattices were fully flooded to achieve an effective infinite H_2O reflection. A KAMAN model A-800 pulsed-neutron generator was located at the midplane ~15 cm from the core. Neutron detectors were placed in the low importance region on the radial centerline of representative fuel assemblies. The measured fundamental-mode prompt-neutron decay constant, α_0^p (meas), was determined by standard pulsed-neutron techniques (Table I).

KENO-IV is a multigroup Monte Carlo criticality code that uses 16-group Hansen-Roach cross sections.⁵ The flexible input geometry permitted exact three-dimensional (3-D) modeling of the 2×3 arrays with the cuboid geometry option. Modeling of the hexagonal arrays required the generalized geometry routine. A 20-cm-thick H_2O reflector with adjoint biased weights⁶ was used to model the actual reflector region. The calculated reactivity, k_{eff} (calc), for each lattice is given in Table I.

KENO-IV was modified to use α_0^p (meas) to transform the calculation from a static to a prompt multiplication solution. The prompt multiplication eigenequation is given by

$$\left[\frac{(1 - \beta_T) \chi_T \bar{P}}{k_p} - \bar{D} - \frac{\alpha_0^p}{v} \right] \phi^p = 0,$$

where $\bar{P}$ and $\bar{D}$ are the production and destruction operators, respectively, χ_T is the total neutron energy distribution from fission, β_T (0.0065) is the total delayed-neutron fraction, v is the neutron velocity, ϕ^p is the prompt-neutron flux in the persisting mode, and k_p is the eigenvalue (or scale factor) that forces the left side to zero.³ The normal k_{eff} search in KENO solves for

Fig. 1. KENO-IV margin of error.

$$\left[\frac{\chi_T \bar{P}}{k_{\text{eff}}} - \bar{D} \right] \phi_s = 0.$$

When the production and destruction operators are transformed by multiplying by $(1 - \beta_T)$ and including α_0^p/v , respectively, a KENO solution for k_{eff} becomes a solution for k_p , where k_p is a direct measure of the margin of error of KENO. The normalized reactivity is given by

$$k_{\text{eff}} (\text{norm}) = \frac{k_{\text{eff}} (\text{calc})}{k_p}.$$

The results are summarized in Table I. The margin of error in the KENO calculations is shown in Fig. 1.

KENO-IV calculates the subcritical reactivity for these massive, highly enriched ^{235}U -Al alloy fuel lattices moderated by H_2O over a wide range of reactivities with an error margin that is less than $+0.012$ in k_{eff} . No significant systematic differences are observed between the square and hexagonal pitch arrays.

TABLE I
Calculated and Experimental Results

Pitch, cm	Lattice ^a	k_{eff} (calc)	k_{eff} (norm)	α_0^p (meas), s ⁻¹	k_p
26.5	2×3	0.982 $\pm$ 0.002	0.977 $\pm$ 0.003	- 538 $\pm$ 3	1.005 $\pm$ 0.002
27.1	2×3	0.957 $\pm$ 0.002	0.945 $\pm$ 0.003	- 810 $\pm$ 3	1.013 $\pm$ 0.002
28.1	2×3	0.921 $\pm$ 0.002	0.912 $\pm$ 0.003	-1266 $\pm$ 8	1.010 $\pm$ 0.002
30.1	2×3	0.846 $\pm$ 0.002	0.844 $\pm$ 0.003	-1967 $\pm$ 11	1.002 $\pm$ 0.002
29.3	Hexagonal	0.970 $\pm$ 0.004	0.967 $\pm$ 0.005	- 643 $\pm$ 5	1.003 $\pm$ 0.003
30.3	Hexagonal	0.912 $\pm$ 0.005	0.905 $\pm$ 0.006	-1217 $\pm$ 5	1.008 $\pm$ 0.004
31.3	Hexagonal	0.875 $\pm$ 0.004	0.872 $\pm$ 0.005	-1579 $\pm$ 11	1.003 $\pm$ 0.003
32.3	Hexagonal	0.843 $\pm$ 0.005	0.845 $\pm$ 0.007	-1915 $\pm$ 29	0.998 $\pm$ 0.005
33.3	Hexagonal	0.816 $\pm$ 0.005	0.810 $\pm$ 0.006	-2269 $\pm$ 52	1.008 $\pm$ 0.004

^a120,000 neutron histories were processed in the KENO calculations for the 2×3 lattices; 30,000 neutron histories were processed for the hexagonal lattices.

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Criticality of Thin Flat Foils Versus Spherical Shells of ^{239}Pu , K. R. Yates (BCL)

Small mass critical configurations have always been of interest and importance in criticality safety. Olson and Robkin¹ have previously calculated critical masses for thin flat foils of ^{235}U and ^{239}Pu immersed in large D_2O reflectors. This study was based on the mathematical observation that an infinite slab of fissionable material with $\eta > 1$, immersed in an infinite nonabsorbing reflector, would approach zero thickness. The temperature of the fissile core and D_2O reflector was lowered to 4 K in the calculations "to rethermalize neutrons striking the core and to take advantage of the absorption characteristics of the fuel." Under these conditions a minimum critical mass of 35 g was obtained for ^{235}U and 22 g for ^{239}Pu . The method of calculation was a two-group diffusion analysis of a thin, centrally located core.

No KENO² calculations have been reported for systems of this type and no attempt has been made to verify these calculations. Furthermore, the concept of infinite slab takes on a different physical meaning if curved space is considered. In curved space, travel from a point in a given "straight" path will eventually bring the traveler back to the starting point. Therefore, an infinite slab in curved space may be better visualized as a thin spherical shell and a thin spherical shell may have a higher K_{eff} than a flat Cartesian slab.

A series of KENO calculations was run to determine the minimum critical mass of ^{239}Pu as a thin square slab and a thin spherical shell immersed in a large volume of D_2O . These calculations assume that the fissile core and D_2O reflector are at room temperature as provision was not available to adjust the neutron cross sections to 4 K as in the calculations of Olson and Robkin. Hansen-Roach cross sections were used in all calculations.

Table I shows K_{eff} for ^{239}Pu square slabs of 550, 505, and 440 g centered in a 240-cm cube of D_2O . The 240-cm D_2O cube was the largest reflector that could be run because of computer time limitations. It can be seen that a square slab with an edge length of 100 cm (0.00276 cm thick), weighing 550 g, achieved criticality in the 240-cm cube of D_2O .

Table II shows K_{eff} for 550-g slabs of ^{239}Pu of various sizes centered in D_2O cubes of 220, 200, and 180 cm. These calculations were done to determine if the different

TABLE I

D_2O Cubical Reflector is Constant at 240 cm

	Slab Edge Length			
	80 cm	90 cm	100 cm	110 cm
Mass ^{239}Pu	K_{eff}^a	K_{eff}	K_{eff}	K_{eff}
550 g	0.986	0.970	1.014	0.991
505 g	0.970	0.954	0.962	0.942
440 g	0.918	0.933	0.903	0.926

^a $1\sigma = 0.012$ in all cases.

TABLE II

^{239}Pu Mass is Constant at 550 g

	Slab Edge Length			
	80 cm	90 cm	100 cm	110 cm
D_2O Edge Size	K_{eff}^a	K_{eff}	K_{eff}	K_{eff}
220 cm	0.966	0.954	0.959	0.954
200 cm	0.938	0.940	0.955	0.942
180 cm	0.941	0.915	0.912	0.919

^a $1\sigma = 0.012$ in all cases.

TABLE III

^{239}Pu Mass is Constant at 390 g

	Spherical Shell O.D.			
	46 cm	50 cm	54 cm	58 cm
D_2O Edge Size	K_{eff}^a	K_{eff}	K_{eff}	K_{eff}
240 cm	0.980	1.018	1.008	0.960
220 cm	0.964	0.982	0.972	0.970

^a $1\sigma = 0.012$ in all cases.

neutron energy spectrum of a smaller cube might result in a greater K_{eff} than in the 240-cm cubes. No other critical configuration was found.

Table III shows K_{eff} for 390-g thin spherical shells of ^{239}Pu of various diameters centered in D_2O cubes of 220 and 240 cm. The greatest value of K_{eff} occurred for a spherical shell of 50-cm o.d. (0.0025 cm thick) filled with and centered in a 240-cm cube of D_2O .

It appears that a thin spherical shell requires only about 70% of the mass of a thin square slab to achieve criticality under essentially identical conditions of D_2O reflection at room temperature. If, at 4 K, the thin spherical shell also requires only 70% of the mass of a thin slab to achieve criticality, perhaps only $22 \text{ g} \times 390/550 = 15.6 \text{ g}$ of ^{239}Pu would be necessary for criticality.

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Moderation Control for Purposes of Nuclear Criticality Safety

Criticality Evaluation of Low-Density Moderation in PWR Fuel Storage, *F. M. Alcorn (B&W), invited*

The safety of fuel assembly storage arrays has been demonstrated at B&W's Commercial Nuclear Fuel Plant (CNFP) for conditions of achievable low-density interspersed moderation. The CNFP has the capacity to store 392 PWR fuel assemblies in two adjacent arrays; in the first, assemblies are stored in a 36- $\times$ 36-in. center-to-center array while, in the other, assemblies are stored in a 21- $\times$ 38-in. array. Reactivity has been evaluated for both arrays assuming that they were subjected to interspersed water of varying density.

Handley^{1,2} has previously documented a peak in the multiplication constant (K_{eff}) at low-density interspersed moderation for arrays of high enriched uranium. Work reported in this paper deals with three aspects of low-density moderation:

1. reactivity for arrays of low-enriched uranium fuel assemblies assuming interspersed low-density moderation

2. methodology for calculating reactivity of these arrays

3. credibility of achieving low-density interspersed moderation.

The fuel assembly assumed for these calculations is the B&W Mark-B canless design. The assembly is a 15 $\times$ 15 square lattice of 208 fuel rods and 17 control rod/instrument positions. The pin pitch is 0.568 in.; the Zircaloy cladding o.d. is 0.43 in.; the active fuel length is 12-ft; and the assembly outer dimensions are 8.45 $\times$ 8.45 in. Maximum enrichment assumed 4.0 wt%.

Reactivity was calculated using both the PDQ-7 and KENO computer codes. Results from the two codes for the same array are shown in Fig. 1; the array calculated is infinite in the floor plane but 12 ft high with concrete reflection on one end. Spacing for this array is 21 $\times$ 31 in. center-to-center. It is noted that the two results are in excellent agreement at full water density but were significantly different at low-density moderation. This illustrates a typical problem in the use of codes. PDQ-7 has been benchmarked and shown to be quite accurate with appropriate cross sections when calculating fully

Fig. 1. K_{eff} vs percent water density in PWR fuel assembly storage array.

moderated arrays of uniformly spaced low-enriched uranium rods. PDQ had been used to evaluate storage arrays in the pre-KENO era and had been used to estimate low-density effects using transported corrected cross sections. With the advent of KENO one has a better calculative model and thus a better evaluation of reactivity at low moderation; or does one? There are no critical experiments on arrays of this type, certainly none with low-density water. One must therefore assume that the higher K_{eff} at low density is governing. The other array encountered in the CNFP is a 36- $\times$ -36-in. array; KENO predicts a maximum K_{eff} for this array of 0.95 at approximately 5% interspersed water.

A number of other parameters were investigated for arrays of fuel assemblies. For example, reactivity of storage arrays is highly dependent on the finite size of the array. K_{eff} for a finite array of 100 assemblies on a 21- $\times$ -38-in. pitch is <0.9 at 5% interspersed moderation versus 1.10 for the infinite array.

What is the credibility of achieving 5% density water moderation. Engineering evaluations have led to the conclusion that levels of moderation that give unacceptable values of K_{eff} cannot be achieved. Saturated steam corresponds to a density of <0.006 g/cm³. Water sprays such as might come from rain, fog nozzles, and sprinklers will not result in such moderation. Examination of work by Yao and Kalelban³ led to the conclusion that four overlapping sprinklers resulted in a water density <0.001 g/cm³. Similarly, water films were ruled out. The probability that a solid stream of water directed into an array would affect even 100 assemblies is so remote as to be negligible. Nevertheless, existing water sources (including fire hose) and flammable material have been held to a minimum (but not excluded) in the storage arrays. There are certainly some solid materials such as styrofoam which if placed in an array in sufficient quantity would lead to an unacceptable K_{eff} value; this is an obviously and easily controllable consideration.

In conclusion, storage arrays for new PWR fuel assemblies may be built with assurance of safety with respect to low-density interspersed moderation. Center-to-center spacing in the arrays and array size may be considered as parameters which may be varied to contribute to the available safety margin but are not requirements. The degrees of low-density moderation which might lead to compromising safety in arrays are not achievable by accident.

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Criticality Evaluation of Large Dry UO₂ Pellet Storage Facility, L. F. Rodriguez, W. C. Peters (GE-NC), invited

A large computer-controlled fuel pellet storage and retrieval facility has been installed at the General Electric Company Fuel Manufacturing Facility at Wilmington, North Carolina. This system, referred to as the pellet stacker, has been demonstrated to meet established nuclear safety criteria as the result of criticality calculations using the KENO Monte Carlo Code.¹ Although the

pellet stacker is not designed to be operated under moderation controls, certain design features were included to preclude water and other moderating media from entering the stacker. This enhances the safety and has permitted the deletion of the pellet stacker from future interacting solid-angle calculations.

This paper summarizes the criticality evaluation of the pellet stacker, which includes: (a) demonstration of the criticality safety of the pellet stacker when subjected to optimum moderation conditions, (b) evaluation of the interaction of the stacker with adjacent units, and (c) validation of criticality calculations.

The pellet stacker consists of two large storage racks ~2 ft deep $\times$ 12 ft high $\times$ 60 ft long, positioned parallel to each other, with 40-in. aisle space between the racks. The main structure is covered with a rainproof metal roof and walls. Both racks are divided into compartments which are used to store two types of containers. Pellet drawers are used for sintered pellets randomly stacked in molybdenum boats. Pellet trays are used to store up to three horizontal layers of pellets positioned on top of a corrugated platform inside the tray. Both the pellet boat drawers and pellet trays are constructed entirely of Type 304 stainless-steel sheets. Both containers include a hinged metal cover also constructed of the same material. The containers are transported within the pellet stacker using a motorized vehicle. Material access and egress is through small openings in the sides of the stacker structure.

Scoping calculations to evaluate different container spacing requirements were performed early during the design stages with General Electric diffusion codes and the KENO Monte Carlo with Hansen and Roach cross sections.² Since the KENO Code predicted much higher reactivity values than the diffusion codes, it was selected as the tool to analyze the final stacker design to assure conservative results.

Figure 1 gives KENO-calculated reflected reactivity values as a function of interspersed water densities between the containers, for both wet and dry fuel regions. The structure was assumed reflected by a thick concrete floor and a thick water jacket on the other 5 sides.

KENO calculations to evaluate the neutron interaction between the stacker and other nearby units were performed for varying accumulations of uranium. These calculations demonstrated the subcriticality of the interaction systems in the unlikely event of optimum water moderation introduced throughout the system. Optimum moderation within the pellet stacker was determined to account for 8% of the system reactivity. It was further demonstrated that the contribution to the system reactivity by a dry stacker was negligible ($<1.5\%$). This small reactivity increase is attributed more to neutron reflection provided by the pellet stacker than to neutron coupling between the stacker and the other units.

As demonstrated by the results described above, and since the stacker is designed to preclude the introduction of moderation media, it is concluded that the stacker array may be neglected during future interaction analyses for this area when using the solid-angle interaction method.

The General Electric MERIT Monte Carlo Code was used to verify that the KENO Code used with Hansen and Roach cross sections adequately predicted the reactivity of the stacker system. The MERIT Code, which uses ENDF/B-IV 190-group neutron cross sections and treats the resonance cross sections on a discrete energy basis, has been extensively validated against heterogeneous

Fig. 1. Reflected k_{eff} for pellet stacker.

UO₂-H₂O systems. A maximum negative bias of 0.9% has been established for uranium criticals.

The validation calculations with MERIT indicated that KENO, used with Hansen and Roach cross sections, overpredicts the reactivity for the water-flooded fuel case by <2% and underpredicts the dry fuel case by <1%. This is consistent with previous KENO validation calculations performed by General Electric in support of a fuel bundle shipment container license application.³

In summary, the subcriticality of the pellet stacker has been demonstrated under optimum moderation conditions. Since the stacker design precludes the introduction of moderating material, it has been demonstrated that the stacker may be neglected during interaction analysis for the area. This result demonstrates, for this particular application, the generally accepted concept (solid-angle interaction method) that accumulations with $K_{\infty} < 1.0$ may be neglected during neutron interaction calculations.

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Criticality Safety Practices for Handling Unmoderated Plutonium in Glove Boxes, C. L. Brown (BNWL), invited

This paper discusses criticality safety practices used to justify the application of moderation control to the processing of large unit masses of unmoderated plutonium in glove boxes. Unit masses up to 5 kg Pu have been

justified. This application of moderation control has been used throughout the nuclear industry for many years and is essentially standard practice.

Moderation control is one of the basic methods used to achieve criticality safety. The term means: limiting to some degree the amount of neutron moderating material that is mixed with fissile material. The moderating materials commonly encountered in plutonium operations are those that contain hydrogen: water, plastics, oils, waxes, organic compounds, etc. The less common moderating materials are carbon, beryllium, and deuterium; these materials are sometimes encountered but are easily assessed and controlled. For homogeneous plutonium systems, degree of moderation is expressed either as moderator-to-plutonium atom ratio (H/Pu, C/Pu, or Be/Pu), or as plutonium concentration.

Moderation control has been applied over a wide range of moderation limits. The most common application is to dry, normally unmoderated systems. Application has been made, however, to partially moderated systems where volume control is also exercised.

The difficulty in applying moderation control as a basis for criticality safety is in assuring the moderation limit will not be accidentally exceeded. The moderators of concern are water, plastics, and oils. Consideration must be given to all potential sources of these moderating materials. If complete control cannot be assured, the allowable plutonium unit mass is reduced accordingly.

The straightforward acceptance of moderation control as a principal basis for criticality safety in dry plutonium processing is attributed (at least in part) to the fact that operations are conducted in glove boxes. The glove-box barrier affords a great measure of control over moderators: Moderators needed in the glove box can be strictly limited and if necessary logged in and out; piping can be evaluated; fire fighting can be predicted and precontrolled; and glove-box floor designs can be taken into consideration in avoiding potentially unsafe situations if accidental flooding is considered credible.

A summary of common practices is given below:

Plutonium Mass

1. Plutonium unit masses are usually limited to ~45% of the unmoderated minimum critical mass on the assumption that only double batching is credible.
2. Higher and lower unit masses are used depending on the credibility of multiple batching.
3. For unmoderated PuO₂, unit masses up to 5.0 kg Pu are allowed.
4. For PuO₂-UO₂ blending operations in which hydrogenous binder is added, up to 4.0 kg Pu and 1 kg of moderator are allowed on the basis that total volume is also limited.

Plutonium Moderation

1. An H/Pu atom ratio up to 5 is commonly allowed on the basis that in the H/Pu range of 0 to 5, the smallest critical mass still occurs at H/Pu = 0.

Water Moderator

1. Small amount of water or other hydrogenous liquid is sometimes allowed in glove box for cleaning purpose on the basis that it will not be mixed with plutonium and the volume is small enough not to present a criticality potential if water were accidentally mixed with plutonium.

2. Furnace cooling coils are approved on one of the following bases: (a) coil water leak cannot result in moderated plutonium; (b) coil coolant is nonhydrogenous; (c) volume of coolant in reservoir is limited such that complete leakage into glove box would not result in criticality potential; (d) coolant is "pulled" through coils, rather than being pumped through under pressure; or (e) coolant flow valve is quick acting, located outside glove box, and is under continuous operator surveillance while cooling water is on.

3. Water for fire fighting is allowed to enter glove boxes based on two conditions: (a) vessels containing unmoderated plutonium would be subcritical if flooded; and (b) plutonium potentially in glove box will not exceed a safe slab thickness based on optimum moderation.

Oil Moderator

1. Lubricating oil is allowed on the basis that the volume of oil is insignificant as a potential moderator.

2. Hydraulic fluid line leakage under pressure is allowed on the basis that plutonium + fluid volumes will be small enough not to present a criticality problem. Containers of plutonium are required to be covered. In some operations, the volume of hydraulic fluid in a press system has been reduced to prevent significant glove-box flooding in the event of oil leakage under pressure. (Allowance can be given for chlorine content of certain hydraulic fluids.)

Binder Moderator (wax, oil, lard, etc.)

1. Binder addition to plutonium, adequate for the process, is allowed on the basis of strictly limited plutonium mass and blender volume.

Plastic Moderator

1. Powdered plastic for blending with PuO_2 is allowed on the basis of strictly limited plutonium mass and blender volume.

2. Plastic sample vials are allowed on the basis of a limited number of vials.

3. Metallographic samples (plutonium specimens mounted in plastic) are allowed based on a fixed range of H/Pu ratios determined by sample sizes.

4. Wherever plastic brushes are used, plastic brush bristles are allowed in the plutonium sweepings on the basis that the amount of plutonium swept together at one time is safely limited.

5. Plastic in single or double plastic bags used for contamination control is negligible from the standpoint of neutron moderation.

Other Liquids

1. Nonhydrogenous liquids such as CCl_4 and Freon-113 are not considered moderators and are allowed in glove boxes without a limit for criticality safety.

The above practices have been used for many years as the basis of accepting moderation control for criticality safety in processing plutonium. These practices perhaps can provide a foundation for evaluating the use of moderation control in other parts of the nuclear fuel cycle.

A Survey of Criticality Parameters for ^{239}Pu in Organic Media, L. C. Davenport, J. K. Thompson (BNWL), invited

At various points throughout the fuel cycle, plutonium-bearing compounds and separated plutonium may occur in

mixtures with various organic media. Such occurrences are most frequent during reprocessing (which uses organic solvents for extraction) and subsequent storage (such as in plastic bags and containers). As has been documented in the past¹ and again more recently,² mixtures containing carbon and hydrogen can moderate neutrons more effectively than water. To quantify this effect, criticality parameters have been calculated for homogeneous mixtures of some common organic compounds with ^{239}Pu . These calculations provide a first step toward a systematic investigation of criticality in organic media, which will eventually provide a generic model for use in criticality investigations.

Criticality parameters of homogeneous ^{239}Pu systems have been calculated for the three basic geometries: sphere, infinite cylinder, and infinite slab. Both bare and reflected (full reflection modeled by 20.3 cm of water) systems have been investigated. The calculations were performed using the one-dimensional (1-D) diffusion theory code HFN,³ and 18-group cross sections generated from the Battelle Master Library by the GAMTEC-II code.⁴ The calculated criticality parameters include (a) sphere mass, volume, and diameter, (b) infinite cylinder diameter, and (c) infinite slab thickness and areal density. The systems investigated are listed in Table I.

Full-water-reflected sphere critical mass and infinite cylinder critical diameter are presented for nine of the moderators in Figs. 1 and 2. (The other three are so poor neutronically that the minimum critical mass values are in the kilogram range, and therefore are beyond the scope of this paper.) Additional results for other parameters calculated, for both bare and fully water-reflected systems, will be presented at the meeting and will be published.

Sphere critical mass is plotted as a function of ^{239}Pu concentration in homogeneous solutions in Fig. 1. The excellent neutronic properties of polyethylene as compared to water are attributable to an increased hydrogen atom density, and the replacement of oxygen by more of the better moderator carbon. The other materials are similar to water, or are less neutronically favorable.

TABLE I
Homogeneously Moderated ^{239}Pu Systems Studied

Moderator	Composition	Density (mg/m ³)	Minimum Critical Mass (kg ^{239}Pu) ^a
Polyethylene	$(\text{CH}_2)_n$	0.96	0.370
Ethylene Glycol	$\text{C}_2\text{H}_6\text{O}_2$	1.109	0.495
N-Dodecane	$\text{C}_{12}\text{H}_{26}$	0.749	0.515
Water	H_2O	1.000	0.530
DBBP	$\text{C}_{12}\text{H}_{27}\text{PO}_3$	0.989	0.535
LA-2	$\text{C}_{24}\text{H}_{51}\text{N}$	0.80	0.545
Polystyrene	$(\text{CH})_n$	1.06	0.640
Benzene	C_6H_6	0.876	0.890
Ammoniated Water	$\text{H}_2.32\text{N}0.32\text{O}.68$	0.826	0.940
Hydraulic Oil	$\text{C}_{40}\text{H}_{33}\text{Cl}_6\text{PO}_2$	1.28	b
Freon-113	$\text{C}_2\text{Cl}_3\text{F}_3$	1.56	b
Carbon Tetrachloride	CCl_4	1.595	b

^aThe fully water reflected sphere is assumed to have a $\pm 5\%$ uncertainty in minimum critical mass, due primarily to cross-section uncertainties.

^bIt appears that the minimum critical mass occurs for theoretical density metal. In the range from 20 to 500 g ^{239}Pu /litre, hydraulic oil has a minimum mass of 13.1 kg ^{239}Pu , while both Freon-113 and CCl_4 have their lowest masses at the 500 g ^{239}Pu /litre end of the curve where the masses are still decreasing rapidly as concentration increases.

Fig. 1. Sphere critical mass, ^{239}Pu -moderator, 100 wt% ^{239}Pu in Pu, fully water reflected.

Fig. 2. Critical infinite cylinder diameter, ^{239}Pu -moderator, 100 wt% ^{239}Pu in Pu, fully water reflected.

Ethylene glycol and dodecane are slightly better neutronically than water, having ~7 and ~3% lower minimum critical masses of ^{239}Pu , respectively, than a ^{239}Pu - H_2O mixture. Dibutyl butyl phosphate (DBBP) and the very similar tributyl phosphate (TBP), which has one more oxygen atom per molecule than DBBP but a slightly lower density, are used extensively in solvent extraction. DBBP is very similar to water in terms of spherical critical mass and critical cylinder diameter. TBP systems were not calculated, but should have nearly the same criticality parameters.

The secondary amine LA-2 and ammoniated water (32 atom% NH_3 , 68 atom% H_2O) were investigated because they include nitrogen which has a moderate thermal-neutron absorption cross section. The nitrogen causes a small shift in the plutonium concentration at which the minimum critical mass occurs. Representative compositions that were assumed for these materials in the calculations are listed in Table I. For the secondary amine LA-2, the critical mass curve crosses that for water. Systems containing LA-2 are therefore more reactive than water-moderated systems at higher plutonium densities.

Polystyrene and benzene have relatively poorer neutronic properties than the other materials for which curves are plotted. This is primarily due to reduced hydrogen density.

The remaining three moderators are so poor neutronically due to the presence of chlorine atoms that the critical mass curves lie outside of the area graphed. It appears that for all three materials the minimum critical mass would occur for theoretical density metal. Hence, for these systems criticality safety limits based on analyses of well-moderated ^{239}Pu -water systems are much more restrictive than necessary.

Full-water-reflected critical infinite cylinder diameters are plotted as a function of ^{239}Pu concentration in Fig. 2 for the same homogeneous systems. Very similar behavior is evidenced, with polyethylene systems being much more reactive than water-moderated systems, n-dodecane and ethylene glycol systems being slightly more reactive, and LA-2 showing a crossover behavior.

From these few examples, the essential ingredients of a generic moderator model can be distilled. Examination of the figures indicates that the most important characteristics for determining criticality are H/Pu ratio, C/Pu ratio, and X/Pu ratio, where X denotes all other constituent atoms. They may be used to categorize materials of interest in terms of neutronic favorability. Recasting the results, there do appear regularities in the data as anticipated. Thus one may conclude that a generic model of criticality for ^{239}Pu in organic media is a realizable possibility, one that can be of general use to criticality safety investigators.

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Moderation Control: It Depends . . . , I. E. Fergus (Allied Chem), invited

Moderation control in operations with fissile material makes some proposals technically feasible and can result in significant cost savings in others. Often designs can be simplified and processes made more efficient. However, there are many problems and pitfalls both in calculations and implementation. Therefore, current criticality safety philosophy at the Idaho Chemical Processing Plant (ICPP) permits moderation control only within shielded facilities capable of containing the consequences of an accidental criticality. This requirement was imposed midstream in the development of several projects and necessitated some redesign. I will discuss (a) a shielded recovery process that was not changed, and (b) the before and after designs of an unshielded storage rack to illustrate the above points.

The Rover Fuels Processing Facility¹ will be partly moderation controlled. The "dry" portion contains, among other items, a series of fluidized bed burners, hoppers, and collection vessels that form a large array of roughly cylindrical units. The fuel is highly enriched

Fig. 1. k_{eff} vs H/X for redesigned product storage rack.

uranium carbide in a graphite matrix that will be oxidized and selectively elutriated. Several of the Rover vessels are critically safe for only limited quantities of moderator. This is necessary because the fissile material is in the form of a low density ash that is pneumatically transferred; large volumes are required for efficient solids separation. Small diameter vessels would probably experience routine plugging in the fines disengaging sections. So, all sources of moderation, including the humidity in the air, will be controlled. Nevertheless, calculations were made for accidental insertion of water into the largest diameter vessel to determine safety margins. Adding homogeneously distributed water gave almost a straight line increase in k_{eff} with H/X . Criticality was estimated to occur at $H/X = 2.9$; but, one should exercise caution in translating this type of data into "critical mass" of water or other hydrogenous compounds. A small wet sphere in the center of the vessel requires much less total moderator for criticality, all other assumptions being equal. Therefore, this system and all interconnections have been carefully studied to ensure that all accidents that could moderate the fissile material have been rendered incredible.

The other problem, the storage rack, also has a calculational pitfall. The rack is a rectangular array of square cells each holding two cans. The fissile material is highly enriched uranium oxide that has been dried to a granular solid in a fluidized bed denitrator. In the original concept the contents of the cans were to be moderation controlled, but interspersed moderation was assumed between rack positions. This was because the design is an open framework, and there are exposed water and sewer lines overhead. An H/X of 1 was conservatively assumed within the cans and maximum k_{eff} occurred at about 5% water density between them. The curve was similar to the array curve in Clayton's *Anomalies of Criticality*,² having one peak at low density and another at full density water. For this study, the positive reactivity effect of interspersed moderator was small. When it was determined that all degrees of internal moderation would have to be considered as well, the optimum interspersed moderation from the first study was used. The spacing between positions was increased and calculations were made for various H/X values. Results are plotted in Fig. 1. The k_{eff} rose rapidly in a straight line fashion

until all voids were filled. Additional water then displaced solids and k_{eff} rose slowly, maximizing near $H/X = 40$. Yet, these expensive studies had overlooked the case of a single position optimally moderated and fully reflected: that was addressed separately and found to be safe. The new design has only 14% less capacity but will be somewhat harder to use.

In conclusion, moderation control is being applied in a limited fashion at ICPP:

1. Protection of personnel by shielding is an absolute must.
2. Cost savings alone is not enough justification. Technical necessity is also required.
3. Moderation in the environment must also be controlled or assumed present.
4. Interconnections and upstream failures must be considered and contingencies allowed for.

Evaluations must be thorough and operational problems must be overcome if this control is to be employed at all. The added risk plus the cost of special precautions must be weighed against the benefits of its use in each case. Hopefully, as experience in using moderation control accumulates, the problems will become clearer and appropriate applications can be identified.

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Criticality Analysis of Dry Fuel in ANO-2 Fuel Storage Pool, J. T. Brantley, R. E. Machado (Middle So. Serv), J. C. Ho (NAIC), T. W. Schnatz (Middle So. Serv)

The Arkansas Nuclear One—Unit 2 nuclear power plant is expected to commence fuel loading in October 1977. The size of the new fuel storage pool necessitates storing two-thirds of the initial core in the spent fuel storage

pool. FSAR criticality analyses guaranteed safe storage of the fresh core fuel in the spent fuel pool under flooded conditions.¹ It was found to be expedient to store the fuel dry. A criticality analysis was performed to ensure subcriticality of the fuel under all moderator (water) densities since a fog or mist introduced into the dry storage rack can bring about a reactivity greater than that at flooded conditions. The PDQ program² was used to calculate k_{eff} 's; the EPRI-CELL program³ was used to generate the cross-section input for PDQ. The KENO-IV program⁴ was used to check the diffusion results. It was found that the dry fuel could be safely stored.

The fuel is confined in the fuel storage racks by square stainless-steel cans with a nominal thickness of 0.25 in. The assembly pitch is nominally 13.125 in. The worst combination of tolerances and dimensions leads to a 12.64-in. assembly pitch. The rack was assumed to contain assemblies loaded with 3.1% enriched fuel with no burnable poison rods.

The calculational strategy was to use EPRI-CELL and PDQ to do sensitivity studies and to use KENO for verification of PDQ results and performing the final criticality calculation. PDQ and KENO k_{eff} 's were to be compared at water densities of 0.15 and 1.00 g/cm³. (The maximum k_{eff} was expected to occur in the water density range of 0.1 to 0.2 g/cm³.) If KENO verified the PDQ results, PDQ would be used to find the water density at which the maximum k_{eff} occurred. A KENO run would then be made at the worst water density so subcriticality could be assured.

The EPRI-CELL program was used to obtain macroscopic cross sections for input to PDQ. EPRI-CELL's capability to discretely represent several material regions was utilized. The fuel cell along with the associated portions of the stainless steel can and interassembly water was represented by one model consisting of seven concentric rings or "zones." The respective zones represented the fuel, the cladding, the moderator, the homogenized water and control rod guide tubes within the assembly, the water between the assembly and the can, the stainless-steel can, and the water between the assemblies. This model accounts for the effect of each component on the neutron flux spectrum as well as for any important self-shielding phenomena. The cross sections for all components came from this single model by appropriately editing the different zones.

The PDQ model is a two-dimensional (2-D) representation of the assembly utilizing four neutron energy groups and boundary conditions simulating zero neutron leakage. Both a nominally dimensioned quarter assembly model and an adversely dimensioned full assembly model of the rack were used. For the adversely dimensioned model, it was assumed that the assembly was located in one corner of the can.

The results of the PDQ and KENO verification calculations are shown in Table I. PDQ and KENO agree well at the greater water density. The calculations indicate that the EPRI-CELL/PDQ model can be used for sensitivity studies and that, for low water densities, EPRI-CELL/PDQ is conservative. Comparisons of KENO results and critical measurements are also presented in Table I; excellent agreement is shown.

EPRI-CELL and the adversely dimensioned PDQ model were used to calculate cross sections and k_{eff} 's over the water density range from 0.05 to 1.00 g/cm³. The maximum calculated k_{eff} of 0.9647 occurred at a water density of 0.15 g/cm³. The k_{eff} was shown to decrease sharply as water density increased or decreased from 0.15 g/cm³. The minimum calculated k_{eff} was 0.8133 at a water density of 0.75 g/cm³.

TABLE I
Results of PDQ and KENO Verifications

EPRI-CELL/PDQ Compared to KENO			
Purpose of Keno Run	Water Density, g/cm ³	KENO k_{eff}	PDQ k_{eff}
Verification of PDQ ^a	1.00	0.818 ± 0.005	0.810
Verification of PDQ ^a	0.15	0.904 ± 0.005	0.950
Final Criticality Calculation ^b	0.15	0.931 ± 0.004	0.965
KENO Compared to Criticals			
Description of Critical	Keno k_{eff}		Measured k_{eff}
Critical #103 by Johnson ⁵	0.998 ± 0.005		1.000
La Crosse cold critical ^{6,7,8}	1.008 ± 0.005		1.009

^aUsing nominal fuel rack dimensions

^bUsing worst-case fuel rack dimensions

A KENO calculation of the adversely dimensioned fuel assembly was performed at the 0.15 g/cm³ density resulting in a k_{eff} of 0.931 ± 0.004. Even after allowance for calculational uncertainty, the rack is shown to meet the ANSI N18.2 limiting k_{eff} of 0.98.

In summary, a criticality analysis of the ANO-2 spent fuel storage pool was performed. It was shown that fuel assemblies of up to 3.1 wt% ²³⁵U could be stored in the racks under dry conditions. The analysis was done using EPRI-CELL and PDQ with a minimum of reliance on Monte Carlo methods.

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Critical Separation Between Subcritical Clusters of 2.35 wt% ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons, S. R. Bierman, E. D. Clayton (BNWL)

An experimental program was initiated during the past year at the Battelle-operated Critical Mass Laboratory to obtain confirmatory, benchmark-type criticality data on conditions typically present in fuel element storage facilities and shipping casks. The results of the initial

series of measurements with 2.35 wt% ^{235}U enriched UO_2 rods in water are covered in this paper.

The experiments were carried out in a 1.8- x 3- x 2.1-m-deep tank provided with lattice grids and the required control and safety devices. The lattice grids

TABLE I
Critical Separation Between Subcritical Clusters
of 2.35 wt% ^{235}U Enriched UO_2 Rods in Water

FUEL CLUSTERS		CRITICAL SEPARATION BETWEEN FUEL CLUSTERS ^b (mm)
NUMBER IN ARRAY	LENGTH x WIDTH 20.32 mm SQ. PITCH (RODS) ^a	
1	20 x 18.08 ± 0.02	∞
3	20 x 17	119.2 ± 0.4
3	20 x 16	83.9 ± 0.5
3	20 x 15	63.9 ± 0.6
3	20 x 14	44.6 ± 1.0

^a DESCRIPTION OF FUEL RODS

CLADDING: 6061 ALUMINUM TUBING SEAL WELDED WITH A LOWER END PLUG OF 5052-H32 ALUMINUM AND A TOP PLUG OF 1100 ALUMINUM

TOTAL WEIGHT OF LOADED FUEL RODS: 917 gm (AVERAGE)

LOADING:

825 gm OF UO_2 POWDER / ROD, 726 gm OF U/ROD, 17.08 gm OF U-235/ROD
ENRICHMENT - 2.35 ± 0.05 w/o U-235
FUEL DENSITY - 9.20 mg/mm³ (84% THEORETICAL DENSITY)

^b PERPENDICULAR DISTANCE BETWEEN THE OUTER CELL BOUNDARIES OF THE FUEL CLUSTERS

were of 12.7-mm-thick Plexiglas containing 12.827 + 0.127-mm-diam holes equally spaced on a 20.32-mm square pitch. The fuel rods used in the experiments were nearly 1 m in length and are described in detail in Table I.

Measurements were made to determine the critical separation between three subcritical water-flooded and -reflected clusters of the fuel rods in a row. Measurements were also made to determine the effect each of the following materials, positioned in the water gap between the clusters of rods, had on the critical separation between the clusters of rods:

Boral, 7.13-mm-thick plate
304L Steel, 3.02- and 4.85-mm-thick plates
1.1 wt% B-304L Steel, 2.98-mm-thick plate
1.6 wt% B-304L Steel, 2.98-mm-thick plate
Copper, 3.37- and 6.46-mm-thick plates
1 wt% Cd-Copper, 3.57-mm-thick plate
Aluminum, 6.25-mm-thick plate
Zircaloy-4, 6.52-mm-thick plate
Cadmium, 0.29-, 0.61-, and 0.90-mm-thick plates.

As indicated, these materials were in the form of plates and were 356 mm wide by 915 mm long. Measurement results were obtained with the above materials (neutron poisons) positioned in at least two different locations in the water gap between the fuel clusters. One location was near the cell boundary of the center fuel cluster and the other was near the midpoint between the fuel clusters.

A diagram showing the arrangement of these plates and the fuel clusters is presented in Fig. 1. (Some of the features shown are concerned with future experiments and are not involved in this initial set of measurements). The critical separation determined in these experiments is indicated as X_c . Also in the figure, the thickness of the plate material is shown as t_p and the distance between the cell boundary of the center fuel cluster and the near surface of the plate is shown as G .

Measurements results from the experiments in which only water separated the subcritical clusters of fuel rods are presented in Table I. Whenever neutron poison plates were positioned in the water gap between the fuel clusters,

TABLE II
Experimental Data on Clusters of 2.35 wt% ^{235}U Enriched UO_2 Rods
in Water with Cadmium Plates Between Fuel Clusters*

FUEL CLUSTERS		CADMIUM PLATES ^a		CRITICAL SEPARATION BETWEEN FUEL CLUSTERS ^d (X_c , mm)	EXPERIMENT NUMBER
NUMBER IN ARRAY ^b	LENGTH x WIDTH 20.32-mm SQ. PITCH	THICKNESS (t_p , mm)	DISTANCE TO FUEL CLUSTERS ^c (G , mm)		
3	20 x 17	0.610 ± 0.025	6.45 ± 0.06	67.4 ± 0.6	036
3	20 x 17	0.610 ± 0.025	14.82 ± 0.70	76.0 ± 0.2	054
3	20 x 17	0.610 ± 0.025	40.42 ± 0.70	93.7 ± 0.3	037
3	20 x 17	0.291 ± 0.010	14.82 ± 0.70	77.8 ± 1.0	050
3	20 x 17	0.291 ± 0.010	40.42 ± 0.70	94.0 ± 0.3	051
3	20 x 17	0.901 ± 0.027	14.82 ± 0.70	75.4 ± 0.3	052
3	20 x 17	0.901 ± 0.027	40.42 ± 0.70	93.9 ± 0.3	053

*ERROR LIMITS SHOWN ARE ONE STANDARD DEVIATION.

^aPLATES ARE 356 mm WIDE BY 915 mm LONG.

^bCLUSTERS OF FUEL RODS ALIGNED IN A ROW.

^cPERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARY OF THE CENTER FUEL CLUSTER AND THE NEAR SURFACE OF THE PLATE.

^dPERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARIES OF THE FUEL CLUSTERS.

Fig. 1. Graphical arrangement of simulated shipping cask critical experiments.

it was found, of course, that the critical separation (i.e., mm of water) between the clusters decreased. Furthermore, it was found that the closer the poison plate was to the center fuel cluster the greater this decrease was. A typical set of data illustrating this point is presented in Table II, and was obtained from the measurements with the cadmium plates.

Benchmark Calculations for Subcritical Clusters of 2.35 wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons, B. M. Durst, S. R. Bierman, E. D. Clayton (BNWL)

A series of experiments¹ was performed recently at the Battelle-operated Critical Mass Laboratory to provide benchmark-type criticality data for shipping casks used in transporting LWR-type fuels. In conjunction with the experiments, a series of calculations was also performed for the purpose of correlating the experiments with theory

and for code and cross-section validation. This paper summarizes the results of these benchmark calculations.

Briefly, the experiments consisted of measuring the critical separation distance between three subcritical clusters of 2.35 wt% ^{235}U enriched UO_2 rods in water. The experiments were of two basic types: (a) those in which only water separated the fuel clusters, and (b) those in which selected neutron-absorbing materials were positioned in the water between the fuel clusters.

Due to the complex geometry of the experiments (described in Ref. 1), the KENO IV computer code² was used. Cross sections for the fuel rod region were generated using the EGGNIT code³ with FLANGE-ETOG-processed^{4,5} ENDF/B-IV data to generate the 17 epithermal groups and THERMOS (Ref. 6) with ENDF/B-III to generate the single-thermal-group cross sections. For the absorber plate region, thermal-group cross sections were generated using the TEMPEST calculation in EGGNIT.

Results of calculations for experiments where only water was between the clusters of rods are summarized

TABLE I

Results of Benchmark Calculations
for Subcritical Clusters of 2.35 wt% ^{235}U
Enriched UO_2 Rods in Water

Fuel Clusters		Critical Separation Between Fuel Clusters (mm) ^a	$K_{\text{eff}}^{\text{b,c}}$
Number in Array	Lattice Configuration (20.32-mm sq pitch) Number of Rods		
3 ^d	20 × 17	119.2 ± 0.4	1.022 ± 0.005
3 ^d	20 × 16	83.9 ± 0.5	1.019 ± 0.007
3 ^d	20 × 15	63.9 ± 0.6	1.017 ± 0.006
3 ^d	20 × 14	44.6 ± 1.0	1.016 ± 0.005
			Av 1.019
1 ^e	20 × 18.08	0.0	1.010 ± 0.003

^aSeparation distance as measured from outer cell boundary to outer cell boundary.

^bKENO IV calculations using 18-group averaged cross sections from FLANGE-ETOG processed ENDF/B-IV data. One sigma limits on Monte Carlo calculations.

^c103 batches, 500 neutrons per batch.

^dGeometry modeled discretely.

^eA smeared rods in water geometry used.

in Table I. The experiment geometry was closely modeled in KENO IV; the aluminum supports and the individual rods, including cladding and aluminum end spades, were described. The calculations of this discrete model indicated a conservative bias of ~2.5% in terms of k_{eff} .

At one of the experiment points, three 20 × 17 lattices were modeled using both a discrete fuel rod model and a homogenized rod in water geometry. Multiple cases were run, using different random number sequencing to obtain a meaningful statistical comparison. As indicated in Table II, the homogenized model agreed more closely with the experiment, being conservative by only about 1%, compared to the 2.5% bias obtained for the discrete case. Hence, there appears to be no particular advantage to modeling the core discretely.

To provide some data in one-dimensional geometry, the results from homogenized DTF-IV (Ref. 7) and KENO IV calculations were compared for a finite and infinite system geometry. Based on these analyses, the critical slab thickness for a slab of rods 60 lattice units long is 244 mm (1219 × 914.4 × 244 mm). An infinite slab of the same rods would have a 213-mm critical thickness, as predicted by DTF-IV.

Calculations were also performed for many of the experiments in which neutron-absorber plates were located in the water gap between the fuel clusters. Results of these calculations also indicated a ~1% conservative bias in terms of k_{eff} in using a homogenized fuel model to model the experiment system.

Summarizing, benchmark calculations were calculated for many different configurations of LWR-type fuel clusters and neutron absorbers. It is possible to very exactly model these systems in the computer code

TABLE II

Comparison of Discrete Model and Smeared Model for Three
20 × 17 Rod Lattices Isolated by 119.2-mm H_2O

Number of Batches	Neutrons per Batch	Number of Initial Batches Skipped to Obtain Convergence	$K_{\text{eff}}^{\text{a}}$
Smeared Geometry			
103	498	57	1.015 ± 0.004
103	499	17	1.007 ± 0.003
103	500	42	1.003 ± 0.004
103	501	22	1.012 ± 0.003
103	502	27	1.005 ± 0.004
Discrete Geometry			
103	498	37	1.024 ± 0.006
103	499	27	1.027 ± 0.006
103	500	11	1.022 ± 0.005
103	501	27	1.026 ± 0.006
103	502	67	1.028 ± 0.008

^aKENO IV calculations using 18-group averaged cross sections from FLANGE-ETOG processed ENDF/B-IV data. One sigma limits on Monte Carlo calculations.

KENO IV. However, there appears to be no decided advantage to doing so, as a homogenized model agrees much more closely with experiment. Using a homogenized model for the experiment system, an ~1% conservative bias in terms of k_{eff} is calculated.

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Measurement and Calculation of Reactor Parameters—I

Quantitative Consistency Testing of Thermal Benchmark Lattice Experiments, D. R. Finch, W. E. Graves (DuPont-SRL)

This paper sets forth a general method to demonstrate the quantitative consistency (or inconsistency) of results of thermal reactor lattice experiments. The method is of particular importance in selecting standard "benchmark" experiments for comparison testing of lattice analysis codes and neutron cross sections. Benchmark thermal lattice experiments are currently selected by consensus, which usually means the experiment is geometrically simple, well-documented, reasonably complete, and qualitatively consistent. The quantitative tests reported here were applied to 19 frequently used thermal benchmark lattice experiments, and only two of these lattices proved to be quantitatively consistent and generally useful as benchmarks.

For a uniform reactor lattice with ^{235}U and ^{238}U as the only fissionable materials, the two-energy-group, neutron-balance condition at critical (or steady-state subcritical) can be written

$$B_m^2 \left[\frac{D_1(\Sigma_{a2} + \Sigma_{r2-1} + D_2 B_m^2)}{\Sigma_{f2}\Sigma_{r1-2}} + \frac{D_2}{\Sigma_{f2}} \right] = (\bar{\nu}_2^{25} - 1) + (\bar{\nu}_1^{25} - 1)\delta^{25} - \bar{\alpha}_2^{25} - \bar{\alpha}_1^{25}\delta^{25} + (1 + \delta^{25}) \times [(\bar{\nu}_1^{28} - 1)\delta^{28} - C^* - \xi] \quad (1)$$

The experimentally determined activation parameters are

$$\rho^{28} = {}^{238}\text{U}(n,\gamma) \text{ Capture Ratio} = \frac{\text{Epithermal } {}^{238}\text{U Captures}}{\text{Thermal } {}^{238}\text{U Captures}}$$

$$\delta^{25} = {}^{235}\text{U Fission Ratio} = \frac{\text{Epithermal } {}^{235}\text{U Fissions}^a}{\text{Thermal } {}^{235}\text{U Fissions}}$$

$$\delta^{28} = {}^{238}\text{U Fast Fission Ratio} = \frac{\text{Total } {}^{238}\text{U Fissions}}{\text{Total } {}^{235}\text{U Fissions}}$$

$$C^* = \text{Modified Conversion Ratio} = \frac{\text{Total } {}^{238}\text{U Captures}}{\text{Total } {}^{235}\text{U Fissions}}$$

Other nonstandard symbols in Eq. (1) are defined as

$$\xi = \frac{\text{Total Captures in Materials Other than } {}^{235}\text{U and } {}^{238}\text{U}}{\text{Total } {}^{235}\text{U Fissions}}$$

$$\bar{\nu}_n = \frac{\nu \Sigma_{fn}}{\Sigma_{fn}} \quad \bar{\alpha}_n = \frac{\Sigma_{cn}}{\Sigma_{fn}}$$

Equation (1) expresses the material buckling (B_m^2) as a function of the integral activation parameters δ^{25} , δ^{28} , and C^* through cross-section-dependent terms calculable to high accuracy by state-of-the-art lattice physics codes. Equation (1) can be written in terms of ρ^{28} rather than C^* using

$$C^* = \frac{C^{28} (1 + \rho^{28})}{F^{25} (1 + \delta^{25})} \quad (2)$$

where C^{28} = thermal ^{238}U captures and F^{25} = thermal ^{235}U fissions. C^* from Eq. (2) is designated the "indirect" C^* ; and the experimental C^* , the "direct" C^* .

^aRelative to a cadmium cutoff energy defining the Group 1-Group 2 break.

Three quantitative consistency tests are defined from Eqs. (1) and (2). Assume that all cross-section-dependent terms in Eqs. (1) and (2) have been determined from calculation by a quality lattice analysis code. The three tests are:

1. The experimental B_m^2 is compared to the B_m^2 from Eq. (1) using experimental values of δ^{25} , δ^{28} , and C^* .
2. The experimental B_m^2 is compared to the B_m^2 from Eq. (1) using experimental values of δ^{25} and δ^{28} , and C^* from Eq. (2).
3. The C^{28}/F^{25} calculated by the lattice analysis code is compared to C^{28}/F^{25} from Eq. (2) using experimental values of ρ^{28} , δ^{25} , and C^* .

In each test, the consistency (or inconsistency) of the experimental parameters is determined from comparison of the indicated parameters and their associated uncertainties.

The three quantitative consistency tests were applied to 19 thermal lattice experiments that are used as benchmarks. The lattice calculations required were performed by the RAHABR¹ lattice analysis module of the JOSHUA System² using ENDF/B-IV cross sections.³

Results of these tests are shown in Table I. The uncertainties in Table I for quantities calculated from Eqs. (1) and (2) are minimum uncertainties as no cross-section uncertainties were included. The 1'st, 2'nd, and 5'th lattices in Table I show reasonably good quantitative consistency, but the remaining lattices show only poor to fair quantitative consistency. The first five lattices in Table I are critical experiments, and the remainder are subcritical experiments.

The general conclusions of this work are:

1. The quantitative consistency tests appear reliable and are easily applied. These tests, however, are not capable of identifying the source of an inconsistency.
2. Only two of the thermal benchmark lattice experiments studied are generally useful as benchmarks; these are Cross-Section Evaluation Working Group benchmarks TRX-1 and TRX-2.
3. Subcritical thermal lattice experiments show systematic inconsistencies; the most notable being the consistently high experimental material buckling.
4. Quantitative internal consistency tests should be applied by experimenters before reporting results of future lattice experiments.
5. Because so few thermal benchmark experiments can be shown to be quantitatively consistent, new thermal lattice experiments are required to supply benchmarks to test lattice analysis codes and neutron cross sections.

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TABLE I
Results of Quantitative Consistency Tests

Lattice Name	Type of Lattice	Pitch	Reference ^a	Material Buckling (B^2), m^{-2}			C^{23}/ρ^{23}		Consistency of Experiment
				Experimental	Equation 1 Direct C ^a	Equation 1 Indirect C ^a	RARAB ^b	Equation 2	
BAPL TRX-1 ^c	One-Region H ₂ O Critical	1.806 cm	4	57.00±1.00	54.66±1.25	57.51±1.10	0.370	0.379±0.005	Good
			5	57.00±1.00	55.74±1.25	57.62±1.10	0.370	0.376±0.005	
			6	57.00±1.00	55.84±1.55	58.06±1.46	0.370	0.377±0.005	
BAPL TRX-2 ^c	One-Region H ₂ O Critical	2.174 cm	4	54.69±0.36	53.05±1.06	55.20±0.96	0.367	0.376±0.005	Good
			5	54.69±0.36	53.39±0.64	55.25±0.96	0.367	0.374±0.003	
			6	54.69±0.36	53.96±1.32	55.66±1.28	0.367	0.374±0.005	
BAPL TRX-3 ^c	Two-Region H ₂ O Critical	1.441 cm	4,5,6				0.377	0.385±0.006	Not determined ^d
BAPL TRX-4 ^c	Two-Region H ₂ O Critical	2.882 cm	4,5,6				0.365	0.371±0.004	Not determined ^d
CRNL ZEEP	One-Region D ₂ O Critical	20.0 cm	7	6.95±0.06	6.89±0.05		0.670	0.670±0.006	Good ^e
MIT O.387 in. Rods	One-Region D ₂ O Sub-criticals	1.5 in.	8	9.55±0.10	8.05±0.26	8.06±0.10	0.508	0.507±0.004	Fair
		2.25 in.	8	12.44±0.11	11.74±0.17	11.83±0.06	0.500	0.503±0.005	
		5.0 in.	8	10.41±0.14	9.97±0.08	10.07±0.08	0.498	0.502±0.002	
MIT-1 ^c	One-Region D ₂ O Sub-critical	4.5 in.	9	8.48±0.10	6.22±0.53	8.08±0.15	0.665	0.709±0.016	Poor
			6	8.48±0.10	6.69±0.54	8.04±0.20	0.665	0.709±0.016	
MIT-2 ^c	One-Region D ₂ O Sub-critical	5.0 in.	9	8.65±0.10	7.36±0.41	8.34±0.11	0.663	0.701±0.015	Poor
			6	8.65±0.10	7.33±0.42	8.28±0.15	0.663	0.700±0.015	
MIT-3 ^c	One-Region D ₂ O Sub-critical	5.75 in.	9	8.15±0.08	7.66±0.28	7.94±0.06	0.661	0.675±0.013	Poor
			6	8.15±0.08	7.65±0.29	7.86±0.11	0.661	0.672±0.013	
SRL Mark SR Coaxial Tubes	One-Region D ₂ O Sub-criticals	5.5 in.	10	-4.92±0.36	-9.49±0.72	-9.13±0.46	0.576	0.580±0.008	Poor
		6.0 in.	10	0.42±0.36	-3.04±0.68	-2.61±0.68	0.572	0.578±0.011	
		7.0 in.	10	6.77±0.29	5.19±0.47	4.72±0.42	0.566	0.558±0.009	
		8.0 in.	10	9.20±0.22	7.98±0.38	8.56±0.43	0.563	0.571±0.011	
SRL Mark IS Coaxial Tubes	One-Region D ₂ O Sub-criticals	6.35 in.	10	2.78±0.20	1.18±0.66	1.56±0.85	0.450	0.453±0.011	Poor
		7.00 in.	10	7.70±0.30	6.19±0.54	6.15±0.45	0.446	0.446±0.008	
		8.08 in.	10	11.82±0.22	10.79±0.46	11.14±0.34	0.442	0.448±0.008	
		9.25 in.	10	13.18±0.22	12.61±0.34	12.93±0.33	0.439	0.446±0.008	

^a Several lattices have more than one set of experimental results reported in the literature.

^b Values calculated with the RARAB lattice analysis module of the JOSHUA System.^{1,2}

^c Indicates a lattice that is a Cross Section Evaluation Working Group (CSEWG) benchmark used in testing ENDF/B cross sections.

^d ρ^{23} was not reported for these experiments. Hardy¹¹ has shown the analysis of these lattices to be very dependent on leakage effects.

^e This lattice did not have an experimentally determined ρ^{23} or k^{23} .

Physics of Thermal Reactors

Evaluation of ENDF/B-IV Data for Tightly Packed UO_2 -Water Lattices, W. O. Olson, F. J. Wheeler (EG&G Idaho)

The ENDF/B-IV data set has been evaluated for use in highly undermoderated reactor cores. The critical experiments analyzed are those originally used in the design of the Power Burst Facility (PBF) reactor core.

Fig. 1. Calculated multiplication vs H/U^{26} ratio. (Numbers within graph refer to cases in Table II.)

These were performed by Phillips Petroleum¹ and G.E.² in 1964 and 1967. The lattices have nonmoderator-to-moderator ratios of 1.9 and 2.3, and H/U ratios of 2.7 and 2.3, respectively. The fuel rods are UO_2 pellets clad in stainless steel, surrounded by thin aluminum sleeves to simulate the ZrO_2 insulators in the PBF core. These rods are described in Table I. The rods are placed in a square mesh of 15.656-mm (0.616-in.) and 15.189-mm (0.598-in.) pitch for the 1.9 and 2.3 ratio cells, respectively. The experimental arrays are approximately right circular cylinders, with radial water reflectors, containing 900 to 1568 pins. These vary from simple clean arrays to configurations containing mocked-up test areas, control rods, voids, etc. The water height was adjusted to achieve criticality.

ENDF/B data are processed into a 68-group library by ETOP³ and a 101-point thermal library by FLANGE-II.⁴ A 90-group library is formed using PHROG⁵ for the fast and INCITE⁶ for the thermal groups. Both codes were run in the B-1 approximation, with the only geometrical dependence being a buckling term. RABBLE⁷ was used for the resonance treatment.

90-group pin-cell calculations were made with SCAMP (an internal EG&G code) in the S_0 approximation, using an equivalent cylindrical cell with semi-isotropic return at the outer boundary. This yielded homogenized fuel cell constants. Cross sections were collapsed to 33- and

TABLE I
Fuel Rod Characteristics

	Phillips	G.E.
UO_2 Density	10.5 g/cm ³	10.08 g/cm ³
Enrichment	4.8 wt%	4.8 wt%
Fuel Pellet o.d.	10.668 mm (.42 in.)	10.668 mm (.42 in.)
S.S. Cladding Thickness	0.508 mm (.020 in.)	0.508 mm (.020 in.)
Cladding o.d.	11.836 mm (.466 in.)	11.836 mm (.466 in.)
Active Fuel Length	0.97282 m (38.3 in.)	0.9144 m (36.0 in.)
Al Sleeve o.d.	14.288 mm (.5625 in.)	14.288 mm (.5625 in.)
Sleeve Thickness	0.889 mm (.035 in.)	0.889 mm (.035 in.)

4-group sets. Collapsing to 4 groups was very sensitive to detail, requiring accurate geometrical models and leakage terms.

Criticality calculations were made with the CORA⁸ diffusion code using 4 groups in the automatic buckling iteration mode. In this mode the code makes the calculation in one direction, using a perpendicular buckling term calculated from the leakage parallel to the remaining directions, with iteration between the directions continuing until convergence is achieved.

The results of these calculations are tabulated in Table II. The calculated eigenvalues are consistently below the experimental value. Also included in the table are values calculated for certain other configurations; namely, calculations for the PBF core and the TRX benchmarks. The apparent trend is the depression of the calculated eigenvalue with decreasing H/U^{26} ratio as shown in Fig. 1. This is in agreement with previous results⁹ which indicated that the epithermal capture of ^{238}U is overpredicted in integral experiments using the ENDF/B-IV data set.

In summary, evaluation of the ENDF/B-IV data set has been extended to very small H/U ratio cores with the most notable discrepancy being a depression of the calculated eigenvalue with decreased H/U^{26} ratio.

TABLE II
Eigenvalues and Moderator/Fuel Ratios

Case	Description	Number of Tests	H/U ²⁵	H/U ²⁸	k-eff(C/E)
1	Phillips 1.9 Ratio Criticals	11	56	2.8	0.9875±.0016
2	Phillips 2.3 Ratio Criticals	7	47	2.4	0.9846±.0013
3	G.E. 1.9 Ratio Criticals	7	58	2.9	0.9876±.0021
4	PBF Core	1	64	15.0	1.0006
5	TRX-1 (Metal)	1	251	3.3	0.9827
6	TRX-2 (Metal)	1	429	5.7	0.9893

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Gaseous Core Reactors and Nuclear-Pumped Lasers

A U(95%)F₆-Helium Circulating Gas Critical Assembly, D. M. Barton, G. A. Jarvis, J. F. Davis III (LASL)

Los Alamos Scientific Laboratory (LASL) and United Technologies Research Center (UTRC-East Hartford, Connecticut) are engaged in a NASA-funded project to study the applications of gas core reactor concepts to advanced power reactor designs.¹⁻³ The experimental part of the program is being conducted using a Be-reflected core consisting of solid and gaseous uranium fuel. The materials are assembled on a critical assembly machine in Kiva I at the Pajarito Critical Assembly Laboratory at Los Alamos, New Mexico. Critical experiments using flowing UF₆-He mixtures are described in this paper.

The critical assembly consists of a 50-cm-thick Be reflector containing 18 boron control drums mounted on an elevated platform about 2 m above the floor of the assembly building. A platen holding the core materials and a 50-cm-thick Be plug is raised up into a 1-m i.d. × 1.5-m cavity in the elevated reflector sector. The core geometry consists of 7.7 kg of solid uranium-graphite Rover fuel elements placed in a single ring on the outside of a 1-m-high × 0.55-m-i.d. × 0.89-m-o.d. Be cylinder. The U(95%)F₆ + He gas mixture is contained inside a double-walled Al canister centered in the Be cylinder. An adjacent gas manifold is connected to the canister with flexible hoses to allow the core to be raised and lowered on a hydraulic ram mounted in the floor.

Figure 1 is a schematic drawing of the gas handling manifold. There are two loops associated with the system. One contains the UF₆ and He gas mixture and the other has N₂ gas. Nitrogen is used to heat the UF₆ pipes in the flexible leads and to heat the walls of the canister to about 90°C. This is necessary to prevent the UF₆ gas from condensing on the walls of the pipes and canister.

During shutdown conditions the UF₆ is stored in a bottle inside the oven at room temperature and pressures as a solid. To charge the canister for a critical operation, the UF₆ lines are filled with He at 600 Torr pressure, the UF₆ supply bottle is valved off and the He gas is circulated for one or two hours in the UF₆ loop to preheat the canister and lines. The UF₆ in the bottle is then heated to the gas phase by turning the oven to 90°C. The He gas stream is next valved through the UF₆ bottle and carries the UF₆ out of the bottle and around the loop.

UF₆ can be removed from the He by cooling the bottom of the supply bottle while the oven keeps the valves open. After all of the UF₆ is condensed from the mixture the bottle is valved off and allowed to cool. Eighty-seven percent of the mixture is in the canister at one time. The amount of UF₆ transferred from the bottle to the manifold is determined by removing the supply bottle and weighing it. This procedure provides the necessary inventory control for experimental measurements and fissionable material accounting.

The axial and radial flux distributions were measured by activation of Al-U alloy wires. The radial flux is essentially flat but the axial fission distribution falls off

Fig. 1. Schematic diagram of uranium hexafluoride Mark II flow loop.

4% at the top end of the canister and 10% at the bottom. Neutron streaming through holes in the bottom Be plug where the pipes go through accounts for the lower flux at the bottom end of the canister.

The gas conditions during critical operations were as follows:

1. 660 Torr total pressure. 640 Torr He and 20 Torr U(95%)F₆.

2. The temperature of the gas was 60°C.

3. The mass flow rate of UF₆ in the mixture was 1.21 g/s.

The core loading consisted of 7.70 kg of 93% ²³⁵U uranium-graphite Rover fuel elements in the outside ring and 28.6 g of U(95%)F₆ gas in the canister.

The reactivity worth of the UF₆ gas was determined to be 2.44 ϵ /g, corresponding to a worth of 3.62 ϵ /g for the (95%)U in the gas.

A Be-reflected core of solid uranium-graphite fuel rods and UF₆ gas has been operated at low power with an He-UF₆ gas flowing at 1.21 g/s. The reactivity worth of the U(95%)F₆ gas was measured to be 2.44 ϵ /g at the center of the assembly.

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Gas Core Reactors: Neutronics, SNM Proliferation, and Resource Utilization, Thomas P. McLaughlin, Patrick D. Soran, Gordon E. Hansen (LASL)

Described herein are the results of criticality calculations for single-, four-, six-, and seven-cell gas core models. These beryllium-reflected and -moderated UF₆ systems all have potential merit relative to both the proliferation of weapons-grade materials and the conservative/efficient utilization of thorium and uranium resources. A brief investigation of the consequences of denaturing the ²³³U with ²³⁸U indicates neutronic feasibility for ²³⁸U:²³³U ratios up to 20:1; however, the practicality and advantages of this approach are questionable.

Two operating modes are considered for the gas core concept: (a) a UF₆ gas in helium at a partial pressure of 1% and a gas temperature of 1000 K with the entire core gas flowing through a heat exchanger loop, and (b) a uranium plasma operating near 5000 K with the fission energy removed by optical radiation to an inert working fluid. Neutronically, the nonplasma system is somewhat superior in that neither a buffer (confinement) gas is required nor is a separate working fluid. Both of these occupy space which tends to enlarge the plasma system relative to the nonplasma, increasing neutron leakage and thus requiring a larger fissile inventory, all else constant.

Neutronics analyses employed the one-, and two-dimensional S_N codes, ONETRAN and TWOTRAN, and the continuous-energy Monte Carlo code MCNK.¹ Reactor concepts investigated include: single-, four-, six-, and seven-fuel-cell systems, both plasma and nonplasma. Two fuel cell sizes were considered: 0.5- and 1.0-m-diam H/D ratios of about 4/1. Surrounding the fuel cells was beryllium at 1000 K, followed radially and axially by breeding (blanket) regions of thorium-bearing molten salt.

The S_N analyses were all performed in R or R-Z cylindrical geometry while the Monte Carlo calculations treated the system in three dimensions. For the four- and seven-cell systems, a central fuel cell was surrounded by either 3 or 6 equally spaced identical cells. Based on preliminary Monte Carlo results, a geometry refinement, which substantially improved the breeding ratio, was to scallop the outer edge of the beryllium matrix surrounding the fuel cells. This scallop design is illustrated in radial cross section in Fig. 1. Monte Carlo calculations for each of the multicell geometries were used to normalize the results from the S_N calculations. The latter were employed to investigate the effect of varying system dimensions; e.g., beryllium thicknesses and fuel cell separation distances.

Critical inventory and breeding ratio for a single-cell (1.0-m-diam), nonplasma reactor would be about 20-kg ²³³U and 1.12, respectively. Depending on operating conditions, i.e., He/UF₆ pressure and flow rate, this concept could produce up to ~300 MW(th). A four-cell reactor is not viable since dimensions cannot be adjusted to yield a breeding ratio greater than unity. Results for six- and seven-cell (0.5-m-diam) plasma concepts are shown in Fig. 2. These systems could deliver up to ~600 MW(th) and, if 1.0-m-diam cells were employed, up to ~2500 MW(th).

Denaturing the ²³³U feed with ²³⁸U indicated that a uranium feed composition of 20:1 (²³⁸U:²³³U) results in only a doubling of the critical ²³³U inventory for single-cell concepts. However, the total uranium inventory is 40 times greater than for the pure ²³³U core, thus making the practicality of diversion of this ²³⁸U/²³³U for weapons purposes most unlikely. The significant feature of a gas core concept from a proliferation standpoint lies in the relatively low amount of fissile material in the system (core plus blanket) at any time.

Fig. 1. Seven-cell scallop reactor concept.

Fig. 2. Critical mass and breeding ratio vs Be thickness.

The most important advantage of these gas core concepts, however, may lie in the fact that they can be designed as breeders but with minimal initial fissile feed requirements. This point is significant for two reasons: (a) the rate at which these gas core reactors could be brought on line would not be dependent on the rate of fuel being discharged from LWRs as would be the case for LMFRs; and (b) the estimated easily recoverable U.S. uranium resources will probably be sufficient to fuel 500 to 1000 LWRs of 1000 MW(e) each, while this same amount of uranium could be used to fuel approximately

100 times as many gas core reactors of equal power output.

In summary, the gas core reactor concept is attractive neutronically, reduces SNM proliferation potential, and offers significantly improved resource utilization over current reactor designs.

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Reactor Theory and Applications

Transport Consistent Diffusion Theory Applications to Fuel Storage Pool Criticality, H. A. Dayem (LASL), O. J. Sheaks (U of Md)

A computer algorithm has been formulated from an analytical method which transforms without approximation the general form of the two-dimensional (2-D) transport equation into a diffusion-like differential equation. The computational economy of this procedure is demonstrated in the application of the technique to the parametric calculations required in determining fuel storage pool K_e -eigenvalues as a function of fuel assembly spacing.

The development of the transport-diffusion algorithm consists of mathematical operations on the transport equation to obtain a relationship between the net current and flux gradient, formally designated as the transport diffusion coefficient. The procedure is similar to that of Pomraning¹ with the noteworthy differences that (a) no analytical approximations are required, and (b) the analytical formulation is constrained to yield a computational algorithm that is consistent with those employed in 2-D discrete ordinates and diffusion theory codes; e.g., TWOTRAN (Ref. 2) and 2-DB (Ref. 3).

To derive the analytical formulation of the transport consistent diffusion coefficient the general form of the multigroup 2-D transport equation² is operated on with the integral operator, $\int \Omega d\Omega$, following an expansion of the angular group flux and scattering cross section in terms of spherical harmonics.

By selecting appropriate definitions and term arrangement, the vector equation can be cast in the form

$$\mathbf{J}_g = -D_g \nabla \phi_g, \quad (1)$$

where the 2-D group current and flux gradient vectors have their standard definitions, and D_g is a diagonal matrix with components expressed in terms of flux and cross-section moments.

$$D_{11} = \frac{\frac{1}{3} \frac{\partial}{\partial x} (\Phi_{0g}^0 + 2\Phi_{2g}^0) - \left(\frac{1}{\sqrt{3}} \frac{\partial}{\partial y} \Phi_{2g}^1 \right) - \sum_{h \neq g} \sigma_{s1h \rightarrow g} \Phi_{1h}^0 + Q_{1g}^0}{(\sigma_g - \sigma_{s1,g}) \frac{\partial}{\partial x} \Phi_{0g}^0} \quad (2a)$$

$$D_{22} = \frac{\frac{1}{3} \frac{\partial}{\partial y} (\Phi_{0g}^0 + \Phi_{2g}^0 + \sqrt{3} \Phi_{2g}^1) - \left(\frac{1}{\sqrt{3}} \frac{\partial}{\partial x} \Phi_{2g}^1 \right) - \sum_{h \neq g} \sigma_{s1hg} \Phi_{1h}^1 + Q_{1g}^1}{(\sigma_g - \sigma_{s1,g}) \frac{\partial}{\partial y} \Phi_{0g}^0} \quad (2b)$$

It is noted that the integrals evaluated to arrive at Eq. (2) result in self-truncation of the flux and cross-section moments, thus making Eq. (1) theoretically consistent with the transport equation. In addition, although the components of the diffusion coefficient in the x and y directions are different, the formulation is such that it is compatible with the numerical algorithms existing in most 2-D diffusion theory codes.

Fig. 1. K_e as a function of assembly spacing and calculational method.

The usefulness of Eq. (2) is that it embodies the effects of an angular leakage algorithm in a form compatible with the diffusion equation for the total flux. In practice, this formulation will provide economic benefits for parametric studies in which the transport diffusion coefficient is insensitive to parameter change. The procedure is to perform a base-case transport calculation to obtain the flux moments for use in Eq. (2). The spatially varying transport diffusion coefficients can then be used to perform parametric diffusion calculations. The results should be improved accuracy over the standard diffusion theory calculations and a considerable savings in computational time.

To illustrate this technique, a PWR fuel storage pool calculation was performed to determine the K_e -eigenvalue as a function of assembly spacing. A transport calculation was performed for the base-case separation spacing of 12 cm to furnish data to determine the transport diffusion coefficient. This transport diffusion coefficient was utilized in subsequent diffusion calculations in which the

Pitfalls and Quality Assurance Practices In Criticality Safety Analysis

Session Introduction, C. L. Brown (BNWL)

Operations that involve the handling, processing, and storage of fissionable materials have an element of risk for accidental criticality. This risk is kept acceptably low by applying properly established limits and controls on equipment design and procedural operations to assure subcriticality. These limits and controls are established by criticality safety specialists using a procedure commonly referred to as criticality safety analysis.

When a criticality safety analysis is completed, those who are reviewing and applying the results and those who are to perform the operations need assurance that the analysis is correct. If not correct, a potential criticality hazard could exist.

Confidence that an analysis is correct comes from various assurances: application of sound quality-control procedures, second-party reviews, and even the reputation for careful, professional work on the part of those who perform and review the analysis. The level of importance (and formality) placed on this assurance usually depends on the amount of fissionable material involved and the complexity of the equipment and process being analyzed.

Certain parts of a criticality safety analysis must be correct and complete or the analysis is null and void. These essential parts are as follows:

1. correctness and completeness of system description
2. correctness, validity, and completeness of the set of credible errors and conditions assumed
3. validity of codes selected for the analysis
4. correctness of cross sections used in analysis
5. correctness of code options selected
6. correctness of code input
7. adequacy of benchmarks selected to validate codes used (extrapolation, interpolation; did validations cover system in question, etc.)
8. correctness of the benchmarks (are experiments properly described; are experimental biases needed, known, etc.)
9. correctness of interpretation of code results
10. correctness of analysis conclusions
11. correctness of final and as-built equipment system
12. correctness of startup and operating procedures.

This session will address these topics. Emphasis will be on actual experiences in recognizing pitfalls, and how

these pitfalls can be avoided through quality assurance and quality control procedures.

Ensuring the Validity of Calculated Subcritical Limits, H. K. Clark (DuPont-SRL)

Great care is taken at the Savannah River Laboratory (SRL) and Plant (SRP) to ensure that conditions calculated to be subcritical will actually be subcritical. Careful attention is given to ANSI N16.9-1975, *Validation of Calculational Methods for Nuclear Criticality Safety*. There are two principal areas of concern. The first is that the method selected be described unambiguously and that it be used in the intended manner, free of input error and of an improper selection of options. The second is that the method be correlated with pertinent experimental data to establish bias and that allowance for uncertainty in the bias be adequate to ensure subcriticality.

Methods in use at SRL range from a simple 12-group B_0 calculation (MGBS),¹ with cylindrical cell capability, which generates two-group parameters for use in one-dimensional diffusion theory calculations performed by an analytical code (TGAN)¹ to Monte Carlo calculations as embodied in KENO-IV,² S_N calculations as embodied in ANISN,³ and B_1 and integral transport cylindrical cell calculations as embodied in GLASS.⁴ All these codes are modules in the JOSHUA system.⁴

The JOSHUA system handles data (e.g., input parameters, cross sections, intermediate data, output parameters, etc.) in named records, which may be written or read by the modules or by the user at a terminal through a template. A template is simply a means of displaying the data with labels to identify items. Such labels tend to help the user avoid errors in supplying input. Modules in the JOSHUA system may be executed in sequence during the course of a job. A driver module, KOKO, has been written to prepare specialized input for the different modules used for criticality safety calculations and to execute them in specified sequences.

The calculation of macroscopic cross sections is not a part of the JOSHUA versions of KENO-IV and ANISN. For Hansen-Roach cross sections⁵ it is performed in a module HRXN which accepts composition data in a wide variety of forms. To provide a standard method for computing the densities of aqueous solutions of U and Pu, apparent densities of aqueous solutions of U and Pu, apparent densities of UO_2F_2 in water solutions and of UO_2 and PuO_2 in nitric acid solutions are incorporated into the module so that only U, Pu, and NO_3 concentrations have to be specified. Mixture densities and compositions by concentration and weight fraction of components are printed so that errors may readily be recognized. In HRXN the macroscopic potential scattering cross section

TABLE I
Fuel Storage Pool Criticality—Comparison of K_e and Computer Running Times

Method of Calculation	$\Delta = 0$			$\Delta = -1$			$\Delta = -2$			$\Delta = -3$		
	K_e	% Error	Time (s)	K_e	% Error	Time (s)	K_e	% Error	Time (s)	K_e	% Error	Time (s)
Diffusion	0.8338	3.04	76	0.8625	1.28	66	0.9026	-0.24	59	0.9558	-1.29	53
Transport	0.8092	---	1384	0.8516	---	1212	0.9048	---	941	0.9683	---	736
Modified diffusion	0.8016	-0.94	77	0.8436	-0.94	78	0.8969	-0.87	72	0.9609	-0.76	82

lattice spacing was varied by $\Delta = -2$ -cm intervals to a minimum separation of 6 cm. The eigenvalues obtained using the modified diffusion theory were compared against those obtained from TWOTRAN calculations and from ordinary diffusion theory. Figure 1 illustrates the improvement in accuracy. The savings in computer time is cataloged in Table I. It is seen that the improvement in calculational accuracy and the savings in computational time are significant. Other applications have been investigated with similar success for cases that did not involve severe flux gradients.

In summary, the computational algorithm developed has the potential for reducing computational costs and improving the accuracy of calculations for which ordinary

diffusion theory is inadequate. The coding of the algorithm was performed to accommodate utilization of most 2-D transport and diffusion theory codes.

1. G. C. POMRANING, "Reduction of Transport Theory to Multigroup Diffusion Theory," *J. Nucl. Energy*, **18**, 497 (1964).
2. LATHROP and BRINKLEY, "TWOTRAN-II: An Interfaced Exportable Version of the TWOTRAN Code for Two-Dimensional Transport," LA-4848-MS, p. 2 (1973).
3. LITTLE and HARDIE, "2DB User's Manual—Revision 1," BNWL-831 REVI (1969).

is computed from data contained in cross-section records and resonance cross sections are interpolated from cross-section records as a function of potential scattering per atom of absorber.

Calculational methods for nuclear criticality safety should be standardized as much as possible since bias established with one selection of options may differ significantly from that obtained with another selection. This standardization is provided by subroutines in KOKO, which write input records for other modules in a standardized manner. For example, in the case of ANISN particular sets of quadrature weights and cosines are selected merely by specifying S_4 , S_8 , or S_{16} . The mesh is selected by a scheme which generally assigns more points in regions of rapidly varying flux. A standard selection of integer and floating point parameters is made. The options selected, by default, when no data are entered, are usually the standard method. The input through KOKO, besides being standardized, is greatly simplified, thereby tending to minimize input errors.

To establish the bias of a method, reference is generally made to the original source of the data; and all available data are included in the correlation. Including all the data tends to provide an estimate of the uncertainty beyond that provided by the experimenters for individual data points since calculations (other than Monte Carlo) should exhibit smooth trends; abrupt changes are presumably caused by experimental errors. The bias established by the correlation is carefully examined for trends, particularly where the conditions being evaluated are outside the range of experimental data. In such cases, the method is often supplemented by other methods, and reasons for trends are sought. An adequate margin of subcriticality is selected. The margin is, of course, arbitrary and is chosen to be ample but not excessive. The smaller the margin, the greater the effort required to justify its adequacy. However, the increase in limits associated with, say, a 1 to 2% increase in estimated reactivity is generally not worth the effort required to justify it. Greater increases in limits are likely to result from restricting an additional parameter such as concentration in addition to mass.

The final part of any criticality safety evaluation is the documentation of calculations and the results of the evaluation. This is a valuable exercise. Often the need for additional calculations is revealed as the writing proceeds. It is important to be clear, complete, and convincing.

1. H. K. CLARK, "Computer Codes for Nuclear Criticality Calculations," DP-1121, E. I. du Pont de Nemours and Company, Savannah River Lab. (1967).
2. L. M. PETRIE and N. F. CROSS, "KENO-IV—An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Lab. (Nov. 1975).
3. W. W. ENGLE, Jr., "A User's Manual for ANISN," K-1693, Oak Ridge Gaseous Diffusion Plant, Oak Ridge, TN (1967).
4. H. C. HONECK, "The JOSHUA System," DP-1380, E. I. du Pont de Nemours and Company, Savannah River Lab. (1975).
5. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos Scientific Lab. (1961).

The Role of Computers in Quality Assurance in the LLL Criticality Safety Program, B. L. Koponen (LLL)

The goal of any quality assurance program in criticality safety is to ensure that limits have been accurately established and will contribute to a safe operation. Computers are relied on to a great extent in most criticality safety programs and consequently are a key element in quality assurance programs in criticality safety. Computers are used at the Lawrence Livermore Laboratory (LLL) not only directly in criticality safety analysis but also in management of numerical and bibliographic criticality data that are needed for the validation of computational methods.

Computational quality control can perhaps be divided into the two categories listed below:

A. Assurance of Input Correctness

"Human Engineering" aspects of code design to minimize the chances of errors being made.

Built-in checks by the code to help find the errors that are made (mass and volume calculations, pictures, output plots, etc.).

Independent checking of code input by other personnel.

B. Code Validation

Checks that show the code is performing the intended operations correctly.

Benchmark calculations of critical experiments.

Comparisons with other codes or techniques; e.g., Monte Carlo vs discrete ordinates.

The author has found the first item of category (A) to be especially helpful in minimizing the chances of code input errors being made. The LLL philosophy is that input errors will be minimized if the number of parameters the user must provide is minimized and if the format of the input parameters is in English words or mnemonics where possible. For example, we have prepared a user-interactive version of the ANISN¹ neutron transport code—SAN. The SAN code requires the user to specify geometry type, material compositions, and regional descriptions. Other parameters such as angular quadrature constants are internal to the SAN code or appended to the cross-section library. A sample of input for a one-region system is given below.

```
EXAMPLE ONE-REGION PROBLEM
SPHERE REFLECTION VACUUM
1 U235 0 3 .0484 LAST
5.75 30 1 LAST
END
```

Since SAN is limited to a K-calculation we still maintain the full-size version of ANISN for searches, source problems, etc.; however, for perhaps over 95% of our transport code criticality calculations, the SAN code is less prone to input error, speeds up problem preparation time, and uses less computer time than ANISN.

A similar approach has been taken with our criticality Monte Carlo code—MORSE.² In MORSE the basic Monte Carlo parameters have been set internally to the values that are needed for most of our problem solving. When necessary these values can be overridden by user input. Where possible, such as in material input specifications, the MORSE input format is compatible with SAN. In many cases it is possible to prepare a complete problem input without having to refer to a users manual.

The second item in category (A) has also been very helpful in minimizing input errors in both SAN and MORSE. The visual displays are much more quickly evaluated than long lists of numerical output and the volume and mass calculations often provide evidence of input errors.

In category (B) we are preparing a computerized data base of bibliographic and numerical critical experiment data. The bibliographic data help us to identify relevant experiments to enter into the numerical data base. The numerical data base contains descriptions of a variety of critical measurements and has provisions for feedback of results of calculations into the data base.

The MASTER CONTROL³ program is used to manage the data base. MASTER CONTROL will perform many useful manipulations of the data such as ordering according to core composition, reflector material, Keff, etc., and will perform numerical calculations such as computing averages of results, differences of results by various methods, or will punch the data into formats required by other codes such as ANISN or MORSE.

1. W. W. ENGLE, "A User's Manual for ANISN," K-1693, Oak Ridge Gaseous Diffusion Plant, Union Carbide Corporation Nuclear Division (1967).
2. E. A. STRAKER, P. N. STEVENS, D. C. IRVING, and V. R. CAIN, "The MORSE Code—A Multigroup Neutron and Gamma-Ray Monte Carlo Transport Code," ORNL-4585, Oak Ridge National Lab. (1970).
3. V. E. HAMPEL and J. R. WADE, "MASTER CONTROL—A Unifying Free Form Data Storage and Data Retrieval System for Dissimilar Data Bases," UCRL-71686, 1968, *Proc. Am. Soc. Inform. Sci.*, 6, 159 (1969).

Quality Assurance for Computer Programs Used in Criticality Safety Analysis, G. R. Handley (UCC-ND)

Computer calculations are used to determine operating limits and design parameters in the nuclear industry. Often these calculations are a necessity and must stand on their own merits since no relevant experimental data are available. An incorrect calculation could result in either conservatism, which imposes unneeded, costly, and cumbersome restrictions, or a nuclear criticality accident, which is, of course, the "significant" failure in terms of consequences. Incorrect answers, which must later be corrected, cause a loss of credibility and effectiveness in any nuclear safety program. Therefore, a quality assurance (QA) plan is needed for computer calculations to provide confidence that the risk of these failures has been reduced to an acceptably low level. The plan should cover more than just getting a calculation correct, for instance: where did the input data come from, and what will be done with the results of their manipulation? One should examine each step in the calculational process from problem description to implementation of conclusions. Little is accomplished by solving an incorrectly stated problem, or by misusing the correct answer.

Table I presents a QA plan used by the Criticality Safety Department in the Y-12 and Oak Ridge Gaseous Diffusion Plants where KENO and ANISN are used with Hansen-Roach 16-energy-group cross sections. Identified are the QA requirements, recognized modes of requirement failure, one or more actions needed to prevent or reduce the probability of each failure, assign-

ment of responsibility to personnel, and the required documentation. The QA plan is assessed periodically by the Technical Division's QA Coordinator, the Criticality Safety Department Head, and the Nuclear Engineering Applications Department Head. Additional assessments are initiated when problems or failures occur. Thus, the plan improves with use.

We have had no known quality failures, but we have had problems. In the hope that they will be instructive, typical problems that have occurred for each requirement identified in Table I, but were detected by QA, are given below:

Assemble Information

1. A written request for analysis specified an operation involving "20 grams" of uranium when "20 kilograms" was intended.
2. A container with a 7-in. diam was analyzed when the designer had changed it to 7.5 in.

Choose Contingencies

The 20-in. bucket birdcages with multipart loadings were analyzed as if flooding were a possibility even though the bucket had a lid, had holes in the side near the bottom, and was elevated above the floor to a height where water could not enter.

Design Mockup

1. The approximation of a storage facility was non-conservative when walls touching the outer arrays were eliminated by assuming there was an infinite array of arrays in two dimensions.
2. Various enrichments of uranium were calculated before the bias of the cross sections was established by validation.

Prepare Input

1. Numbers have been miscopied.
2. A pipe intersection was calculated after starting the neutrons in a flat distribution when they should have all been started at the intersection.

Compute

1. The KENO program did not properly compute a new program option and yielded consistently low answers.
2. Errors were entered into the cross-section library.
3. The bias in the cross sections was found to be energy dependent for low uranium enrichment.
4. The program tape failed due to age.
5. A change was authorized to KENO which resulted in consistently low answers for a particular geometry which was thought to be unrelated to the change.
6. When running KENO on a new computer, all the direction cosines were found to be zero.

Interpret Results

1. A series of ANISN cases yielded consistently non-conservative results due to a coarse-mesh interval which gave the appearance of convergence.
2. KENO yielded a low answer when nonrepresentative sampling kept the largest unit in an array from being seen.

TABLE I
Quality Assurance Plan for Computer Programs Used in Criticality Safety Analysis

REQUIREMENT IDENTIFICATION	FAILURE MODES	ACTIONS NECESSARY FOR CONFIDENCE	RESPONSIBILITY	DOCUMENTATION
Assemble Information	Incorrect information submitted to analyst Incorrect information gathered by analyst Misunderstanding between analyst and others involved in design	Procedurally require two check signatures on analysis request Second independent check Written agreement between analyst and others involved	Head of Criticality Safety Department Second Criticality Safety analyst, designer, or operations Criticality Safety analyst	UCN Form Number 3446 Initial and date worksheet All parties sign and date written communications
Choose the Contingencies	Analyst considers credible events to be incredible, or incredible events to be credible	Independent review	Head of Criticality Safety Department (with input from designer and/or operations)	UCN Form Number 3446
Design Mockup	Approximations to reality are nonconservative or too conservative Range of validated applicability is exceeded	Independent review Independent review	Second Criticality Safety analyst ^a Second Criticality Safety analyst ^a	Initial and date worksheet Initial and date worksheet
Prepare Input Data	Data errors Improper selection of computer program options	Independent check Independent review	Second Criticality Safety analyst ^a Second Criticality Safety analyst ^a	Initial and date worksheet Initial and date worksheet
Compute	Error in computer program Error in cross sections Bias in cross sections Tape or disk ceases to function properly Authorized changes to computer program or cross sections Unauthorized changes to computer program or cross sections System changes to current computer New computer Machine error	Independent check of all program options Independent check of the numbers and validation to establish bias Validate to establish cross section bias No action needed since program will not run Revalidate cross sections and program options Procedurally require concurrence of Head of Criticality Safety Department Routinely recheck program options and cross section bias Revalidate cross sections and program options Never base a decision on only one calculation	Head of Nuclear Engineering Applications Department Head of Nuclear Engineering Applications Department Head of Criticality Safety Department Does not apply Head of Nuclear Engineering Applications Department Head of Nuclear Engineering Applications Department Head of Criticality Safety Department Head of Nuclear Engineering Applications Department Criticality Safety analyst	Letter or report Letter or report Report Does not apply Letter Letter Letter Letter Report
Interpret Results	Misread answer Lack of convergence Nonrepresentative sampling in Monte Carlo methods Improper validation Range of validated applicability is exceeded	Independent check Independent check of input data, output, and convergence criteria Independent check of input and output Independent review Independent check	Second Criticality Safety analyst Second Criticality Safety analyst ^a Second Criticality Safety analyst ^a Head of Nuclear Engineering Applications Department Second Criticality Safety analyst ^a	Initial and date output Initial and date output Initial and date output Letter Initial and date output
Communicate Conclusions	Facts are verbally stated incorrectly or misunderstood Written conclusions contain errors Typographical errors Written conclusions are misinterpreted	Put communications in writing Independent review Proofread Author periodically checks implementation of conclusions	Criticality Safety analyst Head of Criticality Safety Department ^a Criticality Safety analyst Criticality Safety analyst	Report or letter Initial and date final draft Initial and date prepared copy Monthly Progress Reports

^a Also performed by Head of Nuclear Engineering Applications Department for nonroutine problems.

3. A validation yielded k_{eff} values which were significantly lower than a similar validation done previously.

4. Calculational studies have been performed in which the neutron energy spectra were significantly different from those of the validated regime.

Communicate Conclusions

1. Numbers have been typographically transposed.

2. An operations foreman chose to extend a written conclusion by referring to the data and analysis.

Adoption of a formal QA plan is new to this application even though conscientious analysts have long practiced its requirements informally. However, there can be no doubt of its need when one reviews the list of possible failure modes and recounts the many actual problems any analyst involved in computing may relate. It is not intended that this particular plan be adopted by all organizations. It works well for us, but it would be difficult or impossible for many others. They should work out their own plan which protects against the same possible failures in a manner appropriate to their organization, number of analysts, relative skills, location, facilities, etc. Any plan merely imposes the same double-contingency principle on the criticality safety analyst that has long been imposed on operations personnel; and rightly so, since the mistakes of either could have the same consequence.

Quality Assurance Plan for Criticality Safety Calculations at Rocky Flats, Deanne Pecora (Rockwell Int)

The purpose of the Nuclear Safety Group at Rocky Flats is the prevention of an accidental, self-sustaining nuclear chain reaction. The Quality Plans contribute to this goal by defining specific, auditable checks and procedures which must be followed. There are three separate plans applying to different functions: (a) Nuclear Materials Safety Limits (criticality limits); (b) Performance of Critical Experiments, and (c) Criticality Calculations. The three functions are, of course, related, and these interdependencies are indicated in the plans.

Historically, there have always been checks and signatures required for limits and for experiments, while computer codes and calculations have been dealt with on a less formal basis. Part of the reason for this distinction is that the early one-dimensional diffusion or transport codes were not directly applied or applicable to real situations involving fissile material. Plant operations were based on critical or *in situ* measurements, with a large, conservative safety factor.

Today, advances in codes, computer speed, and availability of cross-section data make possible a calculational evaluation of a complicated piece of equipment including effects of adjacent objects, poisons, and changes in process parameters (e.g., concentration or normality). Thus it is imperative that all possible steps be taken to ensure the correctness of the calculational result. For this reason a Quality Plan for the Nuclear Safety Calculational Program was written in 1977 and is currently being implemented. The plan contains some items specific to computer calculations, but the general requirements of documentation, validation, and application are relevant to any calculational method.

The plan is divided into four sections. The Introduction describes the general functions and purpose of the calculational program. The second section, Activities and Responsibilities, lists specific tasks and their purposes and assigns responsibility for performance. The

third section references relevant documentation (e.g., ANSI standards), and the final section describes Quality Plans for specific functions.

Most of the Rocky Flats criticality calculations are done with large codes which have been written at other facilities. Thus, the codes have been used and the results checked, at least for some cases, so that major coding errors are unlikely. The code will usually compile and run with a minimum of changes, and sample problems can be executed.

It should be emphasized that benchmark calculations must be run on the actual computer using the same code and cross-section sets as are to be used for production calculations. The fact that someone at another installation has run benchmark calculations does not serve to validate anyone else's copy of the code. Such reports of benchmark calculations are valuable for defining benchmark calculations and their relation to the experimental data. If one's own computer and copy of the code and cross sections are identical to those used by the author of the report, the same results should be produced.

A file of benchmark calculations is being set up, including input card decks, output, and documentation defining the relation of the calculation to the reported experimental data. The Quality Plan requires that a selection of benchmark calculations be rerun annually to verify that no inadvertent changes have been made to either code or cross sections.

The use of microfiche for storing source program listings and test problem output is strongly recommended, together with the selection of the maximum amount of output information. Cross-section data can also be listed on microfiche. When test problems are rerun or run with a new or modified code, the new and old results can be compared in detail.

The opportunity for error or oversight in criticality calculations is probably greater in the application of a code to a particular problem than in the writing and validation of the code. The first task is to define the required calculations, obtaining information on equipment dimensions, material compositions, and operating procedures. Next, the applicability of available codes must be assessed. Frequently, there are no benchmark data relevant to the system, and some attempt must be made to correlate available data and estimate uncertainties. Then the engineering data must be translated into a form acceptable to the code and punched on cards or entered via a terminal. In either case, some documentation of the actual input data should be maintained. After the calculation is run, the output is checked against the input data and known or expected results. The main protection against error in these steps is an independent double check by another person familiar with the code and the particular problem.

Adherence to the Quality Plan and related procedures is expected to reduce the probability of error in calculations by requiring documentation of benchmarks and applications, thereby causing the user to think through the various steps in a calculation, and by requiring independent double checks of all steps of a criticality analysis from reading dimensions on a drawing to interpreting Monte Carlo results.

Observations on Problem Solving in Nuclear Criticality Safety, R. L. Stevenson (NRC)

The purpose of this presentation is to describe some common errors in nuclear criticality safety analysis uncovered during the review of license applications made in

accordance with Part 70 of the NRC regulations. It is hoped that nuclear criticality analysts will thus be made aware of some specific errors to avoid in the future and be guided in their general approach to problem solving. The errors are broadly categorized as (a) arising from identifiable weaknesses in emphasis, or (b) of specific technical nature.

The following common weaknesses in solving criticality safety problems are listed on the basis of both internal NRC experience and licensee experience.

1. Insufficient consideration of credible off-standard conditions in defining boundary conditions, such as double batching, structural failure, fire sprinkler operation, accumulation of fissile solids in process equipment, flooding, etc. There are lists of such abnormal situations available in the literature.¹ To determine which apply to a particular problem, it is necessary for the analyst to be aware of the operating environment through plant visits, and he should also review the process and controls with knowledgeable operations or process technology personnel.

2. The use of a specific single set of conditions for detailed analysis using a "limiting" case chosen without appropriate analysis. Examples include an arbitrary choice for the unit solution concentration in an array analysis, or arbitrary choice of the limiting process unit for application of the solid-angle method to the process array. To define the limiting conditions with certainty, parametric studies, sensitivity studies, or a survey of unit multiplication factors and subtended solid angles may be necessary.

3. Overemphasis on the mechanics of problem solution, such as development of computer input before detailed review of the problem definition. This may lead to the solution of irrelevant or nonexistent problems, and the corollary may be failure to solve the limiting cases as conceived by the reviewer.

4. Acceptance of computer-generated parameters, particularly values for single point calculations, without adequate confirmation. Comparison of the calculated result with relevant experimental data or other independent calculation is desirable even though the method may have been validated and the input checked. Is it not unusual how often a theoretical explanation can be developed to explain the unexpected in computer output?

5. Overemphasis on the use of the computer so that problems susceptible to the direct use of data in the literature or to simple analysis are solved on the computer. Unnecessary reliance on the computer is of concern because there may be an accompanying loss of understanding of the safety bases and an increased risk of accepting incorrect calculations. There have been some interesting errors which, because of overdependence on KENO, led to conclusions based on essentially irrelevant calculations. One case involved an assumed absurdly low moderator ratio in the KENO input. A more valid conclusion could have been reached on the basis of a modest knowledge of the available literature and a geometric buckling calculation requiring at most only a few minutes with a desk calculator.

6. Failure to allow for the uncertainty in calculated critical parameters in applying widely used safety margins, such as the margins used in TID-7016, Revision 1, to data of unknown validity. If critical values are cautiously predicted, geometry margins of 10 to 20% may be sufficient for the usual uncertainties (e.g., limiting fissile concentration, fabrication tolerances, corrosion) of use; but of course, these same margins may be insufficient to provide for uncertainty both in calculated critical values

and in conditions of use. In this general category of error is the widespread tendency to use, or to quote, a limiting k_{eff} criterion of 0.95. It seems obvious that the implied 0.05 margin and the addition of two or three standard deviations to the k value, calculated from the statistics of the k calculation, are not sufficiently conservative unless the validity of the calculational method has been established for the system in question. Reference is made to the KENO calculations with Hansen-Roach cross sections for the four critical bare arrays,² in which use of a 0.95 k criterion would appear hazardous if appropriate bias corrections were not applied.

Each of the following specific types of errors has been encountered at least twice in licensing review of nuclear criticality safety analyses:

1. The calculation of the effect of a concrete reflector on array reactivity with subsequent failure to make allowance for the effect in setting "safe" conditions. In each of two cases in which this error was made, the effect of the concrete compared to a water reflector was significant—on the order of 0.04 to 0.06 on the array multiplication factor—but was forgotten in applying the calculations.

2. Failure to allow for neutron transmission through concrete between tiers or layers of arrayed fissile units. Calculations reported by Thomas³ and by Welfare⁴ have indicated that as much as an 8-in.-thick layer of concrete is not a general isolator.

Other specific-type errors of analysis are a result of failing to allow for abnormal operations or accidents having a clear credibility of occurrence, and will not be reported in detail. As already noted, these errors include failure to allow for possible double batching of single positions in array analyses, effects of fire sprinkler operation, and dropping of fuel assemblies.

The possibility for errors in analysis exemplified in the foregoing summary is given consideration in the licensing process. In the so-called "broad license," in which the licensee is authorized to make certain changes within a defined technical and administrative framework, the NRC requires that nuclear criticality safety analyses be subject to competent independent review prior to application of the analytical results.

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Pitfalls in Criticality Safety Calculations and Experiences in Operations, G. Chatburn (BNFL-England)

Criticality Safety Assessment work necessitates nuclear, chemical, and engineering studies. In all three fields it is often necessary to use calculation techniques to derive parameters needed in the assessment.

Problems may occur in any of these three fields, but there still remains that area most prone to the unusual event and that is the actual plant operation.

Calculation work is normally predominant at the design stage of a chemical plant, although the requirement can arise when process or laboratory conditions need to be modified.

As design work progresses and individual items of the plant are assessed and accepted by the Safety Assessor, there is a temptation for the designer to reuse "safe units" in a new location without realizing that a changed environment needs a new assessment.

This emphasizes the need for assessment work to be continuous and closely linked with designers. Experience suggests that even when operating plants are slightly modified there is a danger of apparently simple and easy design changes being progressed too far before consulting the Safety Assessor.

Technical difficulties are rather easier to list than unusual events even if it is difficult to quantify some of them. For example, with sophisticated methods like Monte Carlo Codes, errors can be introduced by the use of too few number of stages, to use the techniques in areas where there are too few experimental check points, to oversimplify physical conditions (i.e., modeling errors), to minimize the total number of cases calculated when examining changes of reactivity as parameters such as moderation ratio are changed. Care should be taken to avoid understudying a situation—never rely on one calculation.

Most originators of computational methods have been careful to point out the theoretical limitations of each method, but the assessor must also impose his own limitations in modeling when studying actual plant units. This part of the assessors' study becomes more prevalent once the basic plant design has moved away from the desirable safe-geometry plant into the safe-by-operational-control plant. This then brings into play the calculational techniques of chemical behavior; i.e., dynamic behavior of plants used for solvent extraction operations, and also the noncalculable human behavior or response factor.

Obviously many of the problems are seen at a design or project study phase and are picked up well before they have an effect on real plants, but they can be expensive if detailed design work has been allowed to proceed too far before a safety study has been carried out.

A suggested division of experiences in operations could be into operational (equipment) difficulties, technical information difficulties, and human errors.

At this stage any unpredicted event could create a real and immediate potential for hazard if the safety margins have not been set large enough to cover such events. These margins are always made large enough to cover unexpected events, thus unscheduled changes cause embarrassment rather than a hazard; but where the changes are unpredicted it may be a matter of chance whether real danger of a criticality event arises or not.

Even within a plant unit that has been fabricated on sound safe-by-design principles (safe geometrical sizes well spaced out, with adequate safe geometry spillage collection), the concept that "nothing can go critical" has shortcomings if (a) minor spillages do occur, and (b) cleanup operations are not rigorously controlled.

There will always be the potential for incomplete understanding of safety conditions or limits by the plant operating personnel. In this field it is important to have a clear route through which safe operating conditions or limits are made known and appreciated by the actual plant operators. Experts and Safety Committees are fine and

necessary at the design stages but once a plant is active the major responsibility for safety lies with the operator.

Regular plant inspections or audits by personnel experienced in both operational and safety aspects of chemical plant work are an absolute essential. Experience has shown that minor points of improvement can always be made either in actual assessment or reassessment of minor operational variations or in interpretation of limits.

There is always the temptation to stretch any limit to the full, especially if it is difficult or very time consuming for the plant operator to have a new safety assessment carried out or to have new safe operating conditions specified. An example of this would be where a ^{235}U limit for specific equipment had been set at X percent, the actual material handled could vary a little in enrichment and the operator be tempted to talk in "average" enrichment terms. This need not be an unsafe approach if some further limits to this averaging, i.e., a maximum range had also been assessed and fixed.

Many of the pitfalls of criticality safety assessments have been covered in principle in several excellent papers,¹ but even more insidious are those that rest on the unwary using sweeping generalizations. The so-called safe infinite-sea concentration for water-moderated systems has been used where the high (water) hydrogen concentration is not controlled or guaranteed.

It is not entirely new to point out the types of traps into which the unwary assessor can fall, but open discussion and analysis of such traps can improve the standard and thoroughness of safety assessments.

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QA in Criticality Safety Analysis Needs Experimental Studies, P. R. Lécroché, J. P. LeClerc, J. C. Puit (CEA-France)

Criticality safety calculations are now primarily made using Monte Carlo codes. These codes have such extensive possibilities that it is generally unnecessary to make geometrical approximations. They give more precise results than older and simple methods that had a built-in safety margin (for example, the maximum solid-angle method in interaction calculations). On the other hand, project managers tend to stick more and more to the actual conditions of process installations: physical and chemical composition of fissile mediums (the metal-water dilution law is very seldom used and generally replaced by saltwater or aqueous nitrate dilution law), actual reflectors configuration, etc.

For these reasons, criticality safety quality assurance (QA) needs the qualification of the codes. The problem is complicated by the need to make these qualifications in the many situations encountered in the diverse plants, facilities, and laboratories without omitting the transport packages: uranium with various enrichment, plutonium with different isotopic composition, mediums where thermal- or epithermal- or fast-neutron population is preponderant, homogeneous or heterogeneous mediums coupled with various structure materials.

In France, for criticality safety calculations we use primarily a Monte Carlo code called MORET¹ and Hansen and Roach cross-section sets or cross sections elaborated by a code called APOLLO,² especially in the case of regular lattices of fissile rods. These methods have been tested with experimental results when needed but it now

TABLE I

Medium	Moderation (concentration or H/X)	Geometry
U (93.1%) F_6 —liquid HF U (93.5%) nitrate U (4%) $O_2 F_2$	$5 < H/U < 80$ 30 to 250 g/litre 700 to 1080 g/litre	Sphere Isolated cylinder ϕ 300 and 800 mm Interaction of cylinders ϕ 300 mm - Sphere - Cylinders ϕ 500 and 800 mm
Pu nitrate 3 to 19% ^{240}Pu Pu nitrate 19% ^{240}Pu	40 to 250 g/litre 13 to 150 g/litre	- Cylinders ϕ 300 and 250 mm - Annular cylinder - Parallelepiped 1300 $\times$ 1300 $\times$ 700 mm
Lattices of mixed-oxide rods	Pitch: 1.5 to 3 cm	Cylinder ϕ 500 mm
Lattices of Gd or Hf plates in Pu nitrate 19% ^{240}Pu	22 to 98 g/litre	- Parallelepiped

Fig. 1.

seems necessary to make systematic tests using experiments selected to represent, at best, the range of situations encountered in the installations as described previously.

The choice of existing benchmarks should be the result of international cooperation, which could extend to the definition of additional necessary experiments concerning, for example, low-enriched and low-moderated uranium. The Commissariat à l'Energie Atomique (CEA) is

ready to propose their choice of experiments (especially those listed in Table I) and a series of experiments they are actually performing to study a 4.75 wt% enriched UO_2 rod assembly.

In fact, criticality safety QA goes well beyond the validation of criticality codes. What would be the value of calculations, even using qualified codes, if the hypotheses on which they were based were wrong? There also appears to be a need for experiments to qualify the hypotheses. For example, in the French reprocessing plants the solvent extraction cycles are made in high thickness mixer settlers and criticality safety depends on concentration control. Plutonium concentration profiles depend on the composition and the flow rate of fluids entering the banks; if it is rather easy to predict these profiles in nominal running conditions, it is more difficult to predict Pu concentration peaks in case of perturbations like a diminution of solvent flow rate in the extraction bank. Special calculation codes have been established and they have been tested with experiments conducted in geometry-safe small banks. It has been shown that important perturbations could lead to dangerous Pu concentrations only in such a time span that it is possible to detect and correct the defect in due time. The most sensitive parameters have been selected and are watched particularly.

Another example can be found in fuel element fabrication facilities in which it is necessary to mix UO_2 or PuO_2 in batches as large as possible. From the criticality safety point of view, such operations are possible only if the materials are very low moderated. Experiments have been conducted to measure the maximum humidity absorption ratio of different salts (Fig. 1). Acceptable batches may then be established for a blender taking into account a humidity ratio including a safety margin.

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Data and Analyses for Criticality Safety

Uncertainties in K-Effective Calculations, H. W. Webb (B&W)

Introduction

A basic tool for nuclear criticality safety calculations is the Monte Carlo computer code KENO.¹ The primary use of this code is for determining the neutron multiplication (K-effective) of a wide variety of systems containing fissile material. In the following example, a complex poisoned unit containing low-enriched UO_2 and water is evaluated with KENO. The variations in the resulting K-effective values are due to both the statistical nature of the calculations as well as to the starting neutron distribution. This example illustrates the importance of not relying on the results of only one calculation.

Problem Description

In a fuel processing plant, a tall cylindrical vessel will contain uranium dioxide (UO_2). The nuclear safety demonstration for this vessel involved determining the K-effective of the unit under accident conditions with the use of the KENO code. The geometry of the system as modeled in the calculations is shown in Fig. 1. The most significant feature is a boron carbide rod positioned axially in the lower three-quarters of the cylinder. The model assumed ~196 mm of water reflection around this region. Because of the unsymmetric arrangement of the components, it was necessary to use the generalized geometry option for describing the system.

The accident evaluation assumed all the available space within the unit to be occupied by a homogeneous mixture of UO_2 and water with the maximum enrichment of 4.0 wt% ^{235}U in U. A uranium density of 2.4 gU/cm^3 was used because it resulted in the highest K-effective

for the lower section of this unit when analyzed with ANISN. Both the ANISN and subsequent KENO calculations used the Hansen-Roach 18-group cross sections.²

Calculations

Table I summarizes the results of seven KENO calculations. This table also includes the input parameters which were varied to make the calculations independent. The weighted average K-effective from these seven runs is 0.9303 with the standard deviation of the average being 0.0027. At the 95% confidence level, the actual K-effective, for which the calculated value is an approximation, lies below 0.936. If the criterion for nuclear safety is that K-effective at the 95% confidence level be < 0.950 , then the weighted average of the seven runs would meet the safety criteria.

Discussion

There are two factors that contribute to the variations in K-effective shown in Table I. The first is the fact that KENO calculations are statistical in nature. Varying the number of neutrons per generation results in statistically independent calculations. This is the reason for the variations observed between the fourth, fifth, and sixth calculations. The other factor is the spatial distribution with which the first few generations of neutrons are started. Differences due to two slightly different starting distributions can be seen by comparing the first two calculations.

These data show that the K-effective from any one calculation may vary from the actual value by a significant amount. In fact, another paper on a similar subject³ also illustrates how this pitfall can occur when the fission density distribution does not adequately account for the distribution of fissile material. The system discussed here illustrates a situation where the fission density distribution is strongly influenced by the presence of a lumped neutron poison. This can also contribute to uncertainties in the calculated K-effectives. The user can most easily recognize the potential for errors of this kind by comparing the relative fission densities in localized sections of the system with the fissile material distribution. The optimum condition would occur when the two distributions are the same. In fact, this should be considered any time the fissile material is distributed in an unsymmetric or complex manner. This also must be considered any time there are lumped-neutron absorbers in the system.

Determining the fission densities in localized sections of the system is a fairly easy matter when the system is described using the specialized geometry option. Creating extra box types will allow as much detail as desired. However, when using the generalized geometry option, the fission density distribution data are not available in a useful form. Therefore, in these cases, careful thought must be given as to what the best starting distribution would be. Generally, more than one neutron start distribution should be tried for the more complex geometric configurations. Often the simplest approach is to start most of the neutrons in the region where the highest fission density would be expected.

Fig. 1. Geometry model for KENO calculations.

TABLE I
KENO Calculation Results and Input Parameters

K-EFFECTIVE	σ	# GENERATIONS	# 0^{N1} PER GENERATION	START TYPE	# DROPPED GENERATIONS	KENO VERSION
0.940	0.008	75	165	PT(0,0,-8.45)	12	II
0.934	0.008	75	165	PT(0,0,0)	6	II
0.944	0.006	120	100	COSINE	3	II
0.905	0.006	120	100	COSINE	10	IV
0.935	0.007	60	200	COSINE	8	IV
0.937	0.007	40	300	COSINE	7	IV
0.919 ^a	0.007	44	300	COSINE	4	IV

^aThis case also differed from previous calculation in that the reflector thickness was increased by ~100 mm.

Conclusions

The type of considerations discussed here are of importance primarily with systems where the calculated K-effectives are within a few standard deviations of the acceptance criteria. The data in Table I are examples of the variations that can be obtained with KENO when several independent calculations are made. Extra care must be used when the complexity of the system requires the use of the generalized geometry option in KENO. This is because the best neutron start distribution is difficult to determine. At Babcock & Wilcox we reduce the likelihood of reporting anomalous values of K-effective by investigating the effects of various neutron start distributions and/or by performing a parametric study. Although KENO is an excellent tool for nuclear criticality safety calculations, it, like all sophisticated techniques, does have the potential for misuse. This is not a new observation but rather something all users of KENO should recognize.

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Adventures in Ten Dimensions—Misleading Assumptions in Array Analysis, C. M. Hopper (UCC-ND)

Introduction

A KENO¹ computational study was made to demonstrate the nuclear criticality safety of a fissile material package.² The study revealed certain expected results, but also highlighted the pitfalls of anticipating that the results would be consistent under varying circumstances. Various parameters (dimensions) were varied, and the KENO-calculated k_{eff} values were observed. The independent variables of the packages within the reflected arrays studied were:

1. mass of ²³⁵U loadings
2. ²³⁵U enrichment
3. internal moderation of ²³⁵U loadings
4. compositions of the fissile material (metal, oxides, etc.)
5. density of the fissile material loadings
6. geometry of the loading (solid or hollow cylinders)
7. shape of the loadings (height-to-diameter ratio)
8. number of packages
9. interstitial moderation between packages.

The effects on the dependent variable, the KENO-calculated k_{eff} value, caused by changing single parameters, were categorized and an attempt was made to establish the consistency of these effects.

Nearly half the trends observed were inconsistent. Alterations of uranium enrichment, fissile material height-to-diameter ratios, geometries of loadings, or densities of fissile material may or may not increase the reactivity of selected arrays. However, an increase of ²³⁵U unit mass loadings, numbers of packages, internal moderation of ²³⁵U loadings, or a decrease of interstitial moderation between packages increase the KENO-calculated reactivities for all circumstances investigated.

The container under consideration was quite similar to a 30-gal Specification 6M container, but with a 165.1-mm (6.5-in)-diam inner containment vessel. The loadings examined were:

Form of Material	²³⁵ U Density (g/cm ³)	H/ ²³⁵ U Atom Ratio
U(100) Metal	18.76	0.00
U(100)O ₂ + H ₂ O	8.33	0.43
U(100)O ₂ + H ₂ O	4.55	3.22
U(100)O ₂ + H ₂ O	2.05	10.7
U(100)O ₂ + H ₂ O	1.15	21.4

Single-package loading limits were chosen which were subcritical when fully reflected with water. The calculational array analysis was based on the KENO-calculated subcritical number of packages with full water reflection surrounding the arrays. Masses of ²³⁵U per package were selected to produce calculated $k_{eff} + 2\sigma$ values that were <0.970.

Sequence of Analysis and Assumptions

The stepwise sequence of analysis and the accompanying assumptions are as follows:

1. The mass loadings permitted in each package were varied to produce arrays having calculated $k_{eff} + 2\sigma$ values < 0.970 .

Assumption—The mass increase of each form of ^{235}U per package within a given array size will increase the calculated k_{eff} . The results of this examination demonstrated the truth of the assumption.

2. All packages calculated in Step 1 were considered to be thermally damaged; therefore, it was necessary to examine the effects on the KENO-calculated k_{eff} due to further interstitial moderation caused by the normal content of the insulation or wetting by water.

Assumption—Calculational experience indicates that cane fiberboard insulation in packages of this size tends to overmoderate an array.

This investigation was made by inserting reduced-density water within the insulation and between packages. Addition of water to the thermally damaged package insulation reduced the calculated k_{eff} for all large, intermediate, or small arrays considered, regardless of the form of ^{235}U loading.

3. The calculations in Steps 1 and 2 were performed for full-density solid cylinders. Since various densities of materials were to be permitted in the package, it was necessary to evaluate the effects on the k_{eff} values due to a reduction of material density and geometry.

Assumption—Full-density units within an array will produce the greatest reactivity.

The investigation revealed that this effect was highly dependent on array size and form of the enriched uranium material. The k_{eff} s of small arrays (75 packages or less) are reduced by lowering the density of the loading or changing the shape of the full-density loading into a hollow cylindrical shell. This effect was reversed for large arrays (200 to 1000 packages), depending on the form of the enriched uranium. Handley³ and Koponen⁴ have previously reported such effects in hydrogenously moderated arrays.

4. To permit large masses of material (i.e., greater than the maximum mass of safe water-reflected spheres), it was necessary to investigate the effect of altering the unit shape (height-to-diameter ratio).

Assumption—Increasing the unit height-to-diameter ratio for larger masses within arrays would ensure the safety of the single-package loading when flooded, and it would not significantly alter the array reactivities.

It was calculationaly determined that altering the unit shape would also increase or decrease k_{eff} , depending on the array size. However, the mass restrictions imposed by Step 1 of the analysis and the size of the containment vessel precludes an unacceptable calculated reactivity.

5. The reactivity responses of selected arrays with fixed masses of ^{235}U , but at varying enrichments, were investigated.

Assumption—Decreasing the uranium enrichment while maintaining a fixed ^{235}U mass per package will reduce the array reactivities.

This assumption was proven true for all full-density uranium metal units, but was proven false for all full-density oxide units. The effects on array reactivity were variable for reduced-density and hollow, cylindrical shell units, and depended on the array size.

6. To assure that the analysis would suffice for other compounds of uranium (excluding beryllium and deuterium), graphite at 1.9 g/cm^3 was inserted into the reduced-density dry UO_2 loadings which demonstrated the largest increase in array reactivity.

Assumption—Graphite, at this density, provides greater atomic densities and lower neutron absorptions than other elements of uranium compounds not investigated; and, therefore, will increase the reactivities of permitted array loadings more than other diluents.

The insertion of carbon into the reduced-density dry oxide increased the reactivity of the selected arrays. An independent review of this assumption was obtained from a noted authority on such matters. The results of the review confirmed the assumption.

7. Since alterations of uranium enrichment, fissile material height-to-diameter ratios, geometries of loadings, densities of fissile material, ^{235}U mass per package, number of packages, internal moderation of ^{235}U loadings, and interstitial moderation between packages caused increases in the array reactivities, it was desired to obtain the most reactive combination of parameters that might be encountered in using the container. The parameters of major importance (i.e., reduction of material density, reduction of uranium enrichment, and content of carbon) were examined. It was inferred from the data that the most reactive combination of permitted parameters would be a $13 \times 13 \times 8$ array of packages with loadings of dry uranium dioxide at a uranium enrichment of 83.6 wt% (as 12.35 kg of ^{235}U) with hydrogen-to- ^{235}U and carbon-to- ^{235}U atom ratios equal to 0.43 and 20.03, respectively.

Assumption—A nuclear criticality safety analyst has the ability to comprehend and optimize numerous parameters to develop calculational models that will depict the most reactive circumstances envisioned for array analyses.

A KENO calculational examination of those parameters selected, and other surrounding combinations, did indicate that the chosen set of parameters was the most reactive of all calculations observed. The absolute truth of the assumption is not proven without a prohibitive number of individual case-by-case calculations; therefore, an independent review of all calculational results was obtained from knowledgeable personnel within the field of process analysis and optimization. The goal of the independent reviewer was to obtain the most reactive combination of parameters within the limitations of the permitted contents of the package. This goal was approached by developing a mathematical surface-response equation which modeled the behavior of all KENO-calculated results obtained throughout the various steps of the array analysis. The mathematical model fit all the data available with a considerable degree of confidence. The model was then analyzed by a computer-optimizing program to obtain the most reactive combination of parameters. The result of this review revealed very nearly the same combination of parameters as that selected by the nuclear criticality safety analyst.

Conclusions

It is concluded that nearly all assumptions related to a moderated-array calculational study must be tested. For instance, a calculation of full-density highly enriched uranium loadings is not in itself proof of the safety for all lower densities, lower enrichments, or material diluents that may occur. The only assumptions that may not need testing are that (a) the reduction in mass of fissile material loadings, or (b) a reduction in the number of packages within an array, will produce lower k_{eff} values.

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Limiting Critical Concentration and Eta for ^{239}Pu , R. C. Lloyd, R. A. Libby, E. D. Clayton (BNWL)

The purpose of these experiments is to provide data for determining the thermal value of eta and the limiting critical concentration of ^{239}Pu in an aqueous solution. The limiting critical concentration is that concentration of the fissile nuclide in solution at which k_{∞} would become unity. At smaller concentrations the reproduction factor would be less than unity for systems of any size or shape. This value is useful as a benchmark for testing calculation methods and cross-section sets in well thermalized systems. In previous work, the limiting critical concentration was determined indirectly from null reactivity measurements on a relatively small volume of solution, wherein several corrections and extrapolations were required to determine the concentration.¹ Analysis of the data gave a value of 7.8 to 8.0 g Pu/litre. A recent reevaluation of the data has produced a value of 7.2 g/litre.² Experiments were performed previously on small spheres with high plutonium concentration.³ The present experiments were performed to determine the limiting critical concentration and provide data on a large, low-leakage plutonium system that would be suitable for determining the thermal value of eta.

Critical experiments were completed with plutonium nitrate solutions in a large 4-ft-diam aluminum sphere. This identical sphere was previously used in critical experiments at the Oak Ridge National Laboratory in the measurement of eta for uranium in critical aqueous solutions.⁴ The plutonium used in the current experiments was selected for its low ^{240}Pu content, which was only 2.52 wt% ^{240}Pu . The sphere volume was 949.1 litres. For low plutonium concentrations in aqueous solutions, the neutron leakage from a sphere of this size will be no more than several percent. Measurements were made with $\text{Pu}(\text{NO}_3)_4$ water solutions with Pu concentrations ranging from 11 g Pu/litre to <8 g Pu/litre with nitric acid molarity at about one.

The data from the experimental determinations of the full critical sphere and the chemical analysis for the solutions are given in Table I. The plutonium concentration was determined to be 9.54 g/litre in the solution as used in the experiments to make the sphere critical when just full.

The critical equation for an aqueous homogeneous thermal reactor may be written as

$$k_{\text{eff}} = k_{\infty} P(B) = 1,$$

TABLE I
Experimental Data for Sphere

Measured volume: 949.1 litre				
Material: Al				
Al thickness: 0.77 cm				
Temperature: $23 \pm 0.5^\circ\text{C}$				
Chemical Analysis				
Pu concentration: 9.54 g/litre				
Sp.G.: 1.050				
Free acid M: 1.027				
Isotopic Analysis (wt%)				
^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
0.004	97.386	2.521	0.075	0.014

in which $P(B)$ is the nonleakage probability and the infinite reproduction factor $k_{\infty} = \eta f F$. The factor F accounts for epithermal fission and absorption of neutrons while slowing down; f denotes the common thermal utilization. As the reactor is increased in size, the neutron leakage becomes small and $P(B)$ will approach unity; under these circumstances, η (apart from small corrections for epithermal fissions) will become equal to $1/f$ (reciprocal of thermal utilization). The measurement of eta then depends on two things: (a) knowledge of absorption cross sections for the aqueous solutions in question and ability to compute f , and (b) the nonleakage probability which must be either computed or measured.

A dilute homogeneous aqueous solution was used so the critical size would be large [$1 - P(B)$ small] and the system well thermalized. Both conditions are required for an accurate measurement of eta. Small neutron leakage means less dependence on the accuracy of either the measured or calculated values of the critical (geometrical) buckling, the age, and thermal diffusion length; thermal-neutron absorption cross sections are relatively well known.

Determinations of eta for ^{233}U and ^{235}U have been made by Gwin and Magnuson.⁵ In a later note,⁶ Gwin suggested using this same 4-ft-diam sphere in measuring η for ^{239}Pu and discusses an evaluation method, as suggested by Leonard,⁷ where the critical experiment is used to determine η .

The relation between the 2200 m/s parameters and the measured atomic concentrations for a critical system will be related using a simple four-factor formula for criticality for the large critical system under consideration.

The k_{eff} of the sphere was computed using ENDF/B-IV cross-section data in an 18-group calculation with the HFN code.⁷ Epithermal energy groups were obtained via application of the ETOG⁸ code and the GAM portion of the EGGNIT code,⁹ and the single thermal group data via the FLANGE code¹⁰ followed by the TEMPEST portion of the EGGNIT code. A k_{eff} value of 1.024 was calculated. It is assumed the 0.024 Δk difference from unity is due to bias in the ^{239}Pu cross sections. Applying this Δk to $^{239}\text{Pu} + \text{H}_2\text{O}$ systems results in a limiting critical concentration of 7.7 g Pu/litre for a $^{239}\text{Pu} + \text{H}_2\text{O}$ mixture. Initial analytical calculations of $\bar{\eta}$ for ^{239}Pu provide a value of 2.06.

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Neutron Interaction Analysis of a Large Nuclear Facility, W. R. Waltz, R. C. Ashline (AGNS)

Introduction

This paper describes a systematic approach for evaluating the criticality safety of an array of equipment items in a large nuclear facility. The approach has the greatest utility where complicating factors are present, such as the existence of a large number of units of varying sizes, shapes, orientations, contents, spacings, and variable reflectors.

This approach has been applied successfully in the analysis of the Barnwell Nuclear Fuel Plant's Uranium Hexafluoride (UF_6) Facility, and should have applicability to other facilities.

Background

In the design of a facility (or reanalysis of an existing facility) the question of potential criticality problems arising from neutron interaction effects is oftentimes difficult to address. The explicit representation of a large number of equipment items may well exceed computer capabilities. The task is further complicated if the equipment items do not fit regular patterns as to size, shape, orientation, spacing, and contents.

Another complicating factor sometimes not fully recognized is the effect of variable reflectors within the array, such as the presence of operating and maintenance personnel, which can have a profound impact on the analysis. Consider a case involving 50 equipment items of variable sizes and shapes but containing no reflectors within the array. In this situation the configuration to examine is clearly the entire 50-unit array with, perhaps, the facility walls represented on the array periphery. Next, consider the introduction of reflectors (such as personnel) internal to the array. One could imagine a small subgrouping of, say, two equipment items with personnel surrounding the units forming a close-spaced reflector around the perimeter of the two-unit system. Depending on several factors, the 2-unit array might represent a worse case than the 50-unit array described first. In principle, there are a large number of subgroupings that should be considered. The objective of the interaction analysis is to demonstrate that there is no grouping or subgrouping of equipment items in the facility for which a criticality problem exists. For complex equipment arrangements it will not be apparent which equipment groupings represent potential problem areas.

General Approach

The UF_6 Facility with over 150 equipment items of varying size, shapes, and orientations and the existence of variable reflectors required a formidable interaction analysis program. A systematic approach was needed to reduce the problem to a manageable level. Subsequent effort produced a four-step approach, which is summarized below:

Step 1—Simplifying Assumptions—The objective of the first step was to provide simplifying assumptions for later use in the facility analyses. In this step the reactivity effects of intervening moderation and of the vessel walls were examined. A set of KENO-IV calculations¹ was performed on hypothetical equipment groupings. Results of these calculations, shown in Fig. 1, show that for the systems of interest both the vessel walls (i.e., steel) and intervening moderation could be excluded from explicit representation in the facility interaction calculations. The results show that a small amount of intervening moderation causes array reactivity to increase slightly above that for the base case. However, when the vessel walls are included in the calculation along with intervening moderation, the net effect is to reduce reactivity below the base case. Ignoring the vessel walls and intervening moderation from explicit representation in facility calculations resulted in considerable modeling simplification.

Step 2—Interaction Guidelines—The objective of this step was to provide insight into the question of what equipment arrangements (i.e., number of equipment items, size of units and relative spacings) are candidate interaction problems. This insight was obtained by performing KENO calculations involving hypothetical equipment groupings and varying the basic parameters involved. Results from this step highlighted the extreme sensitivity of array reactivity to the separation distance between units. For example, results showed that beyond a surface-to-surface separation distance of approximately one unit diameter, the k_{eff} of a reflected nine-unit array is less than that for a single unit at full water reflection.

Step 3—Review of Facility Layout—The purpose of this step was to identify specific facility equipment groupings that represent potential criticality problems. This was accomplished by comparing the actual facility equipment layouts with the guidelines provided from Step 2.

Fig. 1. K_{eff} vs the thickness of a water slab between units. The separation distance (D) is a constant ($D = 20.95$ cm, $D/D_0 = 0.5$). Units contain UO_2 and water at optimum moderation; 1.8% enriched.

Step 4—Facility Analysis—For each equipment grouping identified in Step 3, a specific criticality calculation was performed using the KENO-IV program. It was usually necessary to use the generalized geometry option of KENO-IV to achieve a detailed, accurate description. In certain cases involving equipment items with irregular shapes and orientations, it was necessary to make several computer runs varying the reflector boundary to ensure that the worst-case reflector condition of the array had been considered.

Summary

The four-step approach described above was successfully applied to the evaluation of the UF_6 Facility. Results from the screening procedure in step 3 identified 12 specific equipment groupings requiring interaction analysis. Detailed analysis in step 4 identified two areas requiring physical changes. We believe the approach described represents the best available means to attack the interaction analysis of a large facility. The method should have application for similar analyses of other facilities.

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Concrete and Criticality, R. D. Carter (Rockwell Hanford)

To generalize on the reactivity effects of reflecting and interspersed concrete on plutonium can be risky. Concrete composition is quite variable and has different effects with different fissile shapes. Arrays of plutonium oxides in concrete may have higher reactivities with

lower fissile densities. Certain examples will show when some of these effects occur and their magnitude.

Concrete comes in many compositions with a number of different aggregate compositions and water fractions, drying as it ages. At the Hanford site, the normal aggregate is basalt and contains little carbon. The water fraction in Hanford concrete used in critical mass laboratory (CML) experiments has been analyzed as high as 11 wt% (Ref. 1). The minimum water content is the water of hydration (6 wt%).

Concrete from CML experiments was shown to be a better reflector than water for long cylinders and slabs so this wet concrete was used whenever concrete reflection occurred. Subsequent calculations for large fissile storage rooms with concrete walls have shown, however, that the dry concrete is more restrictive, and even later calculations for arrays of concrete-walled storage cells have shown that the k -effective could increase as much as 1% for every 1% decrease in water content. The use of the CML composition would then result in a lower calculated k -effective by 5% than the use of dry concrete; since the concrete composition is not normally known, that now used for large room calculations is the CML composition with the hydrogen content reduced to slightly below the water of hydration equivalent (to 0.50 wt% hydrogen).

For single units, however, the reverse effect can sometimes occur. Recently, calculations for pipes through 254-mm concrete walls were made using dry concrete. A check with the wet concrete showed that it increased the k -effective about 4%.

To give some guidance for determining when a given reflector is best for a single unit, a series of calculations using the DTF-IV code² with GAMTEC II³ cross sections was made for finite cylinders and slabs of plutonium-water solution at 450 kg Pu/m³. The cylinders were 152.4-mm diam with a 254-mm radial reflector. Reflectors were water, or concrete with either 0.5 or 1.23 wt% hydrogen. The best reflector was found to be water up to a cylinder height of 320 mm, the wet concrete was best from 320 to 900 mm and the dry concrete was best above 900 mm.

The solution slabs were 50.8 mm thick, infinitely long, with widths varying from 100 to 200 mm and with a 254-mm reflector on both large faces. The best reflector was found to be water up to a slab width of 300 mm, wet concrete was best from 300 to 540 mm, and dry concrete best above 540 mm in width. At greater than the 540-mm slab width, the dry concrete reflector caused a k -effective up to 4% greater than the wet concrete. Below that point, the reverse occurred.

These calculations show that no single concrete composition is always conservative. Generally, drier concrete should be used for large systems and wetter concrete for smaller systems. The effective system size, however, may not be a function of the size of a fissile container, but of the length over which the reflector is applied (the pipe-through-a-concrete-wall example).

Another application of concrete is as an isolating medium. The isolating thickness was shown in the earliest CML experiments to be 249 ± 20 mm (Ref. 4). Recent calculations show that the isolation thicknesses are strongly dependent on core size. These calculations assumed two facing square slabs of plutonium solution surrounded by concrete, each of half-critical thickness and with varying lengths of sides. The dry concrete had a somewhat larger isolation thickness with isolation

Fig. 1. Reactivity of 2.5-kg plutonium units stored in 203.2-mm holes in concrete. Seven units high on 304.8-mm centers vertically. KENO IV, GAMTEC II calculation.

not occurring for the larger slabs until well past 500-mm slab separation. For less than isolating thickness, k -effective could change in a complex pattern. For example, for 100-cm square slabs, k -effective was higher with the dry concrete only at slab separations between 10 and 50 cm.

A different type of effect was shown to occur in a storage system where fissile units were stored vertically in holes surrounded by concrete. The results are shown in Fig. 1 for 2.5-kg plutonium units. The metal units are the most restrictive unmoderated units for >370-mm center-to-center hole separation. Low-density oxide (2 g Pu/cm³) units were more restrictive at <370 mm and were more restrictive than higher density oxide units (10.11 g Pu/cm³). However, if the high-density oxide units were made into annular cylinders with the outside dimensions the same as the low-density units, the higher density units were slightly more restrictive. If the low-density oxide units were fully moderated, a large increase in k -effective resulted. Therefore, the method of packaging the fissile material can strongly affect the reactivity. High-density plutonium oxide packaged in a number of plastic bags within a container would have some moderation and a large surface area. It would certainly not be conservative to calculate an array of such units as single units of compact dry oxide.

The above data show that material composition should be known as precisely as possible or worst-case assumptions should be used (and verified as being the worst case).

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Reactivity Coefficients of Heavy Isotopes in LASL's Fast Critical Assemblies, C. C. Byers, G. E. Hansen, J. J. Koelling, E. A. Plassmann, D. R. Smith (LASL)

We are attempting a systematic investigation of heavy isotope behavior in the Pajarito fast-neutron critical assemblies. Included will be central reactivity coefficient measurements and fission rate measurements in natural-uranium-reflected plutonium and 93%-enriched uranium metal assemblies and in Big Ten, a depleted-uranium-reflected 10%-enriched uranium metal assembly.

A few reactivity coefficients have recently been measured for ²³⁵U, ²³⁷Np, ²³⁸Pu, ²³⁹Pu, ²⁴²Pu, and ²⁴¹Am. The experimental technique has been described by Stubbins et al.¹ and preliminary results are listed in Table I.

Similar data for some of these isotopes in unreflected plutonium (Jezebel) and 93%-enriched uranium (Godiva) have previously been reported.¹⁻³

Central reactivity coefficients are closely related to neutron net-production cross sections of the samples that, in turn, are related to criticality. By establishing the bias in the computed coefficients, one may obtain improved estimations of critical dimensions. Six-group cross-section sets, developed for the above isotopes, have been used in this fashion to give the following bare-sphere critical mass estimates.

$$m_c(^{238}\text{Pu}) \approx 12 \text{ kg}, \quad m_c(^{242}\text{Pu}) \approx 90 \text{ kg},$$

and

$$m_c(^{241}\text{Am}) \approx 58 \text{ kg}.$$

It is hoped that these data will be useful to the standards work group that is currently developing a criticality safety standard for operations with the actinide elements.

TABLE I

Central Reactivity Coefficient Ratios in Flattop-Pu, Flattop-U(93), and Big Ten

Isotope	Flattop-Pu	Flattop-U(93.2)	Big Ten
²³⁵ U	0.524 ± 0.007	0.528 ± 0.010	0.585 ± 0.006
²³⁷ Np	---	0.42 ± 0.02	0.065 ± 0.006
²³⁸ Pu	0.899 ± 0.014	0.889 ± 0.014	0.661 ± 0.018
²³⁹ Pu	≈1.00	≈1.00	≈1.00
²⁴² Pu	0.494 ± 0.010	0.479 ± 0.010	---
²⁴¹ Am	0.605 ± 0.011	0.530 ± 0.010	---

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Benchmark Critical Experiments: High-Enriched Uranium Solution, Robert E. Rothe, Inki Oh (Rockwell Int)

Accurate criticality data are essential for any validation of calculational techniques employed in safety-related considerations. Some popular calculational methods yield k_{eff} less than unity by various amounts for many experimentally critical uranium solution systems.^{1,2} One principal area of discrepancy has been unreflected, high-enriched uranium solution systems.

To answer this need, 76 critical uranium solution systems are described for a wide range of material and geometry specifications. Experiments were done in free space under minimal reflection and fully reflected by thick-walled cubical shells of concrete or plastic. Both single units and arrays of cylinders were studied, and in both cases the uranium concentration within the solution also varied. The effect of both Type 6061 aluminum and Type 304 stainless steel as solution containers was also determined.

Table I contains experimental critical heights for a single uranium solution cylinder under the reflection condition described. The uncertainties given are half the range between repeated experiments. Two entries enclosed in square brackets are derived from extrapolation of reciprocal multiplication curves because, unfortunately, the tanks were ~2 cm too short for criticality to occur.

The experimental critical heights for arrays of cylinders are presented in Table II. Uncertainties have the same meaning as in Table I. Stainless-steel sleeves were slipped over the aluminum pipe to study the effect of this material instead of fabricating a duplicate set of tanks. The column designated "SS Sleeve" states if such sleeves were present for each experiment.

The uranium solution was uranyl nitrate [$\text{UO}_2(\text{NO}_3)_2$] dissolved in dilute nitric acid and further diluted with water to the concentrations given in the two tables. The uranium was enriched to 93.172 wt% ^{235}U with 234, 236, and 238 present to about 1.0, 0.4, and 5.4 wt%, respectively.

All tanks were open-topped, right-circular cylinders with a flat-bottom disk welded in place. The stainless-steel sleeves were rolled to fit closely over the cylinders and then compression clamped for a better fit once in place. All materials used were nominally 0.32 cm thick except the tank bottoms for the single-unit experiments, which were twice that thick. Aluminum interiors that might contact the slightly acid solution were protected with a thin coat of epoxy paint.

A type of concrete representative of that used in the nuclear industry was selected for one reflector shell for this program. The composition of concrete has proved difficult to determine in the past, so particular attention

TABLE I
Critical Height of a Single Enriched Uranium Solution Cylinder

TANK		CONCENTRATION (g U/4')	CRITICAL HEIGHT (cm)	
Material	I.D. (cm)			
Minimally-Reflected, Approximately centered in a ~10 m cubical room having thick concrete walls.				
SS	27.92	145.68 346.73	31.20 ± .04 28.93 ± .09	
Al	28.01	142.92 357.71	33.55 ± .03 30.91 ± .04	
	33.01	54.89 59.65 137.40 145.68 357.71	39.48 ± .13 36.67 ± .17 23.96 ± .13 23.67 ± .03 22.53 ± .05	
		50.69	63.95	20.48 ± .05
		25.7-cm-thick Concrete Shell (tank centered in shell)		
		SS	27.92	144.38 334.77
Al	28.01	144.38 334.77	31.37 ± .01 28.60 ± .03	
	33.01	59.65 144.38 334.77	34.10 ± .02 22.85 ± .03 21.50 ± .01	
	(tank in one corner of shell)			
SS	27.92	144.38 334.77	24.19 ± .01 21.79 ± .01	
Al	28.01	144.38 334.77	24.70 ± .01 22.33 ± .01	
	33.01	59.65 144.38 334.77	27.27 18.24 ± .01 16.78 ± .02	
	20.6-cm-thick Plastic Shell (tank centered in shell)			
SS	27.92	147.66 345.33	29.71 ± .05 27.60 ± .01	
Al	28.01	60.32 147.66 345.33	[78.1] 31.26 ± .01 28.84 ± .02	
	33.01	60.32 147.66 345.33	34.33 ± .02 22.78 ± .01 21.67 ± .00	
	(tank in one corner of shell)			
SS	27.92	60.32 147.66 345.33	50.52 ± .09 25.03 ± .03 22.75 ± .01	
Al	28.01	60.32 147.66 345.33	51.67 ± .05 25.26 ± .00 22.87 ± .01	
	33.01	60.32 66.33 147.66 345.33	27.70 ± .27 25.10 18.49 ± .04 17.20 ± .03	
	(tank centered on floor of shell)			
SS	27.92	60.32	67.48 ± .20	
Al	28.01	60.32	[77.1]	
	33.01		31.75 ± .03	

TABLE II
Critical Height of an Array of Enriched
Uranium Solution Cylinders

TANK CONFIGURATION			CONCENTRATION (g U/L)	CRITICAL HEIGHT (cm)
I. D. (cm)	Array ^a	SS Sleeves		
25.7-cm-thick Concrete Shell				
21.12	4 x 4 full array	yes	67.28	28.63 ± .03
		no	67.28	27.15 ± .01
		yes	369.96	17.24 ± .10
		no	364.11	17.13 ± .02
	2 x 2 centered in shell	yes	76.09	60.70 ± .06
		no	76.09	62.34 ± .16
		no	80.72	57.88
		yes	360.37	29.49 ± .01
no	364.11	31.11 ± .03		
16.12	4 x 4 full array	yes	83.49	57.34 ± .05
		no	83.49	51.21 ± .02
		yes	360.37	32.32 ± .09
		no	359.55	31.82 ± .01
	2 x 4 ^b	no	359.55	51.45
		no	359.55	65.49
	2 x 2 centered	yes	359.55	101.45 ± .03
		no		104.04 ± .03
20.6-cm-thick Plastic Shell				
21.12	4 x 4 full array	yes	60.32	34.82 ± .02
		no	60.32	31.76 ± .00
		yes	355.94	19.27 ± .01
		no	355.94	18.82 ± .00
	2 x 2 centered in shell	yes	60.32	110.20
		no	60.32	102.29 ± .09
		yes	355.94	31.93 ± .01
		no	355.94	33.20 ± .02
16.12	4 x 4 full array	yes	60.32	105.85 ± .03
		no	60.32	78.40 ± .05
		yes	355.94	38.10 ± .03
		no	355.94	35.56 ± .02
	2 x 3 ^d	yes	355.94	95.20 ± .04
		no		89.78 ± .02

^aSixteen holes in the reflector shell bottom formed a square pattern for a maximum 4 x 4 array with a lattice spacing of 30.5 cm and half that between the walls and perimeter cylinders.

^bAdjacent to one wall. Exactly one-half of a full array.

^cIn one corner of the reflector shell.

^dCentered on one wall of the shell projecting toward the center of the reflector.

was paid to this requirement. The concrete was analyzed by two independent methods at two different laboratories. The resulting elemental compositions are in excellent agreement with one another. The water content of the curing concrete was also carefully monitored. The interior dimension of the shell is ~122 cm and the wall thickness is 25.7 cm. The shell is assembled from six flat slabs. Principal elements, in wt%, are: H(0.75), C(5.55), O(49.29), Si(15.70), and Ca(22.95). The cured concrete density is 2.321 g/cm³.

The plastic reflector shell was constructed of Plexiglas[®] [CH₂C(CH₃)CO₂CH₃]. The interior dimension is the same as the concrete, but the wall thickness is 20.6 cm. It is also assembled from six flat slabs, each lami-

nated from two thicknesses of plastic. The material composition of the 12 sheets varied because some of the plastic proved to be fire retardant. All four side panels ($\rho = 1.186 \text{ g/cm}^3$) contain, in wt%: H(8.03), C(59.72), and O(32.14). The top and bottom panels ($\rho = 1.286 \text{ g/cm}^3$) contain: H(7.18), C(52.68), O(29.40), Br(6.5), Cl(1.6), and P(1.5).

A journal article, which completely specifies all material and geometry properties of all critical measurements, has been submitted to *Nuclear Technology*.

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Analyses and Experiments for Shipping Cask and Fuel Storage Criticality Evaluations

History and Application of SCALE System to Nuclear Criticality Reviews, *R. H. Odegaarden (NRC)*

The purpose of this paper is to present the background and intended use of the standardized computer analyses for licensing evaluation (SCALE) system for evaluating criticality aspects of transport package designs. The SCALE system currently incorporates criticality, radioactive source term, shielding, and thermal computer programs. The principal idea in developing such a system is to provide a standard method for use by NRC licensees and the NRC staff in evaluating package designs.

The NRC staff identified a strong need to develop a standardized method of analysis for evaluations of criticality aspects of package designs. This would allow licensees to know in advance how their package designs would be evaluated. In developing the SCALE system, the following items were considered:

1. operational on both IBM and CDC computer systems
2. readily available
3. minimum computer core
4. minimum running time
5. minimum input description
6. documentation of programs
7. direct correlation with experiments
8. range of applicability
9. publication of regulatory guides.

In developing the SCALE system, NRC contracted with Computer Sciences Division (CSD), ORNL, to develop the control system utilizing previously developed and accepted computer programs. The control system will permit the analyst to input the minimum amount of information into SCALE. SCALE will calculate its own atom densities, Dancoff factors, material properties, etc., unless it is overridden by the user.

The SCALE control module allows the user to pick from a variety of standard criticality analytical sequences characterized by the nature of the analysis to be performed and the complexity of the system being analyzed. Each sequence would define a unique set of input parameters specified in the form of easily visualized physical descriptions and standard material specifications. This approach provides the user with a clear understanding of which functional modules are to be employed (NITAWL, XSDRN, KENO, etc.) in the analysis without having to select such modules or to provide separate (and often redundant) sets of input data.

Since January 1969, the KENO Monte Carlo program has been the primary means NRC uses to determine the subcriticality of fuel material outside a reactor. The latest improvements being made to KENO are super-

grouping (minimizes in-core storage of energy-group-dependent information) and combinatorial geometry (simplifies the description of the large, regular regions typically occurring in shipping packages).

For a number of years the NRC has utilized the Knight modified Hansen-Roach cross-section set for homogeneous systems and the 123-group GAM-THERMOS cross-section set (resonance corrected and cell weighted utilizing the NITAWL and XSDRN processing programs) for heterogeneous systems. Also, in support of this work, CSD, ORNL, has developed a 27-group standard neutron cross-section library based on ENDF/B-IV data. This library is presently being upgraded to be consistent with ENDF/B-V data. The validity of the standard neutron cross-section library has been established based on over 100 critical experiments. A separate 27-group neutron-gamma-ray coupled library, based on the 27-group neutron library, has been developed for shielding and source-term calculations.

Experimental work has been completed at Battelle Pacific Northwest Laboratories (PNL) on 2.35 and 4.3 wt% ^{235}U enriched fuel pins in water with various reflectors and neutron poison materials between simulated fuel bundles. These experiments form the basis of the applicability of the cross sections to shipping packages for LWR fuel materials.

Other work has been undertaken at PNL to attempt to find inconsistencies or possible errors in the ENDF/B Pu and stainless-steel cross-section values. This study would then recommend additional cross-section experiments or data reduction, if necessary, to resolve discrepancies.

NRC has contracted with CSD, ORNL, to provide advice and guidance for the criticality experimental program being conducted at Rocky Flats and PNL for the NRC. The work entails a continuing review of the experimental program to determine if the series of experimental configurations adequately covers the needed ranges of applicability. Also involved is the analysis of selected experiments to determine if the experimental data yield well-defined analytical models.

This experimental information is being used to determine the adequacy of the SCALE cross-section sets.

The SCALE system is being documented in accordance with the requirements of the ANSI N413-1974 Standard. This documentation is being written with the user in mind and includes benchmark calculations and discusses the range of applicability of the various programs contained in the system.

The SCALE modular computing system is currently expected to be completed in FY 1979 and to be available from the Radiation Shielding Information Center (RSIC), ORNL, for both the IBM and CDC computer systems. A limited version of the SCALE program should be available from RSIC after June 1978.

It is the intent of the NRC staff that for any package design which has been evaluated using the SCALE system (within its range of applicability and in accordance with published NRC guides) the staff will independently verify modeling, input parameters and output without performing additional analysis. This is expected to simplify the licensing process. The applicant, of course, may choose to utilize his own programs and data base. In these cases, the NRC staff would expect these programs to be shown to be adequately verified; and the NRC staff would perform independent analyses, as we currently do.

Standardization Cask Analysis and Correlation with Experiment, Robert M. Westfall (ORNL)

This paper describes a modular system of computer programs called SCALE. SCALE is an acronym for "standardized case analysis for licensing evaluation." The system is being developed to perform criticality safety, radiation shielding, and heat transfer analyses of shipping casks and other equipment and facilities associated with the nuclear fuel cycle. In this paper, emphasis is placed on the criticality safety features of SCALE and on the use of pin-lattice critical experiments in validating the SCALE system analytical sequences and data libraries.

The SCALE system consists of a driver module, control modules, functional modules, and a data base. The components of the initial version of the SCALE system are shown in Fig. 1. In the SCALE system, the user selects an analytical sequence of combined data processing and systems analysis which is characterized by the type of analysis to be performed and the geometric complexity of the system being analyzed. The user then prepares a single set of input for the control module corresponding to this standard analytical sequence. The control module input is in terms of easily visualized engineering parameters specified in a simplified free-form format. The control module uses this information to derive additional parameters and prepare the input for each of the functional modules in the analytical sequence.

An example of the simplified input for Criticality Safety Analytical Sequence 1 (CSAS1) is given in Fig. 2. The input is based on selections from a standard com-

Fig. 1. SCALE system components.

```

=CSAS1
34.6 CM RADIUS UO2(NO3)2 CRITICAL SPHERE.
25GROUPAMPX 2 2 2 MULTIREGION 1 1
SOLNUO2(NO3) 1 20.123 0.14 1 298 92234 1.04 92235 93.18
92236 0.27 92238 5.51      END
AL 2 1.0 END
SPHERICAL VACUUM REFLECTED 0
1 34.6 NOEXTERMOD 2 34.92 NOEXTERMOD      END
SZF=2 ICM=180 PTC=1-5 EPS=1-5 IUS=0      END
END

```

Fig. 2. Card images of CSAS1 input.

positions library. In the example shown, the first standard composition is a uranyl nitrate solution specified in terms of the heavy metal density, the excess acid molarity, the temperature, and the isotopic distribution of the fuel in weight percent. This example includes all the information required by CSAS1 to execute NITAWL and XSDRNPM. The control module calculates the isotopic atom densities, prepares the physics parameters required by the resolved resonance analysis, and calculates the system multiplication factor with the XSDRNPM module.

Of particular interest to this session is CSAS2, which performs criticality safety analyses on multidimensional systems using KENO-IV (Ref. 1). CSAS2 provides for automated data processing and simplified materials input specifications. However, all of the features of the three-dimensional geometry specifications have been reserved as options for the user. A combinatorial geometry version of KENO, KENO-IV/CG, and a supergroup version, KENO-V, will be incorporated into new analytical sequences of the advanced versions of SCALE.

The problem-dependent data processing performed in SCALE primarily involves the calculation of the resonance shielding for the system being analyzed. For the Hansen-Roach² 16-group library, the BONAMI module automates this task through the use of the Bondarenko³ shielding factor method. For the cross-section libraries based on ENDF/B, the NITAWL module performs resonance shielding with the Nordheim⁴ integral method. Both modules apply a Dancoff factor supplied by the SUPERDAN⁵ package.

The validation of these data libraries through the calculation of critical experiments has been an ongoing effort. The validation for the uranium-fueled systems was reported⁶ at the last ANS meeting. The cross-section validation for mixed-oxide and plutonium-fueled systems has been completed. Particularly valuable in the cross-section validation for structural and neutron-absorbing materials have been the experiments performed at Pacific Northwest Laboratories.⁷

The standardization provided by the SCALE system arises from the use of well-established functional modules and data libraries in standard analytical sequences. The unified input specified in terms of easily visualized engineering parameters reduces the occurrence of input errors and lends itself to easy verification. Through the selection of standard analytical sequences, the SCALE system user can perform sophisticated state-of-the-art analysis in a commonly understood, well-documented manner.

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Shipping Cask Experimental Program and Data—Battelle-Pacific Northwest Laboratories, B. M. Durst, S. R. Bierman, E. D. Clayton (BNWL)

A research program was initiated in 1976, at the Battelle-Pacific Northwest Laboratories, Critical Mass Laboratory, to provide experimental criticality data that can be used in criticality assessments of LWR-type fuel element shipping casks and storage facilities. The intent of this paper is to provide an overview of this experiment program at Battelle, both in surveying the experiments performed to date and describing future work.

DESCRIPTION OF 4.29 wt% ^{235}U ENRICHED UO_2 RODS

CLADDING: 6061 ALUMINUM TUBING SEAL WELDED WITH RUBBER END CAPS
TOTAL WEIGHT OF LOADED FUEL RODS: 1203.38 ± 4.12 g UO_2 /ROD

LOADING:

ENRICHMENT - 4.289 ± 0.006 w/o U-235
FUEL DENSITY IS 94.9 ± 0.55% OF THEORETICAL DENSITY
URANIUM IS 82.055 ± 0.261 w/o OF TOTAL FUEL COMPOSITION

DESCRIPTION OF 2.35 wt% ^{235}U ENRICHED UO_2 RODS

CLADDING: 6061 ALUMINUM TUBING SEAL WELDED WITH A LOWER END PLUG OF 5052-H32 ALUMINUM AND A TOP PLUG OF 1100 ALUMINUM

TOTAL WEIGHT OF LOADED FUEL RODS: 917 gm (AVERAGE)

LOADING:

825 gm OF UO_2 POWDER/ROD, 726 gm OF U/ROD, 17.08 gm OF U-235/ROD
ENRICHMENT - 2.35 ± 0.05 w/o U-235
FUEL DENSITY - 9.20 mg/mm³ (84% THEORETICAL DENSITY)

Fig. 1.

TABLE I
Results of Benchmark Calculations for Subcritical Clusters of UO_2 Rods Separated by Water and Neutron Absorber Plates*

NUMBER OF FUEL ROD CLUSTERS	CLUSTER CONFIGURATION ^a (25.4 mm SQ PITCH FOR 4.29 w/o RODS AND 20.32 mm SQ PITCH FOR 2.35 w/o RODS) NUMBER OF RODS	ABSORBER MATERIAL	ABSORBER PLATE THICKNESS (mm)	DISTANCE ^c FROM ABSORBER PLATE TO FUEL CLUSTERS (mm)	CRITICAL SEPARATION ^d BETWEEN FUEL CLUSTERS (mm)	a,e,f K _{eff}
3	U(4.29) ₂ RODS 15 x 8	WATER	---	---	117.2 ± 0.10	1.004 ± 0.004
3		304L STAINLESS STEEL	3.02 ± 0.13	32.77 ± 0.32	97.6 ± 0.3	1.010 ± 0.004
3		304L STAINLESS STEEL WITH 1.05 w/o B	2.98 ± 0.06	32.77 ± 0.32	80.8 ± 0.2	1.015 ± 0.004
3		ALUMINUM	6.25 ± 0.01	32.77 ± 0.32	107.7 ± 0.5	1.017 ± 0.005
3		CADMIUM	0.610 ± 0.025	32.77 ± 0.32	74.2 ± 0.2	1.020 ± 0.004
3		BORAL ^g	7.13 ± 0.11	32.77 ± 0.32	67.2 ± 0.1	1.013 ± 0.004
3		COPPER	6.46 ± 0.08	32.77 ± 0.32	94.2 ± 0.2	1.004 ± 0.004
3	U(2.35) ₂ RODS 20 x 17	WATER	---	---	119.2 ± 0.4	1.003 ± 0.004
3		304L STAINLESS STEEL WITH 1.05 w/o B	2.98 ± 0.06	6.45 ± 0.06	75.6 ± 0.2	---
3		CADMIUM	0.610 ± 0.025	6.45 ± 0.06	67.4 ± 0.6	---
3		BORAL ^g	7.13 ± 0.11	6.45 ± 0.06	63.4 ± 0.2	---

* ERROR LIMITS SHOWN ARE ONE STANDARD DEVIATION

^a A SMEARED ROD IN WATER GEOMETRY USED IN CALCULATIONS

^b PLATE HEIGHT OF 915 mm AND WIDTH OF 356 mm

^c PERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARY OF THE CENTER FUEL CLUSTER AND THE NEAR SURFACE OF THE ABSORBER PLATE

^d PERPENDICULAR DISTANCE BETWEEN THE CELL BOUNDARIES OF ADJACENT FUEL CLUSTERS

^e KENO IV CALCULATIONS USING 18 GROUP AVERAGED CROSS SECTIONS FROM EGGNIT PROCESSED DATA

^f ≥ 80 BATCHES, 500 NEUTRONS/BATCH

^g INCLUDES 1.02 mm THICK CLADDING OF 1100 Al ON EITHER SIDE OF THE B₄C-Al CORE MATERIAL

Benchmark calculations performed on both fuel types utilizing the KENO IV (Ref. 3) computer code with cross sections generated by EGGNIT (Ref. 4) and THERMOS (Ref. 5), have on the average been about 1% conservative in terms of k_{eff} , regardless of the neutron absorber placed between clusters.

Additional experiments currently planned in this program are concerned with measuring the effect that biological shielding materials have on the criticality of water-flooded fuel clusters. In these experiments the critical separation between fuel clusters is to be observed as a function of the distance of the fuel clusters from shielding walls of lead and depleted uranium (see Fig. 2).

On completion of this research program, well-defined criticality data will be available for validating computations on shipping casks and similar type systems. These data will encompass both undermoderated and well-moderated LWR-type fuel systems and will cover the neutron poison materials commonly employed in these systems.

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Critical Experiments with Lattices of 4.75 wt% ^{235}U Enriched UO_2 Rods in Water, J. C. Manaranche, D. Mangin, L. Maubert, G. Colomb, G. Poullot (CEA-France)

The development of PWR and BWR reactors in the world has necessitated critical experiments on this type of fuel for the validation of the calculation methods used and to optimize the various related criticality problems: fabrication, dry storage, in-pool storage, transport, dissolution, etc.

At the criticality facility of Valduc (near Dijon), which is in the CEA group, critical approaches have been realized with 4.75 wt% ^{235}U enriched UO_2 rods in lattices placed in a cubic vessel. Pure water is used both for the reflector and the moderator; it is introduced at the bottom of the vessel; the height of this water is the critical approach parameter.

The experimental diagram is represented in Fig. 1. The characteristics of the fuel rods and structural materials are given:

Fig. 1. Experimental setup.

Fuel

diameter: 7.90 ± 0.02 mm
 height: 900 ± 1 mm
 material: UO_2
 enrichment: 4.742 ± 0.001 wt% ^{235}U
 O/U: 2.0035
 density: 10.38 ± 0.04 g/cm³
 boron equivalent of impurities: $5.19 \cdot 10^{-6}$ g/cm³

Cladding

diameters: 9.4/8.2 mm
 material AG 5 density: 2.70
 (Al 98.85—Mg 0.5—Si 0.43—Fe 0.22 wt%)

Plugs

diameter: 9.4 mm
 material: AG 5

Lattice grids and support

plate: thickness 2.5 mm
 holes: 10-mm diam
 material: steel Z 3 CN 18/10 density: 7.90 g/cm³
 (Fe 71.39—Cr 19.16—Ni 9.44 wt%).

The experimental results obtained with regular lattices in water are presented in Table I.

The three triangular pitches, 13.5, 17.2, and 22.6 mm, present the same cell cross sections as the square pitches 12.6, 16.0, and 21.0 mm, respectively.

Removed rods effect: various removed rods configurations were investigated for square lattices 22×22 and 21×21 rods with an undermoderated 12.6-mm square pitch. Some experimental results are given in Table II. Numbers are given to the rows and columns of the lattice; the position of the removed rods can thus be precisely described (see table).

The four first experiments were carried out with 25 rods removed. They emphasize the considerable influence of hole position on critical height. The last experiment shows the advantages of using a full regular lattice with square cells having eight rods and a central hole.

Future Experimental Program

From now up until April 1979, the program consists of studies of the gadolinium poisoning effect, assemblies interaction, and screens studies. The experimental results and their interpretation will be published.

Spent-Fuel Storage Benchmarking Experiments, M. N. Baldwin, G. S. Hoovler, W. D. Bromley, F. M. Alcorn (B&W)

In July 1977, the Babcock & Wilcox Company was awarded a contract by the U.S. Department of Energy (DOE) to conduct critical experiments to support benchmarking of underwater close-spaced storage calculations for LWR nuclear fuel assemblies. The contract is being administered through the Savannah River Operations Office.

TABLE I

Experimental Results with Lattices of UO_2 Rods in Water

PITCH (mm)		NUMBER OF RODS ^a	WATER : CRITICAL HEIGHT ^b (mm)
SQUARE	12.6	22 x 22	906.9
	16.0	16 x 17	735.3
	21.0	15 x 15	779.8
TRIANGULAR	13.5	547	609.3
	17.2	271	680.6
	22.6	217	795.5

^aSquare or rectangular cross section for square pitches, hexagonal cross section for triangular pitches.

^bLevel zero: bottom of fuel part.

TABLE II

Experimental Results with Removed Rods

No.	GRID	NUMBER OF HOLES	ROW ^a	COLUMN ^a	WATER : CRITICAL HEIGHT (mm)
1663	22 x 22	25	1-6-11-16-21	1-6-11-16-21	795.6
1670	22 x 22	25	6-9-12-15-18	5-8-11-14-17	697.1
1674	22 x 22	25	7-9-11-13-15	7-9-11-13-15	687.5
1680	22 x 22	25	1-3-5-7-9	1-3-5-7-9	795.4
1668	21 x 21	49	2-5-8-11-14-17-20	2-5-8-11-14-17-20	890.7

^aFor instance, in 1663 experiments the holes are at the positions (1-1); (1-6) ... (6-1); (6-6) ... (21-22).

The need for this program was brought about by the many delays in increasing nuclear fuel reprocessing capability in the U.S. These delays have greatly increased the amount of fuel to be stored and have placed a premium on optimum use of storage capacity.

Experimental measurements are being conducted with fuel rods in the configuration of LWR assemblies in close proximity water storage. The assemblies, or clusters, are square with 14 fuel rods on each side. Nine of these fuel rod clusters are arranged in a 3×3 critical array. Both the spacing between the fuel clusters and the intervening material are being varied to provide a variety of benchmark loadings. All pertinent data will be documented in the final report in sufficient detail to permit use of these experiments in validating calculational methods.

The facility, shown in Fig. 1, is located at B&W's Lynchburg Research Center. The critical assembly is located within an aluminum-walled tank 152.4 cm in diameter, 198 cm high, and 1.27 cm thick. The fuel rods are 156 cm long by 1.21-cm-o.d. aluminum tubes filled with sintered pellets of 2.46% enriched UO_2 . Pin pitch is 1.636 cm. Moderator level in the core tank is read at the control console by a probe accurate to 0.2 cm.

The experiments are separated into two parts. In the first part of the program, the clusters are separated by water only, or water and B_4C filled pins. Top and bottom egg-crate aluminum grids hold the fuel rods and B_4C pins in place. Critical configurations will be obtained with the clusters separated by an integral number of pin pitches. Preexperimental calculations indicate that this array can be made critical with up to four pin pitch separation or with up to 86 B_4C pins symmetrically located between the nine clusters. These calculations were done with the Monte Carlo code KENO using a 123-energy-group cross-section set.

In the second part of the program the clusters are again separated by an integral number of pin pitches. Poison metal sheets are placed between the clusters. Since the sheets cannot be readily changed with the existing top grid in place, nine 14×14 rod assemblies have been constructed which can be moved as units. The continuous top grid has been eliminated for this part of

Fig. 1. Vertical section through core tank.

Fig. 2. Nine arrays with two pin pitch separation.

the experimental program. To assure that a number of critical configurations can be obtained, the boron content specified for the poison sheets is based on preexperimental calculations.

A plan view of the nine clusters is presented in Fig. 2. It is anticipated that the results of approximately ten benchmark loadings will be reviewed during the oral presentation.

Criticality Aspects of Fuel Storage and Transport in the UK, D. W. Anderson, D. J. Western (CEGB-UK), G. Chatburn (BNFL-UK)

Apart from experience on prototype plants, criticality aspects of the storage and transport of commercial irradiated fuel have only become important in the United Kingdom with the introduction of the Commercial Advanced Gas Cooled Reactors (CAGR). In addition to this fuel from the domestic program, LWR fuel from stations abroad is transported to and stored at British Nuclear Fuels Windscale Works.

CGAR fuel takes the form of a 36-pin cluster of uranium oxide enriched up to 3% surrounded by a graphite sleeve. Each fuel element is 1 m in length and contains around 50 kg of uranium.

Within the Central Electricity Generating Board (CEGB), the "double contingency" principle is adopted in considering criticality in the CAGR fuel routes. Thus, at least two independent low-probability events must occur before criticality can be reached. For clearances based on calculations a criterion is adopted that if one such event takes place, the calculated K_{eff} plus an appropriate allowance for uncertainties should be <0.95 . This uncertainty allowance includes a systematic component and a random component, statistics plus data, etc., taken at the three standard deviation levels. Two computer codes are employed, the lattice code WIMS (Ref. 1) for survey work and the Monte Carlo code MONK (Ref. 2) for

definitive calculations. With these codes the criterion generally leads to the requirement for the calculated K_{eff} not to exceed about 0.9.

Fuel discharged from the CAGRs is stored in cooling ponds prior to transport to the reprocessing plant. Elements are placed in skips made in stainless-steel bearing 0.6 wt% boron. Each skip holds 20 elements on a 5×4 square array with a 264-mm center spacing. Skips may be stacked 2 high and in large closely spaced arrays. Skips are moved between element loading, storage, and flask loading positions. Calculations indicated that at 3% enrichment only 6 undamaged elements in close proximity in water would be critical. This compares to the 20 elements contained in a skip. Thus, a handling accident involving a single skip could have led to a potentially hazardous situation. It was decided that, in the particular context of the power station ponds, the most appropriate course of action would be to load the pond water with 1000 ppm of natural boron. Under these conditions any credible rearrangement of the 20 elements from a skip yields a calculated K_{eff} below 0.9, meeting our criterion. Under normal geometric conditions, K_{eff} is calculated to be below 0.9 without dissolved boron.

CAGR fuel is transported in water-filled cuboidal flasks made of steel with a lead liner. One skip of elements is transported per flask. In accordance with international recommendation no credit is taken for the presence of the dissolved boron in the flask water. Survey calculations were performed using WIMS to determine the effect on K_{eff} of changes in fuel configuration. These showed that significant gains in reactivity were theoretically possible under accident conditions. Drop tests on dummy fuel indicated that while minimal distortion occurred following an end drop significant compaction could occur in a side drop. To date, no drop tests have been performed on the skip itself and we therefore assume that optimum radial compaction of both skip and fuel could occur in the event of a drop onto the side of the flask. Calculations were also carried out on the reflection characteristics of the flask and these showed a maximum reactivity gain of about 3% compared to water reflection. On the basis of these survey calculations a detailed MONK calculation on the evaluated most-reactive credible (MRC) conditions was carried out. This yielded a value of K_{eff} just below 0.9, meeting the reactivity criteria outlined above.

BNFL will be storing CAGR irradiated fuel in water ponds in similar configurations to those used by CEGB in storage and transport except that each 20 element skip will be completely contained in a stainless-steel container. These containers will be positioned in a pond matrix several units high with spacing between each stack of containers. In this case there will be no boron loading of the pond water.

Some CAGR fuel pins, after removal from the fuel element, will be stored in 229-mm-diam canisters in the fuel element positions in the skip previously described. Studies have been made, using MONK, of the reactivity of such an assembly with varying numbers of pins in the canisters. At 3.2% enrichment, K_{eff} rose from 0.775 with 37 pins to 0.875 with 61 pins, falling to 0.767 with 127 pins. The standard deviation due to MONK statistics for the 61 pin case was ± 0.0125 , giving a K_{eff} plus 3 standard deviations of the statistics of 0.9126.

Transport and storage of LWR irradiated fuel have been carried out safely using a variety of transport flasks and holding inserts. In particular, the Excellox Flask utilized for PWR fuel transport with an insert also utilized for pond storage, originally designed for 7 fuel elements,

has been shown to be safe for only 6 such elements. This case is an example of a study that also covered BWR fuel.

Finally, one assumption always used within the UK is that the fuel is unirradiated (optimum reactivity). This has often been criticized and we note that this approach has recently been the subject of some debate in the U.S.

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Methods of Criticality Analysis for Spent-Fuel Storage Facilities, C. O. Brown, L. E. Hansen (Exxon Nucl Co)

Due to delays in reprocessing or disposal of spent fuel, increased on-site spent-fuel storage capacity is being installed at many reactors. In such high-capacity storage arrays where fuel assemblies are no longer neutronically isolated by water, the accuracy of criticality calculations becomes of primary importance. This report summarizes the methods of analysis and typical design-base assumptions employed by the Exxon Nuclear Company to assess the criticality safety of high-density spent-fuel storage arrays, and presents the results of some benchmark calculations that verify the adequacy of the analytical methods.

To assess the maximum reactivity of a spent-fuel storage array, several assumptions are made with regard to worst credible conditions, from a neutronics standpoint, that could exist in the pool. These assumptions include the following considerations:

1. maximum fuel assembly enrichment (fresh fuel)
2. bundle-averaged fuel assembly parameters
3. minimum water gap thicknesses between fuel storage cells
4. temperature variances (20 to 100°C) in pool water
5. possible moderator voiding due to decay-heat deposition
6. minimum storage cell fixed neutron poison content.

Reactivity calculations are then performed assuming an appropriate combination of these "worst" conditions to determine the maximum reactivity of the array.

The three-dimensional multigroup Monte Carlo codes KENO II (Ref. 1) and KENO IV (Ref. 2) are used to calculate array reactivities. The multigroup cross-section data (18 energy group) used in KENO II are averaged using the CCELL (Ref. 3), BRT-1 (Ref. 4), and GAMTEC-II (Ref. 5) computer codes. Multigroup cross-section data from the XSDRN 123-group data library are generated for input into KENO IV using the NITAWL (Ref. 6) and XSDRNPM (Ref. 6) codes. In both cases, these codes are used to prepare spatially cell-weighted cross-section data representative of the fuel assembly for input into KENO.

To verify the adequacy of our calculational model, a number of theory-experiment comparisons have been made. One set of critical experiments consists of small water-moderated critical arrays of fuel rods as described by Grob et al.⁷ We have evaluated several of these UO₂ rod-water lattice critical experiments using the KENO

TABLE I

Calculated K_{eff} Values for Cylindrical Rod-Water Critical Lattices
(2.70 wt% ^{235}U Stainless-Steel-Clad UO_2 Rods)

Square Lattice Spacing, in.	Moderator-to-Fuel Volume Ratio	Exp'tl. Results Cylind. Core Radius, cm	CCELL-DTF-IV Calculated Reactivity (k_{eff})	CCELL-KENO-II (18 group) Calculated Reactivity (k_{eff})	NITAWL-XSDRNP-KENO IV (123 group) Calculated Reactivity (k_{eff})
0.435	1.405	26.820	1.016	$1.008 \pm .006$	$1.007 \pm .005$
0.470	1.853	24.294	1.015	$1.014 \pm .005$	$1.013 \pm .005$
0.573	3.357	23.600	1.011	$1.003 \pm .005$	-
0.615	4.078	24.771	1.009	$1.010 \pm .005$	-
0.665	4.984	27.172	1.005	$1.005 \pm .005$	-

TABLE II

Calculated K_{eff} Values for ORNL Critical Lattices
(4.95 wt% ^{235}U Unclad Uranium Metal Rods)

Case	Lattice Number	Number of Rods	Critical Water Height Above Lattice, cm	CCELL-KENO II (18 group) $k_{eff} \pm \sigma$	NITAWL-XSDRNP-KENO IV (123 group) $k_{eff} \pm \sigma$
Rod-Water Lattice Only					
1	22	203	7.1	$0.988 \pm .006$	-
2	23	195	15.24	$0.998 \pm .006$	$0.999 \pm .006$
Rod-Water Lattice + U(0.185) Block					
3	104 (Run)	245	9.5	$1.001 \pm .006$	$0.993 \pm .006$
Rod-Water Lattice + U(0.185) Block + BORAL TM Sheet					
4	105 (Run)	324	15.24	$1.038 \pm .005$	$1.000 \pm .005$
5	-	359	11.94	$1.037 \pm .006$	-

TABLE III

Calculated K_{eff} Values for Bierman Critical Lattices
(2.35 wt% ^{235}U Aluminum-Clad UO_2 Rods)

Case	Experiment Number	Number of Fuel Clusters in Array, Fuel Rods	Critical Separation Between Fuel Clusters, cm	CCELL-KENO II (18 group) $k_{eff} \pm \sigma$	NITAWL-XSDRNP-KENO IV (123 group) $k_{eff} \pm \sigma$
Rod-Water Lattice Only					
1	002	1	-	$1.008 \pm .005$	$1.004 \pm .005$
2	014	3	8.41	$1.007 \pm .005$	$0.991 \pm .005$
Rod-Water Lattice + 304L Steel					
3	028	3	6.88	$0.994 \pm .004$	$0.997 \pm .004$
4	035	3	11.47	$0.997 \pm .005$	$1.000 \pm .005$
Rod-Water Lattice + BORAL TM					
5	020	3	6.34 ^a	$0.995 \pm .005$	$0.999 \pm .005$
6	016	3	9.03	$1.007 \pm .005$	$0.999 \pm .004$

^a6.33 cm assumed in reactivity calculation.

codes with cross-section data averaged as described above. The results of these calculations are shown in Table I.

In addition to these correlations, several benchmark calculations have been performed for critical experiments utilizing 4.95 wt% ^{235}U metal rod-water lattices⁸ and 2.35 wt% ^{235}U aluminum-clad UO_2 rod-water lattices.⁹ Results of reactivity calculations for the uranium metal rod experiments are given in Table II. Table III lists results for the aluminum-clad rod data, recently reported by Bierman et al.¹⁰ Although no effort has been made to firmly establish an actual calculational bias, the theory-experiment correlations show that the analytical methods used adequately reproduce the experimental results.

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Qualification of Computer Codes for Nuclear Reactor Analysis—II

Monte Carlo Benchmarking of Fast Reactor Computations, E. M. Gelbard, R. E. Prael,* B. G. Reynolds (ANL), invited

A Monte Carlo benchmark program should ideally contain two different components; namely (a) systematic benchmarking of other computational techniques against Monte Carlo standards, and (b) detailed and faithful Monte Carlo modeling of critical experiments. As suggested in Ref. 1, part A, for example, might consist of the following steps: (a) comparison of deterministic and Monte Carlo k_{∞} 's, as well as activation rates, both for homogeneous media and infinite lattices, (b) comparison of k_{eff} 's and fluxes, as functions of B, again for homogeneous media and infinite lattices, and (c) comparison of Monte Carlo and deterministic computational results for simplified reactor calculations. Many computations of types 1 and 2, and some of type 3, have already been reported and will be reviewed here. As for the Monte Carlo analysis of critical experiments, such analyses for fast reactors are still not very common. Recently detailed Monte Carlo critical configuration simulations have been completed for a series of safety-related critical experiments in which grossly exaggerated accident damage was modeled.² Input for the safety-related computations was so massive it was necessary to develop special input preparation routines. Monte Carlo analysis of the safety-related critical assemblies will be discussed in some detail.

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Risk Tree Analysis in Criticality Control

Risk Tree Analysis in Criticality Control, Robert Nertney, John Morfitt (EG&G Idaho), invited

The application of fault tree analysis to nuclear criticality control was discussed in an invited paper of that name by one of the present authors at the 1977 Winter Meeting of the American Nuclear Society.¹ The present tutorial session expands this material into a full special session.

The material presented in the session will be aimed at those who have had little or no previous experience with the Management Oversight and Risk Tree (MORT). The session will emphasize the usefulness and wide applicability of the MORT system rather than concentration on specialized techniques.

In its broadest sense, MORT is a formal disciplined logic that systematically relates and integrates a wide variety of safety concepts. The effectiveness of the MORT system, with over eight years of development and test behind it, has been verified in actual field trials. Its use is now mandatory within the U.S. Department of Energy for the more severe classes of accidents.

MORT can be equally useful, and even more valuable perhaps, as an analytical tool to reduce or eliminate potential sources of accidents. Such uses have been frequent but not widespread. Moreover, their application to criticality safety problems has—with certain exceptions—been neglected. Yet, criticality safety requires the kind of thorough, highly disciplined, and carefully documented analysis that can be greatly strengthened by the use of MORT techniques. In addition, the authors believe that training criticality safety professionals in the use of MORT early in their careers would not only speed up their ability to use their formal training effectively, but

would induce, at an early stage in their career, those mental attitudes and disciplines prerequisite to effective performance.

MORT is based on the premise, among others, that it is possible to construct a fault tree of such generality, breadth, and controlled redundancy that it can be used effectively—with minor adjustments—in every investigation of actual or potential accidents. The new professional does not need to be trained how to construct fault trees—only how to use one particular, readily available version that has been constructed, reshaped, and checked by literally hundreds of people. Even the least experienced trainee immediately inherits this compact, pervasive technology.

The session will begin with a brief exposition on the background, justification, and philosophy of the MORT system. This will be followed by an introduction to fault trees, leading to the MORT tree as a special case. The events near the top of the MORT tree will be examined in some detail: how the logic gates operate, and therefore, how one can trace an event through the tree from top to bottom or vice versa. Enough information will be given on symbols and tracking aids to enable those attending to proceed from this point by self-study (or by taking the MORT course which includes the practice of these techniques in real-life case studies). A major portion of the time will be devoted to the use of decision trees in readiness reviews, with special emphasis on the problems of nuclear criticality safety.

Finally, the session will highlight the characteristics of a modern safety program, and show how the MORT techniques described fit into it. Additional real-life examples will emphasize the points being made during the session.

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Fig. 1. Facility occupancy—use management oversight risk tree.

Soluble Neutron Absorbers and Experimental Data for Criticality Control

A Survey of Experimental Criticality Data Concerned with Soluble Neutron Absorbers, E. B. Johnson (ORNL), invited

Subcommittee 8, "Fissionable Materials Outside Reactors," of the Standards Committee of the American Nuclear Society is concerned with the development of standards that establish criteria for nuclear criticality safety in all parts of the nuclear fuel cycle except in reactor cores. This Subcommittee has established a Work Group charged with development of a standard concerned with the use of soluble neutron absorbers—or poisons—for criticality control of solutions containing ^{235}U , ^{233}U , or ^{239}Pu .

Soluble neutron absorbers have both actual and potential use in the chemical processes that are a part of the nuclear fuel cycle. In particular, they are used in both chemical and electrolytic dissolvers and in storage tanks and other vessels in the process stream that are not "safe-by-shape," thereby permitting equipment to be sized more appropriately to the requirements of the process. The Idaho Chemical Processing Plant (ICPP) has been a pioneer in the use of boron and gadolinium as soluble absorbers for criticality control,^{1,2} and the Barnwell Plant was designed to use gadolinium in its process. It would be misleading not to acknowledge, parenthetically for the purposes of this paper, the use of "solid" neutron absorbers, such as borosilicate glass Rashig rings and boron-impregnated stainless-steel plates, for criticality control in appropriate areas.

It is seldom that criticality data have been developed that are directly applicable to a process under development. It is necessary, therefore, to make use of calculational methods to extend or generalize experimental data to areas of applicability. Such is the case in the development of this proposed standard. The validation of the calculational method, including the cross sections, will be done in accordance with the requirements of *American National Standard for Validation of Calculational Methods for Nuclear Criticality Safety*, N16.9-1975/ANS-8.11.² The validations necessary to the development of this standard can be divided broadly into two phases: first, the use of data from clean, simple solution critical experiments to verify the cross sections and the methods, followed by validation of methods applied to experimental data from heterogeneous systems more like those that could be encountered in various parts of the fuel cycle, such as in dissolvers. The bases for the technical specifications are, therefore, experimental data, generally extended by means of validated computations to areas beyond those actually investigated.

There is a paucity of experimental data in both homogeneous and heterogeneous categories. For the ^{235}U and the ^{233}U clean, simple solutions, the only directly applicable data are those of Gwin and Magnuson.³ These data were generated as a part of the reevaluation of the value of η in the late 1950's. The relevant measurements were made in unreflected spherical vessels of two

diameters in which aqueous solutions of uranyl nitrate were brought to criticality, at different fissile isotope concentrations, first without additional neutron absorber and then with three different concentrations of boron, added as boric acid. The fissile isotopes were ^{233}U and ^{235}U . Because these experiments included the measurement of basic physics parameters as well as the critical dimensions, they are appropriate for the establishment of cross sections for ^{233}U , ^{235}U , and boron in aqueous solution. Staub et al.⁴ reported a careful independent analysis of these data that should prove valuable to the present investigation. Steen⁵ also recalculated the ^{233}U experiments and reported excellent agreement.

Lloyd et al.⁶ report critical data for water-reflected cylinders of plutonium nitrate to which measured quantities of gadolinium nitrate were added to increase the solution critical height. Two plutonium concentrations were used, and sufficient gadolinium was added to produce an approximately fivefold increase in critical mass in the case of the lower plutonium concentration and to approximately double the critical mass of the higher concentration. Analysis⁶ of these data indicate that the effect of gadolinium as a neutron absorber decreases significantly in the solutions of high plutonium concentration and, consequently, harder spectrum.

The experimental data base for validating calculations of lattices of fissile materials in aqueous solutions of absorbers or in solutions of mixed absorber and fissile material is somewhat more extensive. Early data were summarized⁷ for rods of low ^{235}U enrichment latticed in boronated water. Other experiments⁸ were used to validate the use of boron in zirconium-alloy and graphite fuel processes; still others⁹ were the basis for validation of the use of gadolinium in a stainless-steel process. Experiments⁹ in which, *inter alia*, lattices of fuel pins containing $\text{U}(62.4)\text{O}_2$ ^b and BeO were made critical in water, in dilute $\text{U}(93)\text{O}_2(\text{NO}_3)_2$, and in dilute uranyl nitrate solution to which boric acid had been added; these experiments were designed for direct applicability to a proposed electrolytic dissolver.

Interest in mixed oxides of plutonium and natural uranium led to the generation of data concerning the criticality of aqueous solutions of nitrates of these isotopes, with and without gadolinium; these measurements were made in a water-reflected cylinder.¹⁰ Reported in the same paper are lattices of rods containing UO_2 - PuO_2 made critical in nitrate solutions of the mixed isotopes that, in some cases, also contained gadolinium nitrate. These experiments confirmed the earlier observation that the effectiveness of gadolinium as a neutron absorber decreases with increasing plutonium concentration.

Additional data¹¹ also concern heterogeneous systems containing PuO_2 and UO_2 in fuel pins that were made critical in water and in boronated water. Reported also are similar measurements with $\text{U}(2.35)\text{O}_2$ fuel pins. Close agreement between calculated and observed eigenvalues, power distributions, and rod worths was obtained.¹²

²Available from American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60525.

^b $\text{U}(62.4)$ indicates uranium enriched to 62.4 wt% in ^{235}U .

Required for preparation of a complete standard, are data on both homogeneous and heterogeneous assemblies of the several neutron absorbers and the various fissile isotopes, singly and in combinations anticipated for the fuel cycle.

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Subcritical Limits of Boron-Poisoned ²³⁵U-H₂O Solutions, R. D. Carter (Rockwell Hanford), invited

The use of strong neutron absorbers (poisons) in fissile solutions can increase critical masses or critical dimensions with potential economic benefits from increased throughput or decreased equipment cost. A series of parameter calculations has been undertaken to define maximum subcritical values for boron-poisoned U(100)-H₂O solutions.

A number of correlations with uranyl nitrate experiments with both ²³³U and ²³⁵U have been made to prove the adequacy of the various codes. Although the range of boron and fissile concentration is not large, good agreement has been attained (Table I).

Correlations were made using the HFN diffusion theory code,¹ DTF-IV transport theory code,² and KENO

Monte Carlo code³ using 18-group cross sections obtained from the GAMTEC II code⁴ library; the KENO code using Hansen-Roach cross sections⁵; and the MONK4 Monte Carlo code using the UKAEA cross sections.⁶

Correlations were made with 11 bare sphere experiments which included five ²³⁵U-nitrate and six ²³³U-nitrate solution experiments.⁷ The spheres were either 691.9 or 1220 mm in diameter. Correlations with experiment were good with all combinations of codes and cross sections. The fissile material densities ranged from 13 to 28 g/litre and the boron densities ranged from zero to 0.23 g/litre.

In addition to the correlations, the k-effective of the two experiments with the maximum amount of boron for each fissile material (experiments 4 and 9) was calculated without the boron. These showed that the boron had a maximum Δk value of -0.134 for the ²³⁵U solution and -0.078 Δk for the ²³³U solution. The large fractional Δk , combined with the essentially identical k-effective accuracy for both poisoned and unpoisoned experiments indicates that the poison effect was satisfactorily calculated.

The available poison experiments had a limited range of fissile densities. However, earlier calculations⁸ showed that the DTF-IV and GAMTEC II combination provided comparable accuracy in correlation with unpoisoned solution experiments with U(93.2)O₂F₂ at 831 g ²³⁵U/litre, and ²³³U nitrate and fluoride solutions up to 160 g ²³³U/litre. Combined with the current poisoned solution correlations, therefore, the poisoned solution

Fig. 1. Effect of boron on subcritical mass limits for spheres of U(100)-H₂O solutions. Full water reflection. DTF-IV, GAMTEC II calculation.

TABLE I
Uranyl Nitrate Experiments, Bare Spheres*

Exp. #	U,g/litre	B,g/litre	HFN GAMTEC II	DTF-IV GAMTEC II	KENO-II GAMTEC II	KENO-II Hansen-Roach	MONK4
1	20.12	0.0	0.9941	1.0056	0.9963 $\pm$.0065	1.0050 $\pm$.0077	0.9768 $\pm$.0088
2	23.53	0.094	0.9948	1.0031			0.9841 $\pm$.0078
3	26.77	0.186	0.9925	1.0015	0.9978 $\pm$.0065	1.0082 $\pm$.0057	0.9732 $\pm$.0086
4	28.45	0.230	0.9942	1.0028			0.9894 $\pm$.0084
10	15.15	0.0	0.9978	0.9991	0.9988 $\pm$.0043		0.9900 $\pm$.0082
Mean			0.9947 $\pm$.0009	1.0024 $\pm$.0011	0.9976 $\pm$.0033	1.0066 $\pm$.0047	0.9827 $\pm$.0037
5	17.14	0.0	1.0054	1.0145	1.0099 $\pm$.0058	1.0086 $\pm$.0054	1.0076 $\pm$.0076
6	17.86	0.024	1.0062	1.0162	1.0254 $\pm$.0061		0.9894 $\pm$.0080
7	18.53	0.046	1.0062	1.0144			0.9899 $\pm$.0074
8	19.18	0.069	1.0064	1.0147	1.0295 $\pm$.0062	0.9940 $\pm$.0063	0.9814 $\pm$.0082
9	19.82	0.091	1.0059	1.0137			0.9959 $\pm$.0088
11	13.15	0.0	1.0050	1.0064	1.0086 $\pm$.0037		1.0095 $\pm$.0097
Mean			1.0059 $\pm$.0002	1.0133 $\pm$.0014	1.0183 $\pm$.0027	1.0013 $\pm$.0028	0.9956 $\pm$.0034

KENO calculations used 10,000 neutron histories, MONK4 calculations used 25,000.

*From *Nucl. Sci. Eng.*, pp. 263-274 (Dec. 1968). Experiments 1, 2, 3, 4, and 10 are ^{235}U at 93.2 wt% of total uranium; experiments 5, 6, 7, 8, 9, and 11 are ^{233}U at 97.7 wt% of total uranium.

calculations can be made with some confidence over a large range of uranium and boron densities.

Based on the proven accuracy of the above correlations, the DTF-IV code with GAMTEC II cross sections was used to calculate a series of maximum subcritical (k -effective equal to 0.95) parameters for $\text{U}(100)\text{-H}_2\text{O}$ solutions for uranium densities ranging from 20 to 1000 g/litre and boron densities ranging from 0 to 4 g/litre. The $\text{H}/^{235}\text{U}$ values were calculated for various uranium densities using the relationship^a

$$\text{H}/^{235}\text{U} = \left(\frac{26,089}{\text{g } ^{235}\text{U/litre}} - 1.3804 \right)$$

The effect of changes in both ^{235}U and boron densities on the mass of uranium are shown in Fig. 1.

It is common practice to use metal-water parameters to obtain limiting values for other solutions such as uranyl nitrate, since this is normally conservative. One should be aware that this may not be the case with poisoned solutions, since the worth of the poison material is at least partly dependent on the H/U ratio. Therefore, it is possible that at high uranium densities, a smaller critical mass could occur at a given uranium density for a nitrate solution than for a metal-water solution, since the nitrate solution will have a lower H/U ratio for a given uranium density.

The relationships shown in Fig. 1 provide a guide for determining the minimum amount of boron required for a subcritical system. Additional restrictions may be necessary to provide a proper safety margin. No attempt has been made to determine the practical limits on the solubility of boron. The solubility is generally low, but will vary somewhat depending on the type of solution and

the boron compound used. Boron densities significantly greater than 4 g/litre do not appear to be practical cases.

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Subcritical Limits for Homogeneous Gadolinium-Poisoned Pu-H₂O Solutions, G. W. McNair, D. R. Oden, R. C. Lloyd (BNWL), invited

Criticality safety data are needed for soluble neutron absorbers to determine effective limits for criticality control in the storage and processing of solutions containing fissile materials. Gadolinium in certain forms is an excellent soluble poison for this application because of its high neutron cross section, high solubility, and compatibility in separation processes.

A series of calculations has been performed at Battelle's Pacific Northwest Laboratories (PNL) to generate a set of curves for homogeneous, hydrogenous plutonium systems (50 to 500 g Pu/litre) containing varying amounts of a soluble gadolinium poison (0 to 5 g Gd/litre). The calculations reported here were oriented toward providing a set of data for use in a proposed standard on the use of gadolinium as a soluble absorber in nuclear criticality prevention for plutonium systems.

The calculations performed in this study utilized 18 energy-group cross sections generated by the GAMTEC-II computer code.¹ The 17 fast-group constants were averaged over a 66 fine-group slowing-down spectrum using the B₁ approximation to the Boltzmann transport equation. The thermal-group data ($E < 0.683$ eV) were averaged over a Wigner-Wilkins spectrum. The fine-group cross-section data were obtained from the GAMTEC library¹ currently in use at PNL. The effective multiplication factors and biases were calculated by DTF,² a multigroup transport theory code, and KENO,³ a multigroup Monte Carlo code.

Results from the GAMTEC-II-DTF calculational scheme were compared with curves ($K_{eff} = 0.95$) presented in TID-7016 (Ref. 4) for plutonium metal/water systems. This comparison is shown in Fig. 1.

Fig. 1. Comparison of calculated dimensions for subcritical systems of Pu(100)-H₂O solutions (full H₂O reflection).

TABLE I
Dimensions Calculated to Yield $K_{eff} = 0.95$ for Infinite Slabs and Cylinders
[Homogeneous, Pu(95,5) Metal/Water Systems with Full Water Reflection]
(GAMTEC-II-DTF-IV)

Plutonium Concentration (g/litre)	Gadolinium Concentration (g/litre)	Infinite Cylinder Radius (cm)	Infinite Slab Thickness (cm)
50	0	8.869	7.756
	1	24.868	28.060
	2	a	a
	5	a	a
70	0	8.635	6.976
	1	15.076	15.386
	2	a	a
	5	a	a
100	0	8.523	6.466
	1	11.335	10.582
	2	16.380	18.478
	5	a	a
200	0	7.779	6.020
	1	8.523	6.960
	2	9.904	8.670
	5	13.753	13.566
350	0	7.797	5.908
	1	7.690	5.806
	2	8.281	6.566
	5	9.762	8.458
500	0	7.785	5.848
	1	7.379	5.388
	2	7.719	5.836
	5	8.576	6.948

a Infinite Systems.

Fig. 2. Effect of gadolinium on subcritical mass limits for spheres of Pu(95,5)-H₂O solutions (full H₂O reflection).

Additional code validations, against a series of critical experiments performed at PNL's Critical Mass Laboratory for gadolinium-poisoned plutonium-nitrate systems, show a maximum nonconservative bias of 5.2% (at a 95% confidence level) in calculating the gadolinium-poisoned systems relative to the unpoisoned systems. The estimate of the calculational bias due to the presence of gadolinium was obtained by comparing the KENO-generated K_{eff} values for the gadolinium-poisoned vs non-poisoned critical experiments.

Correcting for the bias for gadolinium-poisoned systems, a series of curves ($K_{eff} = 0.95$) for spheres, infinite slabs, and cylinders has been generated for varying concentrations of gadolinium poisons. The curves in Fig. 2 represent poisoned and unpoisoned, homogeneous, plutonium metal/water spheres containing 5 wt% ²⁴⁰Pu with full water reflection. The anomalous behavior of the gadolinium-poisoned systems (dotted lines in Fig. 2) at high plutonium concentrations reflects the overly conservative nature of the bias correction factor.

The results of these calculations show the overall effectiveness of gadolinium as a soluble neutron absorber over a range of plutonium concentrations. The solubility and stability of gadolinium over the range of concentrations used in this study has been demonstrated in previous plutonium critical experiments over a one-month-long test period.⁵

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The Effect of Soluble Neutron Absorbers in Latticed Assemblies, R. C. Lloyd, B. M. Durst, E. D. Clayton (BNWL), invited

Experimental criticality data are needed for soluble neutron absorbers (poisons) to accurately determine their effectiveness in the criticality prevention and control of solutions bearing fissile material.¹⁻³ These data are needed to validate criticality control calculational techniques that are used in the analysis of poisoned systems. This paper summarizes the results of a series of experiments performed at the Battelle-Pacific Northwest Critical Mass Laboratory, on water-flooded uranium rod lattices which were poisoned with soluble neutron absorbers. The poisons utilized, B, Cd, and Gd, were chosen due to their large neutron-absorption cross sections. The experiments were oriented toward providing data on the effectiveness of these materials as soluble absorbers for use in nuclear criticality prevention. Data on integral critical experiments are provided against which calculational techniques can be checked.

The experiments on lattices of fuel rods immersed in water include measurements at each of three different lattice spacings with soluble neutron absorbers. The center-to-center fuel rod spacings were 22.9, 27.9, and 33.0 mm, respectively. UO₂ pellets of 4.3 wt% ²³⁵U enrichment were housed in stainless-steel tubes with a cladding o.d. of 14.4 mm (cladding thickness of 0.813 mm). A detailed description of the fuel rods is shown in Fig. 1.

The critical number of fuel rods was determined at various concentrations of gadolinium, cadmium, and boron in the water moderator with the 22.9-mm lattice, and with gadolinium and boron in the 27.9- and 33.0-mm lattices. Soluble poisons were not intermixed in any of the experiments.

Analysis of the data for the 22.9-mm lattice spacing shows that the critical number of fuel pins increased from about 223 in water without neutron poison to about 422 in water containing 0.21-g Gd/litre. 392 rods were required for 1.1-g Cd/litre, in water, and 433 rods for ~0.6-g B/litre in water.

* MEASURED FROM BOTTOM OF END SPADE TO BOTTOM OF TOP SPADE

Fig. 1. Description of 4.3 wt% ²³⁵U enriched fuel rods.

TABLE I
Effect of Neutron Absorber on Criticality of Heterogeneous Lattice Assembly

Experiment No.	Hexagonal Lattice Pitch (mm)	Number of Fuel Rods Required For Criticality	Neutron Absorber ^a Concentration (g/litre)	Total NO ₃ (g/litre)	Excess Acid Molarity M	K _{eff} ^{b,c}
001	22.86	223.3	-0-	-----	-----	1.010 ± 0.008
002	22.86	289.8	0.2307 B	-----	-----	1.003 ± 0.007
003	22.86	363.0	0.4514 B	-----	-----	0.992 ± 0.006
004	22.86	433.2	0.6053 B	-----	-----	1.003 ± 0.006
006	22.86	282.6	0.4293 Cd	0.722	0.004	1.013 ± 0.006 ^d
007	22.86	391.7	1.060 Cd	1.418	0.004	1.021 ± 0.005 ^d
009	22.86	283.4	0.0722 Gd	0.230	0.0023	1.005 ± 0.007
010	22.86	349.5	0.145 Gd	0.398	0.0036	0.995 ± 0.008
011	22.86	422.4	0.213 Gd	0.527	0.0044	1.016 ± 0.006
012	27.94	166.6	-0-	-----	-----	0.997 ± 0.005
013	27.94	225.6	0.158 B	-----	-----	1.006 ± 0.006
014	27.94	349.9	0.380 B	-----	-----	1.005 ± 0.006
016	27.94	232.3	0.0547 Gd	0.1927	0.0021	0.994 ± 0.008
017	27.94	341.8	0.1169 Gd	0.2707	0.0021	1.006 ± 0.006
018	33.02	187.4	-0-	-----	-----	1.024 ± 0.005
019	33.02	234.0	0.0643 B	-----	-----	1.012 ± 0.005
020	33.02	434	0.2154 B	-----	-----	0.992 ± 0.005
021	33.02	327.5	0.1507 B	-----	-----	1.013 ± 0.006
023	33.02	252.7	0.0257 Gd	0.1571	0.00204	1.009 ± 0.007
024	33.02	316.8	0.0440 Gd	0.1805	0.00207	1.016 ± 0.007

^aB in the form H₃BO₃, Cd in the form Cd(NO₃)₂ · 4H₂O, Gd in the form Gd(NO₃)₃.

^bCalculated using EGGNIT/ENDF-B-IV and THERMOS/ENDF-B-III.

^cThe bottom fuel rod end spade was modeled as part of the 304L stainless-steel support plate and top spades were modeled as homogenized 304L stainless steel in the water region above the fuel. The top lattice plate, made of Lexan, was considered to be water equivalent.

^dCalculated using EGGNIT/ENDF-B-IV and TEMPEST/ENDF-B-IV.

For the 27.9-mm lattice, the critical number of fuel pins varied from about 167 in water without neutron poison to about 342 in water containing 0.12-g Gd/litre and 350 in water containing 0.38-g B/litre.

The critical number for the 33.0-mm lattice varied from 187 in water to about 317 in water containing 0.044-g Gd/litre, and 434 in water containing 0.22-g B/litre.

For each of these critical system experiment determinations, *k_{eff}* values were computed to check how standard calculational techniques agree with experimental results. *k_{eff}* values were calculated using the KENO IV computer code.⁴ The calculations were made using 17 epithermal broad-group cross sections, generated using the EGGNIT code⁵ and FLANGE⁶-ETOG⁷ processed ENDF/B-IV data, and a single thermal group generated using the THERMOS code⁸ with ENDF/B-III library. All rod systems were modeled in a "smeared rod" geometry; i.e., cell-averaged cross sections were generated for fuel rods in water and these cross sections in turn were used across the fuel region in the model. A $\frac{1}{4}$ -in. Lexan top lattice plate which supported fuel during the course of the experiments was modeled as water, while a bottom $\frac{3}{8}$ -in. stainless-steel lattice support plate which the fuel rods rested upon was modeled exactly. The results of these calculations as well as all pertinent experiment data are summarized in Table I. On the average, calculations were ~1% high in terms of *k_{eff}*.

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Critical Experiments with Fast Test Reactor Fuel Pins in Water, B. M. Durst, S. R. Bierman, E. D. Clayton (BNWL)

Criticality safety evaluations for handling mixed Pu-U oxide-type fuel elements depend heavily on computational analysis with experimentally validated computer codes and cross-section data. A series of critical experiments has been performed with fast test reactor fuel pins in water at the Battelle-Pacific Northwest Critical Mass Laboratory in support of the Advanced Fuel Recycle

(a) REFERENCED DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY STANDARD

(b) 261Am CONTENT OF TOTAL PLUTONIUM WAS MEASURED TO BE 0.13 Wt% DURING APRIL AND MAY OF 1972

(c) 35.6 mm FOR TYPE 3.1 FUEL PINS

Fig. 1. Simplified description of fast test reactor fuel pin.

Program (AFRCP) for use in these validations. The results of these experiments, covering a wide range of water-to-fuel moderation, are presented in this paper.

The experiments were carried out in a water-flooded rectangular tank 1.8 m wide × 3 m long × 2.1 m deep, which was provided with lattice plates and the required control and safety devices. A description of the two types of fuel rods used is given in Fig. 1. The fuel region of the rod is ~914 mm long and about 5 mm in diameter, located near the bottom of the fuel rod. The fuel region is reflected above and below by a natural UO₂ insulator pellet and an Inconel-600 cylindrical piece. The remaining portion of the rod is composed mostly of incidental stainless-steel hardware as shown in the figure.

TABLE I
Experimental Criticality Data Water-Flooded
Type 3.2 FTR Fuel Pins

Square Lattice Pitch mm	Water-To-Fuel (Volume Ratio)	Lattice Width (Fuel Pins) (mm)	Critical Number of Fuel Pins
7.696	1.69	36 277	1037 ± 1 ^a
9.677	3.49	28 271	605 ± 1
12.446	6.68	18 224	279 ± 0.6
15.392	10.96	18 277	205 ± 0.5
19.355	18.14	14 271	162 ± 0.5

^aIncludes 65 type 3.1 FTR fuel pins on 15.392-mm square lattice pitch positioned on either side (36 on one side and 29 on the other) of a 27 × 36 fuel pin array of type 3.2 FTR fuel pins at 7.696-mm square lattice pitch.

The fuel pins were latticed in regular-spaced square configurations at lattices pitches of 7.696, 9.677, 12.446, 15.392, and 19.355 mm. These center-to-center fuel pin spacings correspond to water-to-fuel volume ratios of 1.69, 3.49, 6.68, 10.96, and 18.14, respectively, and cover the neutron moderation range from near optimum with respect to minimum critical mass to undermoderated.

The critical assemblies were reflected on all sides by at least 150 mm of water. The pins were immersed in water; however, only to a height of 150 mm above the top of the fuel region of the rod. Consequently, the top reflector region above each fuel rod was perforated by the fuel pin hardware projecting through the top reflector region. High-purity deionized water (500,000-Ω resistance minimum) was used in all experiments.

The results obtained for these five different lattices are presented in Table I. The critical number varied from a low value of 162 pins at a pitch of 19.355 mm to a value of 1037 pins at a 7.696-mm pitch. At all except the 12.446-mm lattice pitch experiment, the critical assembly width was ~270 mm. Except for the 7.696-mm lattice, the critical assemblies presented contained only type 3.2 fuel pins. At the 7.696-mm lattice spacing, the number of fuel rods required for criticality exceeded the number of type 3.2 rods available for measurement. Consequently, some type 3.1 rods were used.

A series of calculations also was performed comparing the experimental results with computer models of the critical configurations. As a result of these experiments, benchmark criticality data now exist for water-flooded arrays of fast test reactor-type fuels, which can be used in the validation of computing techniques and nuclear data.

Use of Soluble Neutron Absorbers for Criticality Safety

Soluble Neutron Poisons for Industrial Criticality Control: Are We Ready?, *H. T. Kerr (ORNL)*

The continuing growth of the nuclear power industry will result in the development of large industrial facilities for recycling nuclear fuels. These facilities will necessarily have high throughput rates (5 MT/day) and large inventories of fissile materials (20 MT); therefore, nuclear criticality control will be a major consideration in their design and operation. Recycle facilities will be sufficiently complex such that optimum process variables should first be defined and then necessary criticality control options should be applied to ensure subcriticality in the facility. Several iterations through this procedure may be required to achieve a safe and functional design for some plant components. There are numerous criticality control options available to the plant designer including geometry, concentration control, fixed poisons, and soluble poisons. Each option affords some advantages and disadvantages relative to the others. Soluble poisons have long been recognized as a viable means of achieving criticality control in systems containing fissile materials. Soluble poisons are easily introduced into most fuel solutions and are effective in all vessel sizes and configurations. However, the use of soluble poisons has generally been limited to a secondary control status because operational errors or unanticipated changes in process variables could cause the loss or redistribution of neutron absorber in the solution.¹ Soluble poisons were previously approved for primary criticality control only when adequate biological shielding was available and a single-contingency principle was acceptable. The nuclear community has only recently begun to accept soluble neutron poisons for primary criticality control in large-scale remote applications. The Allied-General Nuclear Services (AGNS) plant in Barnwell, South Carolina, is designed to use dissolved gadolinium for primary criticality control in many plant components.² The Hot Experimental Facility (HEF) currently being designed for the Advanced Fuel Recycle Program (AFRP) will rely on soluble poisons for primary criticality control whenever possible.³ Due to this rapidly growing interest, a review of the many considerations associated with using soluble neutron poisons for primary criticality control is appropriate.

Many factors influence both the decision to use soluble poisons and the selection of a particular neutron absorber. These factors can be broadly categorized as technical and philosophical requirements. Technical requirements include physical, chemical, and nuclear properties of the poisoned systems; philosophical requirements include remote operations, process flexibility, contingency factors, and other options available for criticality control.

The principal technical requirements for a soluble neutron poison material are that it have a reasonably

high neutron-absorption cross section and that it be adequately soluble in the working solution. Several candidate poisons are available. Boron, cadmium, and gadolinium are commonly used absorbers, but several alternatives are available including europium, tantalum, and hafnium.

Boron compounds have low solubility relative to some cadmium, gadolinium, or europium compounds in nitric acid solutions, and would be more likely to precipitate from solution if high concentrations are required. Cadmium is a desirable poison; however, it may be volatile at temperatures encountered in the waste conversion process and would therefore require an additional process for recovery. Additional requirements for soluble poisons include chemical stability in the solution and processing behavior similar to the fissile material. Each poison's neutron absorption rate is uniquely dependent on the neutron energy spectra of the system; hence, the relative neutronic effectiveness of each poison varies from one system to another. Consequently, a mixture of poisons in a solution may be more effective than either poison alone.⁴

The principal philosophical requirements for soluble neutron poisons include the specification of a contingency policy (i.e., single or double) and the degree of inherent process complexity acceptable in an industrial plant. The guidelines provided in ANSI N16.1 specify that the double-contingency principle shall be applied in all cases except those where adequate biological shielding exists (i.e., a single contingency principle may be applied if adequate shielding exists).⁵⁻⁷ Future recycle facilities will probably be designed for remote operations; therefore, operation under a single-contingency principle appears feasible. Advocates of the single-contingency principle argue that an occasional criticality event in a fuel reprocessing facility is not technically significant if the event is self-quenching and does not result in physical damage to equipment. Proponents of a double-contingency design for recycle facilities acknowledge the safety advantages of remote operation; however, they insist that double contingency is needed to prevent possible plant shutdowns and adverse public opinion following a criticality event. The double-contingency principle is presently being applied in the design of the HEF.

Soluble poisons can have a significant impact on the complexity and flexibility of some process operations. For example, soluble poisons may either be recycled within a plant or discarded. Recycle would necessitate additional equipment, and some chemically similar fission products would unavoidably be recovered with the poisons. If soluble poisons are discarded they will ultimately appear in high-level wastes, thereby causing enormous increases in the volume of wastes to be discarded. The gadolinium concentration in the HEF dissolver may be as high as 5 g/litre, which would result in

25 kg of gadolinium in the high-level wastes for each metric ton of fuel processed.⁸ This compares with about 35 kg of fission products per metric ton of fuel. The economics of storage, shipping, and disposal of high-level wastes will be affected by the soluble poisons. A 5 MT/day plant, for example, would require about 125 kg of gadolinium per day.

The widespread use of soluble poisons for primary criticality control in large industrial recycle facilities will necessarily rely extensively on computational analyses. Validated computational methods will be utilized in the design of HEF as required in ANSI-16.9-1975 (Ref. 9). Some validation benchmark experiments with soluble poisons will be required. Also, ANSI-16.1-1975 stipulates that the concentration of the soluble poison in the solution must be verified. Instruments have been developed within the AFRP to continuously monitor the gadolinium concentration in the feed streams of the HEF to an accuracy of about 5% (Ref. 10). These instruments, coupled with adequate administrative control, can ensure the continued presence of poison in the solution.

Soluble neutron poisons provide a feasible and economical option for primary criticality control in industrial recycle facilities. Current industry standards can be satisfied, and facility designs will not be unduly restricted by criticality safety limits.

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Experience with Soluble Neutron Poisons for Criticality Control at ICPP, R. E. Wilson, S. R. Mortimer (Allied Chem)

Soluble neutron poisons assure criticality control in two of the head-end fuel reprocessing systems at the Idaho Chemical Processing Plant. Soluble poisons have been used successfully since 1964 and will be employed in the projected new head-end processes. The use of soluble poisons (a) greatly increases the process output,

(b) allows versatility in the size of fuel assemblies processed, and (c) allows the practical reprocessing of some fuels.

A soluble poison was first applied in the head-end process for highly enriched uranium-zirconium alloy fuel.¹ The dissolver was basically a 324-mm-i.d. cylindrical slotted fuel basket-charging chute enclosed in a pear-shaped dissolver shell which was 610-mm i.d. on the lower portion and 1041-mm i.d. on the top section. The shell was 2.52 m in height. This dissolver was essentially the same as used previously without soluble poison but with the appropriate safe mass loading. Use of 3.8 g/litre natural boron in the acid feed to the dissolver safely allowed a tenfold increase in the zirconium alloy fuel processing capacity. The dissolver was eventually retired after 23,000 h of operation in which there were no notable problems due to the use of soluble poison.

The replacement uranium-zirconium alloy dissolver² was longer, to allow more fuel element submergence in the acid, and with a smaller shell diameter to reduce the amount of borated liquids necessary for operation and cleanout. This dissolver has an inner cylindrical and slotted 330-mm-i.d. charging chute in a cylindrical 483-mm-i.d. outer vessel that is 4.14 m high. The safety limit for all fluids entering the dissolver is 3.6 g/litre boron. To allow for possible deviations in the measurement systems and drift between analytical sampling periods, the standard practice is to use 3.85 g/litre boron as the lower limit. This dissolver has had 4000 successful hours of operation using soluble poisons.

The electrolytic dissolution process³ depends on soluble gadolinium for criticality safety. This system is used to process high enriched uranium clad in stainless steel. Electrolytic dissolution takes advantage of the anodic corrosion that occurs when a large electrical current is passed through the fuel elements in a corrosive environment.

The electrolytic dissolver vessel is a long slab (162 mm side to side) which tapers to a V-shape trough at the bottom. The height is 660 mm to the top of the overflow channel. The dissolver solution is circulated also through a surge tank and heat exchanger. In processing EBR-II fuel assemblies the safety limit requirement is 1.8 g/litre Gd in the acid feed to the dissolver. The failure limit for this process is considered to be 0.45 g/litre Gd in the dissolver solution. The heat exchanger cooling water loop is required to contain at least 2.6 g/litre Gd.

The electrolytic system incorporating soluble poison has operated for some 4000 h. It is doubtful that the EBR-II fuel could have been reprocessed without it.

As the poisons must be present in acceptable quantities for the several head-end processes to operate safely, a high level of assurance is necessary in the makeup and use of each solution. Poisoned solutions are required for dissolving, for heel cleanout, for cushion water, and for the closed-loop cooling systems.³

Three control methods are used on each head-end system. First, the poison is mixed according to standard operating procedures and the measurements are affirmed by the operator's supervisor. Second, the poisoned solution is stirred, sampled, analyzed, and the analysis reported while still in the mix tank. Finally, a Nuclear Poison Detection System (NPDS) must show an acceptable poison concentration before the solution can be transferred. The NPDS provides a continuous monitoring of the boron concentration in the solution and automatic alarm and shutoff of the acid feed valves should the

poison concentration drift out of specified limits. The NPDS system consists of a neutron source and two independent detection, alarm, and flow stoppage systems. The PuBe source and two BF₃ neutron detectors are immersed in the mix tank. As the poison concentration is increased, more neutrons will be absorbed by the poison and fewer neutrons will reach the detector. The two channels of the system are kept separate to avoid a common-mode failure. A reliability analysis of the NPDS has indicated an anticipated system failure each 350 years of continuous operation. This is but one of three assurance systems and the other systems would also have to fail and an abnormal fuel distribution in the dissolver system would be required before a criticality due to insufficient soluble poison could occur.

Major dissolver process upset conditions that must be considered are (a) precipitation of the poison, (b) precipitation of the uranium, and (c) dilution of the poison. Condition (a) must not be possible. The possibility of condition (b) will likely set the poisons concentration limits and must not result in an unsafe dissolver. The consideration of condition (c) will affect the poisoning of heat exchanger and dissolver jacket fluids and the allowance for water within fuel assembly storage cans.

The major disadvantage of using soluble poisons is the need for very sophisticated control systems and procedures, which require extensive checkout. The need for a poisoned primary heating and cooling system means a secondary system is needed as well. Experience has shown, however, that production enhancement more than makes up for the problems.

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Use of Gadolinium Nitrate for Criticality Control During Fuel Dissolution, R. C. Ashline, W. R. Waltz (AGNS)

Introduction

This paper describes the plans for the use of gadolinium nitrate for criticality control in the dissolver system at the Barnwell Nuclear Fuel Plant (BNFP).^a The particular aspects this paper addresses include (a) a description of the poison solution makeup system and its safety-related controls, and (b) a summary of the criticality analysis of the dissolver unit.

Poison Solution Makeup System

Gadolinium oxide is dissolved in 12 M HNO₃ to prepare a gadolinium nitrate stock solution containing 46 Gd/litre. Prior to processing each lot of fuel, a portion of this

stock solution is mixed with dilute acid in the proportions required to obtain a solution of the proper gadolinium concentration for criticality control in the dissolver. The diluted solution is temporarily stored in a holding tank with sampling capability. Prior to introducing the gadolinium solution into the dissolver system, the solution in the holding tank is sampled and analyzed to verify the gadolinium concentration.

The diluted gadolinium solution is used for several purposes. Its primary use is for controlling the reactivity of the dissolver. In addition, to alleviate concern over possible dilution of the gadolinium poison in the dissolver, the gadolinium solution is used for two other purposes: (a) for replenishing the poisoned water seal on the dissolver cover, and (b) for replenishing the dissolver heat exchanger poisoned water inventory. The latter uses prevent the possibility of a criticality accident occurring from a reduction in the gadolinium concentration in the dissolver solution due to accidental dilution.

The gadolinium content of the poisoned solution entering the dissolver is continuously verified by two independent in-line gadolinium monitors. These monitors determine the gadolinium concentration by measuring the neutron absorption characteristics of the solution. The monitors are calibrated frequently during operation.

The monitors control two shutoff valves in series—one for each monitor. Prior to processing each lot of fuel, the trip and alarm points on both monitors are set at predetermined values corresponding to the gadolinium concentration required to assure safe operation of the dissolver. Should either monitor sense a low gadolinium concentration, a shutoff valve is automatically closed, preventing transfer of any additional solution to the dissolver.

Criticality Analysis

The criticality analysis of the dissolver was made by Battelle Pacific Northwest Laboratories. The model used

TABLE I

Summary of Calculations: Gadolinium Concentrations for $k_{eff} = 0.90, 0.95$, and 1.0 in Dissolver* (Assumes No Fuel Burnup)

Initial Fuel Enrichment, wt% ²³⁵ U	Gadolinium Concentration, g/litre		
	For $k_{eff} = 1.0$	For $k_{eff} = 0.95$	For $k_{eff} = 0.90$
PWR fuel Rods ^a			
5.0	7.8	12.5	20.8
4.5	5.7	9.0	14.2
4.0	4.2	6.6	10.0
3.5	2.9	4.6	7.1
3.0	2.0	3.3	4.8
BWR Fuel Rods ^b			
3.0	2.2	3.2	4.6

*Obtained from EGGNIT¹/HFN² codes with GAMTEC³ cross sections, and corrected for gadolinium cross-section bias. (Bias determined by performing benchmark calculations using the experimental data of Lloyd et al.⁴)

^a $V_w/V_f = 0.588$: rod diameter = 1.02 cm; pitch = 1.22 cm.

^b $V_w/V_f = 0.588$: rod diameter = 1.27 cm; pitch = 1.52 cm.

^aAlthough alternative, more proliferation-resistant flow sheets are being investigated for potential use at plants such as the BNFP, this paper uses the current plant flow sheet for illustrative purposes. Changes in the dissolver system that may be required for these alternative flow sheets are not considered here.

by Battelle consists of a closely packed, triangular pitch lattice arrangement of UO_2 rods in water circumscribed by an 80.0-cm-diam circle. The fuel rod diameters correspond to the maximum sizes for PWR and BWR fuels—1.02 and 1.27 cm, respectively. The fuel height is assumed to be infinite, and full water reflection around the dissolver is assumed in the calculations.

The model employed by Battelle contains several elements of conservatism. These relate to the values for the UO_2 density and the lattice pitch used in the model, and the choice of water rather than nitric acid solution as a surrounding medium for the fuel rods.

In the model, the fuel rods consist of UO_2 at theoretical density (10.96 g/cm^3). Actual fuel rods are less dense—90 to 95% theoretical density. Therefore, the reactivity of the actual system is less than that predicted using the dissolver model.

The water-to-fuel volume ratio assumed in the model is 0.588. At this ratio, the separation distance between rods is about 0.25 cm (i.e., a very tight packing). This value is conservative for determining gadolinium requirements for criticality control since previous calculations by Battelle demonstrated that the maximum reactivity of gadolinium-poisoned UO_2 rod lattices occurs at water-to-fuel volume ratios < 0.588 for enrichments in the range considered (i.e., 3 to 5 wt% ^{235}U) and since the actual water-to-fuel ratio in the dissolver is expected to be considerably larger than 0.588.

Conservatism is also introduced into the model by neglecting the presence of HNO_3 in the dissolver solution, since the nitrogen atom in HNO_3 is a neutron absorber.

The principal results of the reactivity calculations for poisoned fuel rod lattices are presented in Table I. This table gives the concentrations of gadolinium required to limit the k_{eff} of the dissolver to 0.90, 0.95, and 1.0 for different fuel types (PWR or BWR) and enrichments.

In the dissolver application, the gadolinium concentration in Table I corresponding to the k_{eff} value of 0.90 is prepared in the poison solution makeup system. The gadolinium monitors are set to trip at the gadolinium concentration corresponding to a k_{eff} of 0.95.

As indicated, the gadolinium concentrations shown in Table I represent the poison requirements assuming no fuel burnup. If a future advancement in nondestructive assay techniques makes possible the accurate determination of fuel burnup, as opposed to calculations based on reactor performance, the gadolinium requirements shown above can then be reduced by a factor of up to 2.

It should also be noted in Table I that a cutoff exists at 3% enrichment. This is due to the lack of sufficient experimental data at enrichments below 3% for very tightly packed lattices to permit proper validation of the model. For this reason, gadolinium requirements for LWR fuels with initial enrichments below 3% ^{235}U are currently based on the data obtained for 3% enrichment.

Conclusions

In summary, use of gadolinium nitrate as a neutron poison for dissolver applications is planned for the BNFP. Using a soluble poison to control reactivity permits fuel dissolution to be carried out safely in a large (80.0-cm diam) vessel. This procedure provides significant advantages as opposed to alternative dissolver designs (e.g., smaller diameter vessel, annular vessel, etc.). The BNFP dissolver design not only allows greater

throughput capacity for a given space, but also accommodates the large fuel element end pieces, thus eliminating the need for their separate removal and disposal.

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Criticality Control by Gadolinium for U/Pu Systems, W. Heinicke, W. Thomas (GRS-Germany)

For the head end of the newly planned reprocessing facility at Gorleben, criticality safety shall be guaranteed by two principal control parameters: content of remaining fissile material resulting from a specified and measured burnup, and homogeneous poisoning by gadolinium. From a preliminary flow sheet, processing of standard fuel is characterized by:

1. an equivalent enrichment of $\leq 1.8\%$ ^{235}U (corresponding to a fuel burnup of about 30,000 MWd/MTU)
2. a gadolinium concentration of 3 g/litre.

An upper limit of 4% ^{235}U enrichment is set for head-end design.

For the dissolver vessel, a cylinder diameter of about 90 cm is anticipated with a dissolver basket of 75-cm diam. The following storage vessels for feed solution and the centrifuge are not restricted to favorable geometry. Therefore, in the head end, restrictions of the dimensions are of less concern for criticality safety.

To get a broader background for licensing, a criticality study was undertaken to answer the following questions:

1. Is each of these principal control parameters alone sufficient to maintain subcriticality?
2. What is the margin of safety?
3. What is the influence of important parameters such as moderation, HNO_3 molarity, and content of fission products?

Basis for the investigation was UO_2 fuel with an initial enrichment of 3.5% as PWR fuel elements of the Biblis type. This enrichment was chosen because a PWR fuel element enriched to 4% is not yet clearly specified. The burnup of the fuel reaches 36,000 MWd/MTU, the subsequent cooling time is one year. Remaining contents of fissile isotopes and fission products were calculated with ORIGEN.¹ Criticality calculations with HAMMER² were made for lattices and homogeneous solutions with various concentrations of gadolinium. Similar calculations were performed for mixed-oxide fuel with 6 wt% PuO_2 (isotope vector: 1.82 wt% ^{238}Pu , 58.29 wt% ^{239}Pu , 24.32 wt% ^{240}Pu ,

11.61 wt% ^{241}Pu , 3.96 wt% ^{242}Pu). Comparison of the calculational method against experiments with solutions containing soluble gadolinium³ indicates a Δk of about -0.01 for most cases, with a maximum value of 0.015 for 20 g Gd/litre.

The series of calculations for lattices was performed assuming pure water of density $\rho = 1.0 \text{ g/cm}^3$ as moderator, fission products in the fuel were neglected. The results show a strong shift of the maximum k_{∞} value to the undermoderated range of small $\text{H}_2\text{O}/\text{fuel}$ volume ratios, if the Gd concentration increases. In case of UO_2 lattices with 3 g Gd/litre in the moderator maximum k_{∞} occurs at $\text{H}_2\text{O}/\text{fuel}$ volume ratios below 1.0. For mixed-oxide lattices the shift is not so prominent, the corresponding maximum k_{∞} occurs at $\text{V}_{\text{H}_2\text{O}}/\text{V}_{\text{Fuel}}$ ratios of 1.0 to 1.25. Only a slight dependence of maximum k_{∞} on the burnup state was found. The results clearly demonstrate that only a small reduction of k_{∞} can be expected by additional poisoning over 3 g Gd/litre in the undermoderated range.⁴

From the results, a correlation of gadolinium concentration present in water and the degree of burnup were established for UO_2 lattices in water (Fig. 1). Not accounting the content of fission products in the fuel, the acid molarity in the moderator and geometric restrictions, it can be concluded from Fig. 1 that

1. A Gd concentration of 3 g/litre and a content of remaining fissile material corresponding to a fuel burnup of 30,000 MWd/MTU result in a k_{∞} far below 0.9.

2. A Gd concentration of 3 g/litre gives a $k_{\infty\text{max}} \sim 1.0$ for fresh $\text{U}(3.5)\text{O}_2$ fuel.

3. The content of remaining fissile material alone is not sufficient for subcriticality.

To maintain a margin of $\Delta k = 0.10$ of subcriticality, the burnup should not be less than 15,000 MWd/MTU, the Gd concentration not less than 1 g/litre. A correlation very similar to Fig. 1 has been established for the mixed-oxide fuel; the conclusions are approximately the same as above. Taking account of 90-cm diam, the dissolver is subcritical with 3 g Gd/litre even for fresh fuel ($k_{\text{eff}} = 0.91$).

Fig. 1. Correlation of gadolinium concentration and burnup to maximum k_{∞} . UO_2 -lattices (PWR) in water, initial enrichment 3.5%.

If pure water were substituted by 7 M HNO_3 as moderator, a reduction of $\Delta k \approx 0.04$ was found for maximum k_{∞} . To simulate the content of nonvolatile fission products in the fuel, an equivalent gadolinium content was derived from ORIGIN data by weighing over thermal absorption cross sections. A remarkable reduction of k_{∞} occurs in this case only for unpoisoned lattices ($\Delta k = -0.08$ for $k_{\infty\text{max}}$). For a Gd concentration of 3 g/litre in the moderator, the fission products in the fuel reduce $k_{\infty\text{max}}$ only by 0.01.

The results for homogeneous solutions and (U + Pu) concentrations up to 1200 g/litre lead to analog correlations as in Fig. 1. A gadolinium concentration of <1 g/litre is completely sufficient for criticality control. The analog investigation of U/Pu solutions from dissolved mixed-oxide fuel reveals that the gadolinium concentration has to be duplicated to get about the same k_{∞} .

1. M. J. BELL, "ORIGIN—The ORNL Isotope Generation and Depletion Code," ORNL-4628 (1973).
2. J. E. SUICH and H. C. HONECK, "The HAMMER System," DP-1064 (1967).
3. R. C. LLOYD, E. D. CLAYTON, and L. E. HANSEN, "Criticality of Plutonium Nitrate Solutions Containing Soluble Gadolinium," *Nucl. Sci. Eng.*, 48, 300 (1972).
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Proposed Plans for the Use of Soluble Nuclear Absorbers at the Idaho Chemical Processing Plant, J. L. Lee (Allied Chem)

Soluble neutron absorbers are proposed for criticality safety control in future processes at the Idaho Chemical Processing Plant. Solutions of neutron poisons have been used in the past for criticality control in processing various reactor fuels. No problems were encountered in the safe use of the neutron poisons although dissolution of different types of fuel occasionally required reevaluation of the poison concentrations. Proposed plans include the uses of soluble neutron poisons in the Rover fuel dissolver, the Fluorinel dissolver, and in increased concentrations in the electrolytic dissolver. The purpose of this paper is to present these proposals and to discuss the criticality safety aspects.

Use of a neutron poison partially relaxes the restrictions of geometry, volume, vessel size, and fissile material concentration. In general, this allows for greater production rates and flexibility in operations. In addition, soluble poisons can sometimes improve overall solution characteristics and provide additional margins of safety. Some disadvantages of nuclear poisons include requirements for additional equipment for poison monitoring, requirements for additional sampling and analysis, and possible solution stability problems or negative effects on other processes. Soluble poisons were chosen for use in future processes because the advantages are considered to outweigh the disadvantages.

The Rover Fuels Processing Facility provides a head-end system designed specifically for reclaiming uranium from graphite matrix nuclear rocket fuels. Residual ash from fluidized bed burners is charged to the dissolver and is dissolved and complexed in a four step process: (a) nitric acid dissolution; (b) water dilution; (c) hydrofluoric acid dissolution; and (d) fluoride complexing. Quantities of reagents are determined by the

mass of the ash charged, and all of the solutions contain a minimum of 3.5 g/litre of boron to assure criticality safety. The dissolver heating and cooling water contain a minimum of 5 g/litre boron which protects against criticality in case of dilution of the dissolver solution by a leak between the dissolver and the cooling or heating jackets. Solution stability for process or upset conditions was proven in laboratory tests.

The criticality safety of the Rover Fuels Processing Facility is assured by means of engineering design, soluble nuclear poison (boron), and administrative controls. A k_{eff} of 0.93 was calculated for a full dissolver containing uranium at 1159 g/litre, boron at 2.64 g/litre, and nitric acid. Interactive effects of other vessels, each in the most reactive condition, will raise the maximum k_{eff} to 0.953. To ensure the neutron poison concentration, the dissolution reagents must pass through a nuclear poison detection system (NPDS) before entering the dissolver. The NPDS will automatically close shutoff valves between the feed tanks and the dissolver if the solution is below the required concentrations. A single failure analysis of the NPDS concludes that no single system failure or human error will result in an out-of-limits borated solution being transferred to the dissolver.

The Fluorinel Dissolution Process is a specially designed system for processing a number of metal-clad and oxide fuels. A soluble nuclear poison, cadmium, is essential for providing the required production rate and flexibility. Fuel batches are dissolved by (a) removal of the fuel cladding, (b) dissolution of the fuel material, and (c) dissolution of the residual components. Removal of the cladding destroys the fuel element geometry and the fissile material and residual components will fall to the bottom of the dissolver. The residual components con-

stitute only a small fraction of the total volume of the fuel assembly and do not affect criticality safety.

Criticality in the Fluorinel dissolver could result from exceeding the critical mass or from an inadequate poison concentration. The minimum critical mass for uranium oxide dispersed in a water solution containing 24 g/litre cadmium is ~19.0 kg of ^{235}U . This is based on a homogenized sphere of uranium oxide and poisoned dissolver solution at optimum density with a 30.48-cm-thick reflector of poisoned dissolver solution. The mass limit chosen for the dissolver is 10.5 kg of ^{235}U giving a k_{eff} of 0.88. Accumulation of a critical mass is prevented by positive identification of fuel units, administrative controls, procedures, and design of equipment to preclude double batching. Criticality due to an inadequate poison concentration could result from an inadequate starting concentration, dilution or precipitation of the poison. All reagents added during the dissolution cycle contain cadmium at 24 g/litre. Equipment design including automatic control systems with nuclear poison detectors, administrative controls and procedures will protect against low nuclear poison concentrations. Laboratory studies have shown solutions of 24 g/litre cadmium to be stable under any expected process or upset conditions except the presence of concentrated sulfuric acid.

The electrolytic dissolution facility is an existing facility at the ICPP for dissolution of stainless-steel fuels. Gadolinium is used as a soluble neutron absorber in the nitric acid dissolving reagent and the cooling system. Stainless-steel fuels planned for processing in the future will require reevaluation and adjustment of the gadolinium concentration to retain adequate criticality safety. Equipment design, administrative controls, sampling and procedures are used to assure criticality safety.

Shielding, Radiation Transport, and Dosimetry Applications

Dose Rate and Criticality Calculations for Plutonium Air Transportable Package (PAT-1),
W. H. Schmidt (Sandia Labs)

Introduction

The PAT-1 package¹ is to be used for the shipment of new nonspent PuO_2 reactor fuel by air. It is about the size of a 50-gal oil drum and is shock and fire resistant. Inside a double-walled stainless-steel drum are closely fitted redwood sections to protect the stainless-steel fuel containment vessel located at the center of the assembly. This vessel is approximately cylindrical with a height of 21.0 cm, outside diameter 13.6 cm, and wall thickness 1.4 cm.

The package is specified to hold 2-kg PuO_2 powder and is restricted to a radioactive decay power of 25 W to avoid overheating.

Radiation dose rate and nuclear criticality calculations have been made for the PAT-1 package which show that it meets the most stringent code of Federal Regulations (CFR) requirements.^{2,3}

Radiation Source Strength

The radiation source strength of the PuO_2 fuel was calculated with the ORIGEN⁴ computer program by assuming that a ^{235}U and ^{238}U reactor charge mixture is irradiated in a Hanford-like reactor in a conservatively long irradiation cycle. The plutonium is then assumed to be chemically separated from the reactor charge, and it is the oxide form of these plutonium isotopes that is to be packaged for shipment.

The radioactive decay thermal power of the PuO_2 fuel, as well as the gamma and neutron source strength, increases with time (to 70 yr) due to beta decay of ^{241}Pu into ^{241}Am , which is a strong alpha emitter. This is illustrated in Fig. 1 for a fuel load normalized to 25-W decay power (1.65-kg PuO_2) at an arbitrarily long decay time of $13\frac{1}{2}$ yr (approximately the half-life of ^{241}Pu).

Dose Rate Calculations

Dose rate calculations were made with the ANISN⁵ discrete-ordinates transport program using Hansen and Roach⁶ 16-energy-group neutron cross sections and an 11-energy-group gamma cross-section library. A spherical geometry approximation was used for simplicity. The source is $13\frac{1}{2}$ -yr-old fuel normalized to 25 W. Dose rate calculations were made for the undamaged package and a hypothetically damaged package in which all of the material outside of the fuel vessel or TB is assumed to be stripped away. The calculated results are given in Table I.

Nuclear Criticality Calculations

To be conservative, criticality calculations were made for a package of 3.15 kg of PuO_2 (instead of the 2-kg limit) containing only the fissionable ^{239}Pu isotope. Calculations were made with the KENO IV Monte Carlo Program⁷ for large finite and infinite arrays of closely

Fig. 1. Package fuel source strength vs decay time.

TABLE I

Calculated Radiation Dose Rates
(25-W decay power, $13\frac{1}{2}$ -yr-old fuel)

Radiation	Outer Surface (mrem/h)	91.4 cm Away (mrem/h)
Undamaged Package		
Primary γ 's	11	0.5
Secondary γ 's	0.2	9×10^{-3}
Subtotal	11	0.5
Neutrons	22	0.1
Total	33	2
Damaged Package (Bare TB)		
Primary γ 's	3.2×10^2	1
Secondary γ 's	2	8×10^{-3}
Subtotal	3.2×10^2	1
Neutrons	6.8×10^2	3
Total	1.0×10^3	4

packed undamaged and hypothetically damaged packages. Damaged package dimensions were obtained from impact tests¹ designed to simulate the worst possible accident.

Calculated criticality results show that all closely packed configurations of packages which were modeled are subcritical. The highest calculated effective neutron multiplication factor is 0.37, which was obtained for an array of 1000 closely packed hypothetically damaged and burned-out packages.

Conclusions

Radiation source strength calculations for new PuO₂ fuel indicate that the radioactive decay thermal power and radiation dose rate increase with time due to ²⁴¹Pu decay. The calculated dose rates are within the CFR limits^{2,3} of 200 mrem/h at the surface and 10 mrem/h at a distance of 91.4 cm for an undamaged package, and 1000 mrem/h at the surface of a damaged package. The criticality calculations show there is little neutronic interaction between packages and indicate that large closely packed arrays of damaged or undamaged packages are subcritical. Therefore, the PAT-1 meets the CFR criteria for a Fissile Class I package.²

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2. *Code of Federal Regulations*, 10CFR71, Chap. 1.
3. *Code of Federal Regulations*, 49CFR173, pp. 392-399.
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Risk Analysis and Human Factors Engineering

Risk Projection and Human Factors as Related to Nuclear Systems, *R. J. Nertney (EG&G Idaho)*

It is becoming increasingly important that individuals involved in design and evaluation of nuclear systems be knowledgeable in the area of quantitative risk evaluation.

This is necessary to permit the comparison of accident/incident risks with other managerial risks; failure to produce on schedule, failure to produce within budget, and failure to produce a functionally adequate product. It is also necessary to make adequate cost-risk-benefit presentations and decisions.

There are two basic methods for performing quantitative risk projections.

The first of these is through use of quantitative fault trees or other failure propagation models. In this case the system is described in an appropriate analytical model, failure probabilities are assigned to elements of the model, and overall system failure probabilities may be determined. These probabilities may then be associated with the consequences of failure to yield risk information.

These methods are widely known and utilized in the nuclear industry and we will only comment that the difficulties in practical use of the methods lie in three areas:

1. difficulties in obtaining component failure rate data which may be related to the system under analysis on a valid one-to-one basis
2. difficulties in evaluating the confidence levels for risk projections due to lack of such one-to-one correspondence on input data
3. difficulties in obtaining a complete frequency-severity spectrum of potential accidents, since each accident scenario must be individually analyzed.

The method is also dependent, of course, on the degree of quality assurance exercised to assure that the physical system "matches" the risk projection model being used. A final consideration is the validity of the analytical methods utilized to generate the consequence data.

The second method for projecting risk is the so-called actuarial approach—that is, the use of "actual" past experience on an overall systems basis.

In this approach to risk projection the relative frequencies and severities of accidents actually experienced are studied to determine the frequencies and severities of accidents that might be expected in the future.

The outputs of such a system are identical to those used in conventional insurance rating:

1. maximum potential loss
2. most probable maximum loss

3. frequency-severity distributions

4. mean annual loss

5. identification of special-risk types.

Inputs for such an approach may be obtained in the form of "hard data" obtained from actual operational systems, the "opinions" of experts, or usually from a combination of the two.

The advantage of this method is that the "answers" are projected directly without detailed analysis of individual component and subsystem failure probabilities. The weakness, in the case of nuclear systems, is the lack of serious accidents to provide a projection base.

This situation suggests that the proper method of approach involves a combination of the two methods. The quantitative fault tree methodology should be used to provide "fix points" in the extremely low-probability, high-consequence range. Operational data should be used at the lower end of the spectrum to project from the low-severity, high-frequency accidents actually experienced. In this way, as additional operating data (on both a component and total system basis) are accumulated the two methods compliment one another in creation of increased confidence for risk projection at all frequency-severity levels.

A final consideration in risk projection is the important role of the human in risk evaluation and projections. Overall system reliability is totally dependent on the humans who design, install, operate, and maintain the systems. The high degree of variability in both human performance and in the conditions under which human reliability data have been obtained in the past means that sophisticated analysis is necessary to obtain valid human reliability data. Such validity is necessary because these data represent as important an input to the risk projection models as the hardware reliability data.

This is true even in areas where operator-override devices apparently remove the influence of the operator from the system. In this situation the effectiveness of the override and safety devices themselves is dependent on human design, installation, testing, and maintenance.

This dependence becomes particularly important as the required system reliability becomes greater.

The papers that follow will deal further with some of the practical considerations involved in risk projection and in consideration of the human element in nuclear systems.

Practical Risk Analysis, *G. J. Briscoe (EG&G Idaho)*

Risk analysis can be used to project future losses including catastrophic events, optimize allocation of safety resources, and do specific cost/benefit analyses. Risk can be

Fig. 1. Electrical risk.

defined by the equation

$$R = P \times C$$

where

R = risk (degree or magnitude)

P = probability of occurrence

C = cost or consequence of occurrence.

To avoid adding apples and oranges, the consequences must be expressed in a common unit such as dollars. The consequences must also include direct and indirect tangible

costs and intangible results. SSIC-11 gives factors by which direct tangible costs can be multiplied to obtain indirect tangible and intangible losses associated with the direct loss. (Intangible losses from nuclear incidents are considered ten times greater than the tangible losses.) While some may not agree with the values placed on intangibles, a wider use of similar values will lead toward a consensus.

In addition to knowing the amount of risk, knowledge of the frequency/severity distribution is also very useful. This distribution can be obtained by the use of log-normal paper. The data will generally lie on a straight line and can be extrapolated to include the risk of large accidents. The straight line on log-normal paper can be transferred to log-log paper to obtain a curve which can be compared to the "line-of-balance" curve discussed by Dr. Nertney. An example is given in Fig. 1, Electrical Risk. Integration of this curve gives a risk of \$30,000/yr, whereas the observed loss over the previous five years was only \$13,000/yr. Based partly on this information, an electrical engineer was added to the safety staff.

Another type of analysis which is useful as a guide in resource allocation is a comparison of the risk for each energy category against the safety resource allocated to that type of energy. Figure 2 is such a graphical comparison for a typical DOE contractor and indicates a very unbalanced matching of resources to risks. (The resources in Fig. 2 include Safety Division resources but not operational costs.) As shown in the Risk Management Guide, the most effective safety program is balanced with large resources allocated to large risks and small resources allocated to small risks. These types of data can thus be used to optimize allocation of safety division resources.

A final example of risk analysis application is cost/benefit analysis for managerial decision making. The value of criticality evacuation alarms can be determined by calculating the risk from criticality incidents with and without an alarm system. The difference is the risk-reduction value of the alarms. Data for criticality incidents will be presented and an

Fig. 2. Safety investment vs residual risks.

exercise in risk analysis will be performed with audience participation. Audience participation will permit the gathering of data illustrating the range and variance existing among individuals in making subjective judgments. (The results will be submitted to the ANS for publication, if they so choose). The results should not be interpreted to mean that everyone should or should not have an alarm system. The results should be used by management as a significant factor in determining whether alarms should be upgraded in high-risk areas or whether the requirements might be relaxed in low-risk areas.

Risk Analysis should never be the final or sole factor in any decision related to safety. Rather, its use will make for more rational management decisions as our society inches its way to "answering" the question of how safe is safe enough, recognizing that contrary to some individuals belief, there are limits to the amount of available resources that can be allocated to safety.

The Human Component: An Off-the-Shelf Subsystem, M. C. Stewart (EG&G Idaho)

To assure system safety and reliability, we must analyze the functions served by all the subsystems, including the personnel subsystem. In this presentation, we will examine some factors that can determine whether the personnel subsystem can be an asset or a liability to nuclear safety.

There are two types of effort that demand thorough analysis of human performance: design and development of a new system and investigation of human-initiated malfunction¹ in an existing system. In this presentation, existing systems will be stressed, since resolution of problems in these is more difficult.

Human performance becomes a problem when there is a discrepancy between the level of performance the system requires and the level of performance contributed by personnel. In investigating, it is desirable to observe human performance first hand, and to interview at the working level. We need to analyze the working situation to determine why errors are made. It is sometimes helpful to imagine that tasks are arranged along a continuum from "few" to "many," and to plot these, considering the available time.

In conditions such as shown in 2, there will be errors made where a large number of tasks must be performed in a limited amount of time, and also where a small number of tasks do not absorb the allotted time. In the latter case, the problem increases when people must respond to aperiodic low-frequency signals. The monotony of monitoring a low-level input degrades human reliability.

Tasks can also be considered on a continuum from "simple" to "complex." A greater number of simple than of complex tasks can almost always be performed (considering time as a constant), but there are limits to the number, even

of simple tasks, that can be performed without degrading reliability.

The tendency to add to the number of simple tasks per time period becomes a problem, especially if the system is not optimally designed for human performance. One example of the negative impact of System Design is where the system (rather than the operator) establishes task time parameters, including operator reaction time. More errors are made when people cannot establish their own pace.

In an existing system where human-initiated malfunctions have occurred, the first corrective step is analyzing the task to identify the point at which human performance diverges from requirements and to determine whether the cause lies in the input to the human or in the implementation of the human output. If a defective input is identified, then it becomes appropriate to examine information sources, including display design, instruction manuals, training courses, etc. If there is a problem in the output, we must consider control design, force application, etc. In each case, we consider design of equipment and facilities.

It is frequently a temptation to recommend training as a solution to human performance problems in existing systems, due to the costs of hardware modification. However, it is essential to know what training can and cannot do. Training is an excellent means to improve many (but not all) input deficiencies: it is a very poor method to use to modify behavior so as to assure reliability. People have habitual modes of reacting, and are sensitive to pacing and task overload, so consideration of these characteristics in design is a significant factor in improving output reliability. To improve system performance, we will work only with those errors that have some system impact, and not on human error *per se*.

In evaluating alternative recommendations for existing systems, we can run trade-off studies just as in new system design. A significant input to the trade-off or decision process is the calculation of a human performance reliability (HPR) estimate² for the system as it stands, and also as it would be under alternative modifications. The HPR figure is also a useful input to fault tree and other analyses.

Because of the cost factor involved in redesign of existing systems, we should use at least an informal cost/effectiveness evaluation that will yield an estimate of the ratio between design costs and the costs of tolerance of human error related to a particular system.

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2. Such as that used by ALAN SWAIN of Sandia Laboratories, "Technique For Human Error Rate Prediction (THERP)."

Data and Analysis for Nuclear Criticality Safety

A Computerized Library of Bibliographic Criticality Data, Brian L. Koponen, Thomas P. Wilcox (LLL)

Throughout the world in the past 35 years considerable effort has been expended to obtain criticality data that are necessary for the design of nuclear reactors, chemical processing plants, storage facilities, and transportation equipment. In spite of this effort, however, there has been little work in cataloguing the sources of criticality data. It is very likely that many experiments and calculations have been performed unnecessarily because existing experimental data were not readily available.

A few bibliographies have been published,¹⁻⁵ the most comprehensive and up to date being that of Paxton.⁵ These bibliographies, however, are quite limited in their usefulness, as they have not been kept up to date, are restricted in the types of experiments covered, or lack sufficient indices or concordances to enable a user to quickly locate data of interest. None of these bibliographies has been computerized, and only two^{3,4} include abstracts.

This paper presents the results of a project at Lawrence Livermore Laboratory (LLL) that has produced a computerized bibliographic data base in a form that is readily available to government and industry for use in locating criticality data needed in their programs. Computerization permits easy searching of the data base through on-line searching, indices, or multiple-term concordances. This data base should be useful in the extension of criticality handbooks, construction of benchmark libraries, and in validation of computer codes and cross-section sets, as outlined in American National Standard ANS 8.11, "Validation of Calculational Methods for Nuclear Criticality Safety."

The present data base contains more than 1070 documents pertaining to experimental criticality measurements. Calculational studies of experimental measurements are included; however, theoretical calculations of criticality data, reactor design studies, and criticality safety studies are omitted. Document sources consist of: (a) government, university, or industry reports, (b) papers presented at conferences or symposia, and (c) journal articles (see Table I). These documents contain data on exponential, pulsed-neutron, critical and subcritical measurements for single units, fuel element lattices, and arrays of fissile units. The neutron-energy spectra for the measured systems vary from fast to thermal. In addition to authors, dates, titles, and report numbers, each document record includes an abstract and a description of the site or program under which the measurements were made. If available, the Nuclear Science Abstracts Number (NSA) is given and the conference, symposium, or journal is identified.

Various permuted indices and concordances on authors, sites, and key words or phrases in the title and abstract assist the user in locating wanted data. The bibliographic data are stored and presented according to standardized formats. This feature is of considerable help to the user in his search of the indices and concordances. When a document is thus located, the user may then turn to an expanded listing of the records and obtain the abstract and further information. Based on this material, he may decide to get the full-text report. Examples of the types of listings available will be presented, one of which is shown in Fig. 1.

The MASTER CONTROL⁶ computer program was developed at LLL to store and organize the bibliographic data base as well as the numerical data extracted from the bibliographic data (and has since been adapted for use as a generalized data storage and retrieval system). Development of a computerized benchmark library was begun some years ago.^{7,8} With the completion of the bibliographic data base, the numerical data base will be expanded and diversified to produce a more comprehensive benchmark library.

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2. W. E. BOST, "Fast Neutron Pulse Reactors and Fast Neutron Critical Assemblies (A Literature Search)," TID-3571, U.S. Atomic Energy Commission (Apr. 1962).

TABLE I
Sources of Documents in the Bibliographic Data Base

Number of Documents	Source
5	Bibliographies
26	General Compilations of Calculations
19	General Descriptions of Laboratories or Compilations of Experiments
2	Aerojet General
5	Alco Products
1	Allis-Chalmers
142	Argonne National Laboratory
25	Atomics International
4	Australia
19	Babcock and Wilcox Company
4	Battelle Memorial Institute
34	Brookhaven National Laboratory
11	Canada
1	Combustion Engineering
1	Cornell University
38	France
12	General Atomics
29	General Electric
20	Germany
186	Hanford
4	Italy
7	Japan
50	Livermore
140	Los Alamos
2	Martin Corporation
3	Massachusetts Institute of Technology
5	NASA, Lewis Research Center
2	Netherlands
1	North American Aviation
198	Oak Ridge National Laboratory
1	Phillips Petroleum Company
4	Poland
60	Rocky Flats
1	Sandia
24	Savannah River
1	South Africa
6	Sweden
58	United Kingdom
15	USSR
26	Westinghouse
1	Yugoslavia

SAVANNAH RIVER - U(3%) RODS IN WATER

1015	ROGERS, W. B. / KINARD, F. E. /	NSEAO, 20, 266-271	64-11-00	MATERIAL BUCKLING AND CRITICAL MASSES OF URANIUM RODS CONTAINING 3 WEIGHT PERCENT URANIUM-235 IN WATER
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SAVANNAH RIVER - URANIUM-235, URANIUM-233, PLUTONIUM MINIMUM CRIT. MASS

1259	CLARK, H. K.	TANSAO, 08, 200-201	65-06-00	EFFECT OF DISTRIBUTION OF FISSILE MATERIAL ON CRITICAL MASS
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SAVANNAH RIVER - URANIUM(1.1%) SLUGS IN WATER

1523	SMOTREL, J. R.	TANSAO, 17, 266-267	73-11-00	LIGHT-WATER-MODERATED REACTIVITY OF TUBULAR SLUG PAIRS CONTAINING 1.1 WT% ENRICHED URANIUM
1694	SMOTREL, J. R.	DP-MS-73-41	NONE	LIGHT-WATER-MODERATED REACTIVITY OF TUBULAR SLUG PAIRS CONTAINING 1.1 WT% URANIUM-235

SOUTH AFRICA - PELINDUNA CRITICAL EXPERIMENTS U(2%)METAL-HEAVY WATER

126	DE VILLIERS, J. W. L.	PEL-170	68-02-00	SAFETY REPORT ON THE PELINDUNA CRITICAL EXPERIMENTS
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SWEDEN - FRO ZERO ENERGY FAST REACTOR (U(20%) - GRAPHITE)

379	HELLSTRAND, E. / ANDERSSON, T. L. / BRUNFELTER, B. / KOCKUM, J. / LONDEN, S. O. / TIREN, L. I.	AE-207	65-12-00	EXPERIMENTAL STUDIES ON ASSEMBLIES 1 AND 2 OF THE FAST REACTOR FRO
788	ANDERSSON, T. L. / BRUNFELTER, B. / CECCHI, P. F. / HELLSTRAND, E. / KOCKUM, J. / LONDEN, S. O. / TIREN, L. I.	AE-233	66-06-00	A SUMMARY REPORT ON ASSEMBLY 3 OF FRO
918	ANDERSSON, T. L. / BJOREUS, L. / HELLSTRAND, E. / HAGGBLOM, H. / LONDEN, S. O. / TIREN, L. I. / KOCKUM, J.	ANL-7320, 159-185	66-10-10	EXPERIMENTAL AND THEORETICAL WORK AT THE ZERO-ENERGY FAST REACTOR FRO
1421	SHER, R.	TANSAO, 12, 714	69-11-00	INTEGRAL MEASUREMENTS OF FAST-NEUTRON SPECTRA IN SOME URANIUM- AND PLUTONIUM-FUELED FAST REACTOR CORES

SWEDEN - HEAVY WATER LATTICES

1952	PERSSON, R. / BUSTRAAN, M. / BLOMSJO, E.	GENEVA-1, 05, 239- 241	55-08-00	SOME EXPERIENCES ON AN EXPONENTIAL PILE OF URANIUM AND HEAVY WATER
1948	PERSSON, R. / BLOMSJO, E. / ANDERSEN, E. / ASPELUND, O. / DODERLEIN, J.	GENEVA-2, 12, 364- 373	58-09-00	EXPONENTIAL EXPERIMENTS ON HEAVY WATER NATURAL URANIUM METAL AND OXIDE LATTICES

Fig. 1. Bibliography of critical experiments (LLL). Portion of the concordance on sites or programs.

3. J. FURNISH, "The Data of Nuclear Reactor Physics. A Bibliography," LA-3740, Los Alamos Scientific Lab. (Apr. 1967).
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Interlaboratory Computational Comparisons of Critical Fast Test Reactor Pin Lattices, J. F. Mincey, H. T. Kerr (ORNL), B. M. Durst (BNWL)

An objective of the Consolidated Fuel Reprocessing Program's (CFRP) nuclear engineering group at Oak Ridge National Laboratory (ORNL) is to ensure that chemical equipment components designed for the reprocessing of spent LMFBR fuel (among other fuel types) are safe from a criticality standpoint. As existing data are inadequate for the general validation of computational models describing mixed plutonium-uranium oxide systems with isotopic compositions typical of LMFBR fuel, a program of critical experiments has been initiated at the Battelle Pacific Northwest Laboratories (PNL). The first series of benchmark experiments¹ consisted of five square-pitched lattices of unirradiated fast test reactor (FTR) fuel (see Fig. 1), moderated and reflected by light water. Calculations of these five experiments have been conducted by both ORNL/CFRP and PNL personnel with the purpose of exploring how

accurately various computational models will predict k_{eff} values for such neutronic systems and if differences between k_{eff} values obtained with these different models are significant.

A summary of the initial calculational results is given in Table I. Three sets of computations are compared; each set utilizes virtually the same KENO-IV code² and geometrical model but different cross-section libraries. As noted in Table I, two of the experimental lattice grids were recently discovered (by PNL personnel) to have slightly smaller pitches than values used in these three sets of calculations. Updated calculations are in progress; meanwhile, the slight corrections to the pitches do not affect present computational comparisons since the pitch values among the three calculational sets are consistent.

As shown, the last two sets of calculations performed by PNL and ORNL/CFRP personnel with the NITAWL/KENO-IV codes generally constitute better results. Comparing these two sets, the ORNL/CFRP k_{eff} values appear lower than PNL results by about 1%. PNL personnel used a forerunner³

TABLE I
Comparison of Interlaboratory Calculational Results

Experiment number	Square pitch (mm)	Lattice width (rods)	Total number of pins in lattice ^b	k_{eff}^a		
				PNL ¹ EGGNIT-THERMOS/ KENO-IV ^c	PNL ¹ NITAWL ³ /KENO-IV ^d	ORNL/CFRP NITAWL ⁶ /KENO-IV ^e
003R	7.696	36 ^f	1038			
		36 ^g	1037 ^h	1.012 ± 0.005	1.019 ± 0.005	1.002 ± 0.004
		36	1296	1.016 ± 0.004		1.004 ± 0.005
		36	1224	1.006 ± 0.004		
		36	1188	1.003 ± 0.004		
005	9.677 ¹	28	606 ^h	1.043 ± 0.007	1.025 ± 0.005	1.012 ± 0.005
		28	605 ^h	1.023 ± 0.005		1.009 ± 0.003
		28	560	1.006 ± 0.006		
		28	504	1.059 ± 0.004		
001	12.446	18	306		1.011 ± 0.005	0.993 ± 0.004
		18	280			0.994 ± 0.004
		18	279 ^h	1.031 ± 0.004		
		18	252	1.001 ± 0.005		
004	15.392	18	205 ^h	1.055 ± 0.005	1.016 ± 0.005	1.007 ± 0.004
		18	204			1.001 ± 0.004
		18	180	1.014 ± 0.006		
006	19.355 ¹	14	163 ^h	1.052 ± 0.004	1.020 ± 0.004	1.009 ± 0.004
		14	162 ^h	1.024 ± 0.005		1.015 ± 0.005
		14	140			
		14	112	0.954 ± 0.005		

^aError limits are one standard deviation on the KENO-IV² calculations.

^bType 3.2 FTR fuel pins unless noted otherwise.

^c18-energy-group EGGNIT⁷-THERMOS⁸ averaged cross sections from FLANGE⁹-ETOG¹⁰ processed ENDF/B-III and ENDF/B-IV data.

^dPredecessor to the ORNL DLC-43B/Criticality Safety Reference Library collapsed into 27 groups using the MALOCS module in AMPX-1.^c

^eORNL SCALE version⁶ of the criticality safety reference library; derived from ENDF/B-IV data using a preliminary version of the Scale system⁶ and collapsed into 27-group form.

^f66 type 3.1 FTR fuel pins on a 15.392-mm lattice pitch positioned on either side (36 on one side, 30 on the other) of a 27 x 36 fuel pin array of type 3.2 pins at a 7.696-mm square pitch.

^g65 type 3.1 FTR fuel pins on a 15.392-mm lattice pitch positioned on either side (36 on one side and 29 on the other) of a 27 x 36 fuel pin array of type 3.2 pins at a 7.696-mm square pitch.

^hExperimentally determined critical size.

¹Experimental pitch is slightly less than this value.

(a) REFERENCED DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY STANDARD

(b) ²⁴¹Am CONTENT IS 0.24 ± 0.02 wt %

(c) 35.6 mm FOR TYPE 3.1 FUEL PINS

Fig. 1. Simplified description of the fast test reactor fuel pin.

to the 218-group Criticality Safety Reference Library* (CSRL) and the AMPX-I⁵ version of NITAWL. The ORNL/CFRP personnel used a broad-group subset⁶ (27-group structure) of the CSRL⁴ produced with a preliminary version of the SCALE⁶ system (containing an updated NITAWL module). Hence, the discrepancies in calculated k_{eff} values are presently attributed to these two different processings of ENDF/B-IV data. Known differences in the libraries are: the extension of resolved resonance treatments below 3.05 eV, and the shielding of resolved elastic scattering resonances for heavy isotopes in the SCALE⁶ version.

This supposition is currently being investigated by running several ORNL-generated NITAWL/KENO-IV models of these experiments at PNL. The discrepancies between PNL EGGNIT⁷-THERMOS⁸/KENO-IV and NITAWL/KENO-IV calculated k_{eff} values are also under investigation.

Summarizing, of the calculational models examined, results obtained via preliminary SCALE system modules agree more closely with the experimental k_{eff} of unity for the three lattices whose pitches are not in question. Equally important is the existence of significant discrepancies between results obtained with different versions^{5,6} of AMPX-related modules. Such discrepancies reemphasize that appreciable differences can exist between different versions of a calculational tool as well as between different computational models. Only by freezing computational techniques and evaluating them against sufficient benchmark data can the user have some assurance that he will not be caught unawares by their inaccuracies.

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SSTR Dosimetry in Critical Mass Measurements, E. P. Lippincott, R. Gold, F. H. Ruddy, C. L. Long (Westinghouse Hanford), J. H. Roberts (Macalester College)

As part of the Advanced Fuel Recycle Program, critical mass measurements have been made of several lattices consisting of mixed oxide ($\text{PuO}_2\text{-UO}_2$) fuel pins in water.¹ In addition to providing a reliable technological basis for nuclear criticality control, these measurements are designed to provide benchmark data for validation of cross sections and calculational techniques used for mixed plutonium and uranium systems. To characterize some of the lattices more thoroughly, spatial fission rate distributions were measured using solid-state track recorders (SSTR). These fission rate traverses cannot only be compared with calculated shapes, but since SSTR observations provide absolute fission rates, the data were also used as input to the multiple-foil unfolding code, SAND-II,² to derive adjusted neutron spectra. The extremely high sensitivity of the SSTR method affords *in situ* dosimetry for critical mass measurements with negligible spatial and spectral perturbations.

The SSTR measurements were carried out using mica track recorders with isotopic fission deposits of ^{235}U , ^{238}U , ^{232}Th , and ^{237}Np . Fissions in the deposit create damage tracks in the mica, which can be counted under an optical microscope after suitable etching.³ To measure spatial and isotopic fission rates simultaneously, it was necessary to adjust deposit thicknesses to obtain track densities as close as possible to the optimum range for manual scanning. The optimum range for high accuracy results varies from 10^4 tracks/cm² (to obtain statistical accuracy) up to 2×10^5 tracks/cm² (to avoid track pileup). To obtain these densities, two types of deposits were used. Thin deposits provide low fission track densities with negligible fission fragment self-absorption. The exact mass of such deposits must be determined and, if only part of the track recorder is to be counted, the deposit must be highly uniform. The second deposit type is one with a thickness greater than the range of all fission fragments. The sensitivity of these "asymptotic" foils is independent of the thickness and has been determined previously.^{3,4}

Results from exposures at the middle of the lattice are shown in Table I for two pitches. In these measurements, each SSTR was scanned by two independent observers. Each observer scanned roughly 10^3 tracks, so the statistical precision of the measurements is slightly greater than 2%. Errors due to the uncertainty in the mass of the fission deposit as well as the error in the optical efficiency of mica must also be considered. In this manner, an overall error of roughly 3 to 5% (at the 1σ level) was attained for SSTR absolute fission rate observations in these critical facility studies.

The measured fission rates were input to the SAND-II code together with a starting spectrum¹ calculated with the KENO⁵ Monte Carlo code. The SAND-II code produced an adjusted calculated spectral shape which agreed with the measured data. Table I presents these SAND-II results for the thermal flux, flux of neutrons with energy >0.1 and 1 MeV, total flux, and the mean neutron energy. A detailed error analysis was not made, but on the basis of similar analyses, uncertainties in these spectral values are estimated to be 10 to 15% (1σ).

The extent of modifications in the calculated spectra may be inferred by a comparison of measured and calculated $^{235}\text{U}(n,f)$ to $^{238}\text{U}(n,f)$ ratios. For the 0.77-cm pitch lattice this ratio was measured to be 107 as compared with the calculated (KENO) value of 119, an 11% difference. For the

TABLE I
Mid-Lattice Fission Rates and Group Neutron Fluxes

	Irradiation CS2	Irradiation CS6
Lattice pitch	0.77 cm	1.90 cm
Relative fission rates (fissions/atom-s):		
$^{235}\text{U}(n,f)$	1.28×10^{-15}	2.28×10^{-15}
$^{238}\text{U}(n,f)$	1.19×10^{-17}	2.00×10^{-18}
$^{232}\text{Th}(n,f)$	3.60×10^{-18}	4.00×10^{-19}
$^{237}\text{Np}(n,f)$	8.52×10^{-17}	9.50×10^{-18}
Relative flux (n/cm ² -s)		
Thermal	1.85×10^6	4.31×10^6
E > 0.1 MeV	9.04×10^7	9.32×10^6
E > 1 MeV	3.48×10^7	4.48×10^6
Total	1.25×10^8	1.92×10^7
Average energy (MeV)	0.92	0.70

1.90-cm pitch these were 1140 measured and 1236 calculated, an 8% difference. At other lattice positions, comparable differences were observed. As indicated by this one spectral index, therefore, the KENO calculations produce a reasonable approximation to the neutron spectrum. However, improvements in flux-spectrum values can be obtained from SAND-II, which may then be used for comparisons with lattice calculations or in perturbation calculations to derive effects of poison or geometry changes.

The results of this program to date have shown that accurate SSTR measurements of fission rates can be made in critical mass facilities at very low exposure levels. Such measurements when combined with the multiple-foil technique have great potential for use as additional benchmark data for improvement and validation of calculational techniques.

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Critical Experiments with Solid Neutron Absorbers and Water-Moderated FTR Fuel Pins, B. M. Durst, S. R. Bierman, E. D. Clayton (BNWL)

The Advanced Fuel Recycle Program is concerned with the safe reprocessing of mixed plutonium and uranium oxide fuels, characteristic of fast reactors. The safe handling and storage of these fuels hinge on calculations and these in turn depend on clean, well-defined experiment data for validation. Benchmark experiment data have been acquired for fast test reactor (FTR)-type fuels for unpoisoned systems and systems intermixed with soluble poisons.¹⁻³ However, there are no data now available, that explore the criticality of these fuels intermixed with solid neutron absorbers (poisons). In this paper, we will present the results of experiments performed at the Pacific Northwest Laboratory (Critical Mass Laboratory) on fast test reactor fuel elements intermixed with solid neutron absorbers. The poisons used were Boral and cadmium plates and gadolinium cylindrical rods. Each absorber was separately examined to see its reactivity effect on lattices of FTR fuel pins in water; none was intermixed.

Measurements were performed with fuel pins at uniform square lattice spacings ranging from 9.68 to 24.89 mm. The fuel pins are described in Fig. 1. An isometric drawing of the experiment assemblies is shown in Fig. 2.

Four polypropylene lattice templates were used to uniformly position the fuel pins. These templates were slotted in the middle and edge fuel rows so that Boral and cadmium neutron absorber plates could be positioned and adequately held in place. Gadolinium pins were loaded like normal fuel pins. The neutron absorbers extended above and below the fuel region, except for a 4-mm gap located near the centerline of the fuel region, for the Boral plates and a 29-mm gap for the cadmium plates. The gadolinium pins, vertically, spanned the complete fuel column.

The critical assemblies were reflected on all sides by at least 150 mm of deionized water. The rods, however, were immersed in water only to a height of 150 mm above the top of the fuel region. Consequently, the top reflector region above each fuel rod was perforated by fuel pin hardware, projecting through the top reflector region.

Eight types of experiments were performed for each particular lattice pitch. These include:

1. experiments with no poisons added
2. experiments with boron, cadmium, and gadolinium poisons placed in the array center row
3. experiments with the above poisons placed in the core edge row
4. gadolinium pins interspersed in the middle of 5 X 5 fuel rod clusters across the fuel core.

One complete set of data is shown in Fig. 3 for the 15.39-mm lattice. As is readily seen, the most effective absorber is Boral, centrally positioned, requiring 369 rods (28.29 rows), compared to an unpoisoned value of 197 rods (13.11 rows). We acquired similar sets of data for lattice spacings of 9.68, 12.45, 19.36, and 24.89 mm, and these data will also be presented.

As a result of these experiments, we now have experiment data for FTR-type fuel elements interspersed with solid neutron absorbers. These data can be used to benchmark calculation schemes used in the criticality analysis of these types of fuels.

Technical drawing of the FTR fuel pin assembly, showing dimensions and components. The assembly is a multi-layered structure mounted on a carbon steel tank floor.

Dimensions:

- Overall width: 552 mm $\pm$ 1 mm
- Overall length: 584 mm
- Top plate width: 292 $\pm$ 1 mm
- Top plate length: 508 mm $\pm$ 1 mm
- Distance from top plate to middle slot: 1629 mm
- Distance from middle slot to bottom of fuel pin: 1285 mm
- Distance from bottom of fuel pin to top of fuel: 1187 mm
- Distance from top of fuel to bottom of fuel: 1120.1 mm
- Distance from bottom of fuel to bottom of fuel pin: 630 mm
- Distance from bottom of fuel pin to bottom of fuel: 205.7 mm
- Distance from bottom of fuel to bottom of fuel pin: 73 mm
- Distance from bottom of fuel pin to bottom of fuel: 0 mm
- Distance from bottom of fuel pin to bottom of fuel: 554 mm
- Distance from bottom of fuel pin to bottom of fuel: 479 mm
- Distance from bottom of fuel pin to bottom of fuel: 352 mm
- Distance from bottom of fuel pin to bottom of fuel: 508 mm $\pm$ 1 mm
- Distance from bottom of fuel pin to bottom of fuel: 552 mm $\pm$ 1 mm

Components and Features:

- POLYPROPYLENE TEMPLATE
- FTR FUEL PIN
- HOLE DIAMETERS: 6.02 mm $\pm$ 0.10 mm
- EDGE SLOT WIDTH: 4.67 mm $\pm$ 0.06 mm
- SLOT DEPTH: 4.60 mm $\pm$ 0.22 mm
- NEUTRON ABSORBER PLATE
- MIDDLE SLOT
- PLEXIGLASS BASE PLATE (25.4 mm)
- 6061-T6 AL CHANNEL 153 x 50.8 x 0.35 mm
- CARBON STEEL TANK FLOOR

Fig. 2. Isometric view of fuel and absorber plate experiment setup.

EXPERIMENT NUMBER	NEUTRON ABSORBER ⁽⁴⁾ / POSITION IN LATTICE ⁽¹⁾	ARRAY WIDTH (PINS)	CRITICAL LENGTH ^{(2) (5)} (PINS)
013	NONE / ---	15	13.11
014	Cd / ROW 1 (EDGE)	14	15.26
015	BORAL / ROW 1 (EDGE)	14	15.77
041	Gd / ROW 1 (EDGE)	14	15.22
016	Cd / ROW 8 (MIDDLE)	14	21.94
020	BORAL / ROW 8 (MIDDLE)	14	28.29
042	Gd / ROW 8 (MIDDLE)	14	20.91
040	Gd / Gd PINS CENTERED IN 5 x 5 FUEL CLUSTERS ⁽³⁾	15	15.93

- (1) "ROW" REFERS TO LINE OF LATTICE HOLES RUNNING PARALLEL TO LONG SIDE OF TEMPLATES. (SEE FIGURE 2). THE "EDGE" ROW MEANS THE ROW OF TEMPLATE HOLES NORMALLY OCCUPIED BY THE FIRST ROW OF FUEL. THE MIDDLE ROW, LIKEWISE, REFERS TO THE ROW OF HOLES IN THE ARRAY CENTER.
- (2) THE CORRECT INTERPRETATION OF THE CRITICAL LENGTH IS TO ASSUME THIS AS A FRACTION OF A FULL CELL ALONG THE ENTIRE ARRAY WIDTH. FOR EXAMPLE, IF THE ARRAY LENGTH IS 13.11 ROWS THIS IS 13 ROWS PLUS 0.11 ROWS. DO NOT EQUATE THIS WITH NUMBER OF PINS IN A ROW, OR THE POSITION OF THESE PINS IN THE FRACTIONAL ROW.
- (3) A Gd PIN IS POSITIONED AT THE CENTRAL POSITION OF EVERY 5x5 CLUSTER OF FUEL. THE ENTIRE ARRAY, IN TURN, IS PUT TOGETHER IN A REPETITIOUS MANNER, OF THESE 5x5 CELLS.
- (4) THE BORAL AND Cd PLATES STRETCHED TO THE EDGE OF THE FULL 508 mm TEMPLATE LENGTH. CONSEQUENTLY, SOME PORTION OF THE PLATE WAS OUT IN THE WATER REFLECTOR REGION. THE Gd PINS, HOWEVER, DID NOT EXTEND OUT INTO THE WATER BEYOND THE FUEL.
- (5) THE EXPERIMENT ERROR IS $\leq 0.3\%$ IN CRITICAL LENGTH.

Fig. 3. Critical array size for 15.39-mm lattice for various neutron absorber positions in fuel array.

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Critical Experiments Using High-Enriched Uranyl Nitrate with Cadmium Absorber, W. E. Converse, R. C. Lloyd, E. D. Clayton (BNWL), W. A. Yuill (Allied Chem)

The need for experimental criticality data on high-enriched uranium solution systems containing soluble neutron poisons has been indicated.¹ Data have been presented previously on soluble poisons in plutonium systems²⁻⁴ and in low-enriched uranium systems.⁵ To supply data required on high-enriched uranium, a series of experiments was performed at the Pacific Northwest Laboratory (Critical Mass Laboratory) using high-enriched (~ 85 wt% ^{235}U) uranyl nitrate solution with cadmium nitrate as a soluble neutron

poison. These experiments were specifically undertaken to support the design of the dissolvers to be used in the Fluorine Dissolver Process and Fuels Storage (FAST) Facility at the Idaho Chemical Programs Plant (ICPP). The ICPP is operated by the Allied Chemical Corporation, Idaho Chemical Programs, for the U.S. Department of Energy. The experiments supply data for use in criticality safety and as benchmarks to test calculational techniques.

This series of experiments was performed using a 241.8-mm-i.d. stainless-steel (Type 304) cylindrical vessel reflected on the sides and bottom with at least 200 mm of water or water plus cadmium nitrate. The experimental vessel thickness was 0.79 mm on the sides and 6.35 mm on the top and bottom. The cylindrical reflector tank was made of carbon steel with a 1016-mm o.d. and 2.77-mm-thick walls on the sides and bottom. For all experiments, the reflector solution height was maintained at the top of the experimental vessel. A detailed description of the experimental assembly is shown in Fig. 1.

The uranyl nitrate used in these experiments contained ~ 500.0 g U/litre with 0.1 M excess nitric acid. The average enrichment was ~ 85 wt% ^{235}U .

Each experiment was performed by incrementally increasing the uranium solution level in the experimental vessel until the critical level was determined. A safety and

DIMENSIONS OF EXPERIMENTAL VESSEL:

Inside Radius, mm	120.9
Inside Height, mm	1066.8
Side Wall Thickness, mm	0.79
End Thicknesses, mm	6.35

EXPERIMENTAL DIAGRAM:

Fig. 1. Description of experiment.

control rod (positioned external to the experimental tank) were used. At each solution level, the safety rod was fully withdrawn and the control rod was withdrawn until it no longer influenced the reactivity of the experiment. The first two experiments were done with no cadmium added to the uranyl nitrate. The first experiment had water only in the reflector tank and the second had water plus cadmium nitrate (at ~15 g Cd/litre) in the reflector tank. For the remaining experiments, water alone was used in the reflector tank and the cadmium concentration in the uranyl nitrate to the experimental vessel was varied. The results of the experiments including details of solution concentrations and critical solution heights are presented in Table I.

Calculations were performed for each experiment by modeling the system using the KENO-IV⁶ computer code. Cross sections for KENO were generated using the EGGNIT⁷ computer code. Seventeen epithermal- and one thermal-group cross sections were used.⁸ Input data for EGGNIT was FLANGE⁹-ETOG¹⁰-processed ENDF/B-IV data. The calculated k_{eff} 's are also shown on Table I. For unpoisoned cases the codes are calculating about 3% high in k_{eff} . For poisoned cases the codes calculate k_{eff} about 2% low.

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TABLE I

Results of Experiments on the 241.8-mm-i.d. Cylindrical Vessel Containing Highly Enriched Uranyl Nitrate* and Cadmium Nitrate

Solution Cd ^a Concentration (g Cd/litre)	Reflector Cd ^a Concentration (g Cd/litre)	Critical Height (cm)	Calculations KENO/EGGNIT ^b $k_{eff} \pm 1\sigma$
0	0	22.33	1.024 $\pm$ 0.008
0	15	30.58	1.040 $\pm$ 0.010
1.98	0	28.20	0.971 $\pm$ 0.010
3.98	0	37.75	0.998 $\pm$ 0.012
6.35	0	76.05	0.985 $\pm$ 0.007

*As uranyl nitrate, $UO_2(NO_3)_2$, 478.7 g U/litre, S.G. = 1.6592, 0.194 M excess nitrate. Isotopics: (wt%) = ^{234}U = 0.94, ^{235}U = 85.02, ^{236}U = 3.46, ^{238}U = 10.58.

^aAs cadmium nitrate, $Cd(NO_3)_2 \cdot 4H_2O$.

^bKENO-IV calculations using 18-group-averaged cross sections from EGGNIT. One sigma (1 σ) values are derived from KENO code statistics.

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Reactivity Effects due to Spherical Fuel Particles in Uranium-Water Systems, T. J. Trapp, J. P. McNeece, D. R. Oden (BNWL)

It is a well-known fact that the maximum reactivity (k_{∞}) attainable in a homogeneous fuel-moderator mixture can increase when the fuel is lumped. A familiar illustration of this effect is the observed difference in the minimum critical enrichments for homogeneous (~ 1 wt% ^{235}U) and heterogeneous (natural U) uranium-water mixtures.¹ There are many computational methods and considerable experimental data available to the criticality safety analyst to allow the evaluation of the safety of either homogeneous fuel-moderator mixtures or lattices of fuel rods immersed in a moderator. However, the analyst is sometimes faced with the practical problem of evaluating an "intermediate" case consisting of a mixture of fuel particles in a moderator. Both the absence of experimental data and the lack of simple computational methods to address this specific problem have led us to develop a computational model for this type of system. This paper presents the results of a series of survey calculations performed using the model to examine the reactivity effects of spherical uranium fuel particles in a water moderator. The computational model is described in a companion paper.² Some of the variables considered in the survey, in addition to spherical particle diameter, include fuel material form (metal, oxide), uranium enrichment, and particle volume fraction (degree of moderation).

Reactivity changes from lumping fuel into particles result from two primary effects. The first effect is that a larger fraction of the neutrons is slowed down in the moderator, thus escaping resonance capture in ^{238}U . This increases the reactivity of the system. The second effect is that neutrons that slow down in the moderator travel some distance in the moderator before encountering a fuel particle. Therefore, a larger fraction of the neutrons is absorbed in the moderator in the heterogeneous mixture than in the homogeneous system. This effect produces changes in the thermal utilization and tends to decrease reactivity.

Modified versions of the GRANIT³ and EGGNIT⁴ computer codes were used to calculate the thermal and epithermal parameters and k_{∞} . The epithermal resonance calculation in EGGNIT was done using integral transport theory.⁵ The thermal calculation in GRANIT was also done using an integral transport theory⁶ method. k_{∞} was calculated based on two-group parameters from a 30-thermal-group and 68-epithermal-group spectrum. In both the thermal and epithermal calculations a Dancoff factor is required for the

Fig. 1. The ratio of the heterogeneous k_{∞} to the homogeneous k_{∞} for a solution of 5 wt% enriched UO_2 spherical particles in water.

fuel particles. An expression for the Dancoff factor for an infinite array of spherical particles is derived and discussed in a companion paper.²

Figure 1 shows the effects of fuel particle sizes on k_{∞} for 5 wt% enriched UO_2 particles in water. The fractional change ($\times 1000$) in the heterogeneous k_{∞} from the homogeneous k_{∞} is displayed as a function of the particle size. Each curve on the graph corresponds to a system with a fixed volume fraction of uranium (and fixed H/U). The maximum reactivity changes occur with a volume fraction of uranium of ~ 0.1 .

Fig. 2. The ratio of the heterogeneous k_{∞} to the homogeneous k_{∞} for a solution of 95 wt% enriched UO_2 spherical particles in water.

The curves show there is a small increase in reactivity below a particle diameter of 0.01 cm. This reflects the decrease in the resonance absorption of ^{238}U and corresponding increase in the number of neutrons available for thermal fission. As the particle size increases the resonance absorption continues to decrease and reactivity increases. For large particles the change in thermal utilization becomes larger than the change in resonance absorption and the reactivity decreases.

Figure 2 displays the same quantities for 95 wt% enriched UO_2 fuel in water. In this case, the ^{238}U resonance absorption is small. Slowing down in the moderator increases reactivity primarily by scattering neutrons from energies with low capture-to-fission ratios to energies with low capture-to-fission ratios. In this situation the absorptions in the moderator provide a greater contribution to the reactivity. This can be seen in the smaller magnitude of the positive changes and negative reactivity changes for some volume fractions and particle sizes.

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The Dancoff Factor for Cubic Arrays of Spherical Fuel Particles, J. P. McNeece, T. J. Trapp (BNWL)

Criticality safety analysis of systems containing fuel particles in a moderator requires an evaluation of the resonance and thermal self-shielding effects of the particles. An expression for the Dancoff factor is needed to calculate the self-shielding. Lane, Nordheim, and Sampson¹ derived an expression for the Dancoff factor for systems having small volume fractions of particles. Knight² has calculated the Dancoff factor for two spheres. In the study of the reactivity effect of particles in uranium-water mixtures³ we needed an expression for the Dancoff factor valid for systems having both large numbers and large volume fractions of particles. In this paper an expression is derived for the Dancoff factor which reduces to the Lane, Nordheim, and Sampson expression for small particle volume fractions and agrees with the results of a Monte Carlo calculation for large particle volume fractions. The derivation follows.

The probability that a neutron born with equal probability in any position in the absorber particle makes its next collision with a moderator atom is⁴

$$P_o^* = P_o \left[\frac{1}{1 + \frac{C \ell_o P_o \Sigma_p}{1 - C}} \right] \quad (1)$$

where

P_o = isolated particle escape probability

ℓ_o = mean cord length in the particle = $4R/3$ for a sphere

Σ_p = macroscopic total cross section of the particle

C = Dancoff factor—defined as the probability that a neutron escaping from a particle crosses the surface of another particle before making a moderator collision.

For a homogeneous system,⁵

$$P_o^* \approx P_o \left[\frac{V_m \Sigma_m}{V_m \Sigma_m + V_p \Sigma_p} \right] \quad (2)$$

where

V_p = volume fraction of the particles

Σ_p = macroscopic total cross section of the particle

V_m = volume fraction of the moderator ($1 - V_p$)

Σ_m = macroscopic total cross section of the moderator.

Equating Eqs. (1) and (2) yields the following expression for the Dancoff factor:

$$C = \frac{3V_p}{4R\Sigma_m V_m + 3V_p} \quad (3)$$

where R = sphere radius.

Since Eq. (3) was derived assuming homogeneity, it seems unlikely that it would give a good representation for the Dancoff factor for large volume fractions or large diameter spheres. Equation (3) gives the same value for the Dancoff factor as the expression of Lane, Nordheim, and Sampson for every dilute systems (volume fraction of particles ≤ 0.1). To check the validity of Eq. (3) for other than very dilute systems, a Monte Carlo computer program was written to calculate the Dancoff factor for cubic arrays of spheres.

In the Monte Carlo program a neutron is started isotropically from the surface of a sphere and tracked until it intersects another sphere. The distance from the starting point to the intersection is determined and the neutron weight is calculated as

$$W = \exp(-\Sigma_m X) \quad (4)$$

Fig. 1. The Dancoff factor for a cubic array of spheres.

where

W = neutron weight

Σ_m = moderator macroscopic total cross section

X = distance neutron travels.

The sum of the weights divided by the number of neutrons tracked is the Dancoff factor. The code was used to calculate Dancoff factors for sphere volume fractions from 0.005 to 0.5, sphere radii from 0.005 to 0.1 cm, and moderator macroscopic total cross section from 1.0 to 3.0 cm^{-1} .

The Dancoff factors calculated using the Monte Carlo method and those calculated using Eq. (3) are shown in Fig. 1. Values obtained from the two methods compare very well. Equation (3), therefore, is a good representation of the Dancoff factor even for large volume fractions of spheres.

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Some Results of Solid-Angle Study Using the SNAKE Code, J. S. Tang, J. T. Thomas (UCC-OR)

Although the solid-angle method of specifying allowed neutron interaction between units of fissile material has been widely used by the nuclear criticality industry for more than two decades, recent studies^{1,2} of the method have revealed areas where it fails to provide safe configurations. The Computer Sciences Division at the Oak Ridge National Laboratory, under a contract with the U.S. Nuclear Regulatory Commission, has undertaken a detailed study of the application of the solid-angle method to nuclear criticality of fissile materials. The purpose of the study is to examine the range of applicability of the solid-angle method regarding fissile material, array reflection, enrichment, unit shape, and array shape. This paper presents some results of the initial study which examines the effect of unit shape on the solid angle calculated with the SNAKE³ computer code. Cubic arrays of U(93.2) metal units were used in the calculations. The Monte Carlo criticality program, KENO,⁴ with Hansen-Roach 16-group neutron cross-section data sets⁵ was used for the k_{eff} calculations.

The SNAKE program, developed at the Battelle Pacific Northwest Laboratories, calculates the total solid angle at a unit in arrays of spheres, cylinders, and/or boxes using exact

Fig. 1. Ω_f versus unit k_{eff} of water-reflected cubic arrays of U(93.2) metal units.

Fig. 2. Influence of unit shape on solid angle of water-reflected cuboidal arrays of U(93.2) metal cylinders. Darkened symbols = water-reflected array at $a_x = a_y = 27.118$ cm, $a_z = 28.105$ cm. Open symbol = unreflected array at $a_x = a_y = 15.027$ cm, $a_z = 16.013$ cm.

analytic formulas. The code has several options useful in applied criticality safety; however, this study uses the basic option of the code to calculate the total solid angle subtended at the center of the most centrally located unit of an array.

Figure 1 shows the total fractional solid angle, Ω_f , versus k_{eff} of individual unit of critical cubic arrays of U(93.2)

metal cylinders and spheres reflected by 12 in. of water. Interaction curves of spheres and cylinders of height-to-diameter ratios (h/d) of 0.1, 1.0, and 4.0 are shown. This region in Ω - k plane encompasses intermediate h/d ratios. The safety curve for allowed unit interaction from TID-7016 (Ref. 6) is included for reference. The three data points not falling on the curve for array of cylinders of h/d of 0.1 are results of using the differential albedo option for array reflector in KENO calculations. For relatively small arrays ($4 \times 4 \times 4$), the differential albedo data underapproximate neutron leakage at array corners, and allow more neutrons to be reflected leading to an underestimate of the unit mass required to achieve array criticality. This phenomenon also occurs for small arrays in the other three curves, but it is not as apparent.

Figure 2 illustrates the influence of unit shape on array total solid angles of 12-in.-water-reflected and unreflected arrays of U(93.2) metal cylinders. The fractional total solid-angle change, $(\Omega - \Omega_0)/\Omega_0$, where Ω_0 is the total solid angle at h/d of 1.0, is plotted against the unit shape parameter, β . The darkened symbols represent water-reflected arrays and the open symbols represent unreflected arrays. It is noted that the dependence on β of the fractional total solid-angle changes of both reflected and unreflected arrays is almost identical, although the values of the corresponding total solid angles are considerably different.

Based on results of Fig. 1 and 2, one may be able to use the interaction curve for spheres to establish a safety curve for three-dimensional arrays of cylinders even though there are situations (at extreme values of h/d) that result in slightly lower Ω_f values than those of spheres of the same k_{eff} .

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Special Process Equipment for Nuclear Criticality Safety

Critically Safe Concentrator, R. C. Lloyd, E. D. Clayton (BNWL)

A solution concentrator is needed to efficiently carry out the experimental program with plutonium nitrate solutions at the Critical Mass Laboratory. During the course of the experiments the solutions are often diluted so they need to be concentrated for further use, storage, or for shipping out. Also, waste solutions sometimes need to be cycled through the concentrator so they can be shipped out. A concentrator unit was designed to handle these materials used at the lab and to be critically safe by geometry during its operation.

This concentrator unit consists of the concentrator, condenser, and condensate tanks. The concentrator is an annular tank having steam heating coils in the inner annular region. The condenser is a coil of 12.7-mm tubing which is raw water cooled. The condenser cooling water and conden-

sate from the steam heating system discharge to the waste cooling water drain. The condensate tanks are two 22-litre glass units. The condensate is pumped to the waste holdup tank for storage and reuse.

Material in the concentrator may be plutonium nitrate or solutions containing mixtures of plutonium and uranium. The solution volume operating range is between 6 and 18 litres. A high-level trip stops the pump at the 18-litre level. Solution level and density are displayed on the operation panel continuously, which ensures indication of buildup of material inside or material being left in the concentrator, although such conditions would not be expected.

The CML concentrator is of a special annular design. The overall dimensions are 406-mm o.d. X 838 mm high. The solution annulus is 39.1 mm thick X 787 mm high, with 0.762-mm Cd metal cladding on the outside; total

TABLE I
CML Concentrator Dimensions

Region	As Designed		Used in Calculations	
	Material	Radius, mm	Material	Radius, mm
1	Void	131.8	Void	131.8
2	Titanium	135.6	SS	135.6
3	Pu Nitrate	174.6	Pu Nitrate	174.6
4	Titanium	177.8		177.8
5	Cd (0.03-in. Clad with 0.02-in. SS)	179.6	Cd	178.6
6	Fiberglass Resin	205.0	H ₂ O	204.0
7	-----	-----	Void	609.6
8	-----	-----	H ₂ O	711.2
Annulus		39.1 mm	39.1 mm	
Annulus Height		790 mm	787 mm	
Total Volume of Annulus		30.0 litres	30.0 litres	
<u>Catch Tanks (2)</u>		<u>Condenser</u>		
Diameter	152.4 mm	<u>Water Tank</u>		
Length	1.219 m	Diameter	304.8 mm	
Volume	22.2 litres	Height	152.4 mm	
		Volume	11 litres	
		<u>Condensate Coil</u>		
		Tubing i.d.	15.88 mm	
		Length	~ 1.9 m	
		Volume	~ 0.4 litres	

capacity is 30 litres. Operating capacity is 18 litres. Steam coils of 12.7-mm diam are located within the inner section for heating the solution. The dimensions and geometry are given in Table I. Regions are listed, starting with the center void as region 1. For the calculations, stainless steel was used for the cladding in place of titanium, since cross sections for the latter were not in the code. For the thicknesses involved, this change does not significantly affect the results.

A 30.48-mm-diam X 152-mm-high condenser is located adjacent to the concentrator, near the tip, and two 152-mm-diam X 1.219-mm-long condensate catch tanks are located perpendicular to the concentrator, on the opposite side. The catch tanks are in the same vertical plane and 203 to 279 mm apart, edge to edge. These tanks are located in the solution preparation hood, 457 mm above a slab storage tank.

In normal operation, plutonium nitrate solutions may be concentrated to 300 to 400 g Pu/litre. The maximum expected concentration is 600 g/litre. The nitric acid content will range from 1 to 10 M. For the criticality safety analysis, it is assumed that the concentrator contains plutonium nitrate. Plutonium concentration is not limited; however, for the calculations, the most reactive concentration from the standpoint of criticality is assumed to be 175 g Pu/litre, as experiments have shown the minimum critical volume for $\text{Pu}(\text{NO}_3)_4$ solutions to occur at about this concentration.¹

For the concentrator, k_{eff} was calculated using the transport theory code DTF-IV and Monte Carlo code GEM-3.

TABLE II
Criticality Calculations for Concentrator

Plutonium Nitrate Solution Composition	
	g/litre
Pu	175
NO_3	182
H_2O	937
Free acid	0
Specific gravity	1.293
H/Pu	142
N/Pu	4.01
System Description	
	k_{eff}
1. Concentrator geometry as designed, but no SS walls; filled to capacity with 175 g/litre $\text{Pu}(\text{NO}_3)_4$; no free acid; 127-mm H_2O reflection	0.661 ± 0.019
2. Same as (1), but with SS walls	0.637 ± 0.015
3. Same as (2), but changed reflector to 25.4-mm H_2O /203-mm void/203-mm H_2O	0.633 ± 0.012
4. Same as (2), but full water reflection	0.643 ± 0.015
5. Same as (2), but Cd removed	0.836 ± 0.014
6. Same as (1), but 79.4-mm annulus	0.987 ± 0.010
7. Same as (1), but no nitrate, 1000 g/litre Pu as $\text{PuO}_2 \cdot \text{H}_2\text{O}$ in annulus; 203-mm water reflection outside; 6.4-mm water reflection inside	0.901
8. Same as (7), but 15-mm water inside	0.974
9. Same as (7), but 20-mm water inside	1.002
10. Same as (7), but 25.4-mm water inside	0.952
11. Same as (7), but Cd removed and outside water reflector of 25.4 mm	0.918

Experience with DTF-IV code indicates an accuracy of about ± 20 mk for plutonium systems: GEM-3, an accuracy of about ± 30 mk. The results are summarized in Table II. Interaction calculations show that k_{eff} is < 0.85 for the array of vessels filled to capacity with 175 g Pu/litre $\text{Pu}(\text{NO}_3)_4$ solution.

It is concluded that k_{eff} for the concentrator will be < 0.95 for all foreseeable conditions. To obtain criticality, it would be necessary to (a) exclude nitrate, (b) obtain a concentration of 1000 g Pu/litre (tank full), (c) increase the neutron reflection on the outside from nominal to full, and on the inside from 12.7-mm-thick steam to a uniform layer of water 20.3 mm thick. It is concluded that the operating contingencies have afforded an adequate margin of safety. This concentrator design has proved successful in operation and has been used intermittently over a period of 14 years during which several thousand litres have been concentrated.

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Compact Handling and Storage of Bulk Solutions of ^{233}U , ^{235}U , and ^{239}Pu , B. F. Gore, R. I. Scherpelz (BNWL)

Slab tanks have been successfully used in PNL's Critical Mass Laboratory for twenty years. Since its construction in 1959¹ thousands of litres of fissile solutions have been safely mixed, stored, and transferred through its solution handling system. The heart of this system is a battery of slab tanks having an inside slab thickness of 63.5 mm ($2\frac{1}{2}$ in.), made of stainless steel 6.35 mm ($\frac{1}{4}$ in.) thick. Tanks are located parallel to one another and separated by hydrogenous moderator and Cd sheeting at least 0.5 mm thick.

Primary tankage at startup included a 70-litre mix tank (with temperature control) and two auxiliary tanks of 140- and 160-litre capacities. These three tanks were located side-by-side in a large glove-box hood. The layout was quite conservative due to the limited analysis capabilities existing at the time. They were separated by about 0.1 m of polyethylene moderator having Cd sheets attached to each side. Cadmium and polyethylene were also placed on the room side of the outer tanks to isolate them neutronically from external effects. Neutron leakage from these 0.6-m-tall tanks was enhanced by placing styrofoam spacers between the tanks and the Cd-sheathed moderator blocks.

Major additions to CML slab tankage were made in 1964 and 1977, and another is planned in 1979. These added tanks are all of 400-litre capacity, standing 1.1 m tall X 6.1 m long, with an inside slab thickness of 63.5 mm ($2\frac{1}{2}$ in.). To prevent bulging of the 6.35-mm-thick walls under hydrostatic pressure, cross-ties are welded inside the tanks 0.46 m up from the bottom of the tanks.

Two tanks were added in 1964, for use with Pu solutions. They were separated by 0.2 m of hydrogenous moderator and Cd sheets. They were placed in the experimental cell against a thick concrete wall, from which they were separated by 0.1 m of moderator. No moderator was placed against the large face of the room-side tank.

Two more tanks were added in 1977, for use with ^{235}U solutions. Reasons for installing slab tanks instead of Raschig-ring-filled tanks, include spatial constraints, the need to minimize surface holdup, the need to handle a wider range of concentrations, and location of the neutron poison material out of contact with the solution. Due to great improvements in calculational analysis tools, less conservatism was

necessary in design of these tanks. They were placed side by side, separated by 0.1 m (4 in.) of polyethylene moderator and Cd sheets. They were located across a corner of the experimental cell from the earlier tanks. No polyethylene was placed between the wall-side tank and the concrete cell wall, or against the face of the room-side tank.

Planning is under way to add two more tanks this year, to handle ^{233}U solutions. These tanks will be located adjacent to those installed in 1977, separated from them and from each other by 0.1-m-thick slabs of polyethylene and sheets of Cd. This grouping of four tanks will be capable of handling 1600 litres of fission solution in a space just 0.8 m wide X 6.1 m long X 1.1 m high.

Calculations have been performed, using the KENO-IV Monte Carlo code² and Hansen-Roach cross sections,³ to assess the degree of subcriticality of four such 400-litre tanks, separated by 102 mm of polyethylene, with 0.51-mm-thick Cd sheets placed against all large tank faces. Reflection is assumed to be by a concrete floor and wall, with full water reflection elsewhere. The tanks were modeled for the calculations by ignoring fittings and glass-epoxy coverings, which protect the Cd sheeting. Tank walls (inside the Cd) were included, as their neutron reflection increased system reactivity. The results reported are for uranium or plutonium solution concentrations of 600 g/litre, which is the maximum concentration planned for routine use. Aqueous solutions of the nitrate were assumed, with no excess acid. For conservatism, fissile isotope content was assumed to be 100%. The value of neutron multiplication calculated when all four tanks were assumed filled with $^{235}\text{UO}_2(\text{NO}_3)_2$ is $k_{\text{eff}} = 0.836 \pm 0.007$. When the four tanks were assumed filled with $^{239}\text{Pu}(\text{NO}_3)_2$, $k_{\text{eff}} = 0.864 \pm 0.007$.

Calculations performed assuming the same contents in all tanks indicate that the reactivity of the tank next to the concrete reflector exceeds that of the other tanks. This is due to enhanced reflection of epithermal neutrons by the concrete. Solutions of ^{233}U , which are more reactive in slab geometry than solutions of ^{239}Pu or ^{235}U , will only be placed into the new tanks located away from the concrete wall. (This prevents contamination of other tanks by radioactive ^{232}U decay daughters.) Calculations assuming that the two tanks closest to the wall are filled with $^{239}\text{Pu}(\text{NO}_3)_2$ and that the two outer tanks are filled with $^{235}\text{UO}_2(\text{NO}_3)_2$, yield $k_{\text{eff}} = 0.858 \pm 0.006$. This is about the same as when all tanks are filled with ^{239}Pu solution, and indicates that the reactivity of a polyethylene-reflected tank of ^{233}U solution does not exceed the reactivity of a concrete-reflected tank of ^{239}Pu solution. Additional calculations show that the subcriticality of the four-tank grouping is not jeopardized by off-normal construction well beyond that to which control is readily achievable.

Thus, side-by-side slab tanks, partially but effectively neutronically isolated from each other by neutron moderator and poison materials located out of contact with tank contents, have been demonstrated to be effective vessels for the accumulation, mixing, and storage of fissile solutions. The specific examples discussed herein are primarily of value as indications of the effectiveness of the method and as a starting point for designing other systems employing this method of criticality control.

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CPVC Piping as a Secondary Criticality Control, E. L. Maceda (B&W), D. K. Sgarlata (B&W-Apollo), F. G. Welfare (B&W)

This paper describes the tests and calculations¹ that were used at the Babcock & Wilcox Company to justify the use of Chlorinated Polyvinyl Chloride (CPVC) pipe as a substitute for Borosilicate-Glass Raschig rings as a secondary criticality safety control. Several large tanks (2000 to 4000 litres) are used to store uranyl nitrate solution at the Apollo, Pennsylvania plant. The uranium is enriched to a maximum of 5 wt% and the tanks are limited to a maximum ^{235}U concentration of 13.5 g/litre. By controlling the concentration, the tanks are nuclearly safe, but because of the size of the tanks they are also poisoned as a secondary criticality control. Formerly, Raschig rings were used in these tanks, but because of the corrosive chemical environment the rings had to be replaced as frequently as once a year.

In 1970, Goodyear Atomic Company^{2,3} reported their evaluation of Polyvinyl Chloride (PVC) and CPVC piping as a neutron poison in their UF_6 steam-heated-vaporization bays. They considered both empty piping and pipes that were filled with boron. CPVC pipe was found to be a slightly better poison when the pipes were empty, and PVC pipe was a marginally better poison than the CPVC pipe when they were filled with boron. It was the poisoning ability of these pipes that encouraged B&W to consider using them in uranyl nitrate solutions.

Before any change in neutron poison material could be implemented, both the nuclear safety and physical stability of the pipe had to be demonstrated. To evaluate the nuclear safety of the pipe, k-infinity was calculated for a highly enriched uranyl nitrate solution (at maximum-allowed ^{235}U concentration) poisoned with Raschig rings in accordance with the ANSI Standard N16.4-1971.⁴ The ANISN computer code⁵ with the 16-group Hansen Roach cross-section⁶ set was used in all the calculations. K-infinity of 0.89 was found when the volume fraction of the solution occupied by Raschig rings was 0.32. It was concluded the PVC and CPVC would be acceptable for solutions where they reduced k-infinity to 0.89 or less. Next, k-infinity was calculated for 5 wt% enriched uranyl nitrate solutions poisoned with various-size PVC and CPVC pipe. The densities of pipe used was 1.18 g/cm³ for PVC and 1.34 g/cm³ for CPVC corresponding to the lowest manufacturing tolerances for these types of pipe. CPVC pipe was found to be a better poison than PVC. The largest size pipe acceptable from a nuclear safety standpoint is 15.24-cm (6-in.) schedule 80 CPVC pipe for solution densities up to 1100 g U/litre of 5 wt% enriched uranium. This density is greater than four times the density of ^{235}U allowed in the uranyl nitrate tanks.

Next, corrosion tests were done on 1.27-cm ($\frac{1}{2}$ -in.) CPVC Schedule 80 pipe. Uranyl nitrate, raffinate, and dissolver solution maintained at 93°C (200°F) were used in these tests. The worst case was dissolver solution which leached out 0.3825 g from an initial mass of 82.248 g of chloride in 17 weeks. This corresponds to a leach rate of 0.71 wt%/six months. In the uranyl nitrate solution, the leach rate was 0.088 wt%/six months. Ten Raschig rings were placed in a

Fig. 1. Top view of tank with CPVC pipe in place.

uranyl nitrate solution at 57°C (135°F) in another corrosion test and the rings dissolved at a rate of 0.0633%/100 days. In a similar test with CPVC pipe, the chloride leached at a rate more than 600 times slower than this. The actual temperature of the solution in the tanks in the plant is slightly above room temperature (~27°C) and this lowering of temperature will decrease the rate of chloride leaching.

Figure 1 shows how the tank looks with CPVC pipes installed. The concentric circles are stainless-steel lips that hold the pipe in place on the bottom of the tank. The large pipes are 15.24-cm Schedule 80 CPVC pipes while the others are smaller diameter Schedule 80 pipe, one of which is used for samples when the periodic inspection of the pipe takes place. There is a manhole on top of the tank for access.

To date, the tanks with CPVC pipe have been in operation for one and one-half years and none of the pipe has had to be replaced. In addition to being more corrosion resistant, there is no settling problem (associated with Raschig rings) because the pipes are placed in the tanks vertically. Also, there is less sludge buildup and liquid holdup in the tanks. To date, the tanks with pipe have performed as expected, which simplified maintenance. Therefore, CPVC pipe is a suitable replacement for Raschig rings in this application and because of its cost and ease of maintenance, it should be considered in other applications where appropriate.

1. Docket 70-135, "CPVC Pipe Used as a Secondary Means of Criticality Control," SNM-145 License Amendment 66 (1974).
2. J. L. FEUERBACHER, "Relative Efficiency of Various Means of Adding Nuclear Absorbers to Direct-Steam-Heated Vaporization Bays for 10-Ton and 14-Ton UF₆ Cylinders," GAT-DM-1136, Goodyear Atomic Company (1970).
3. J. L. FEUERBACHER, "Computer Check of 2½-Ton UF₆ Cylinders at 5% Assay in Direct-Steam-Heated Vaporization Bays Containing Chlorinated Polyvinyl Chloride Pipes Filled with Boric Acid," GAT-DM-1137, Goodyear Atomic Company (1970).

4. ANSI N16.4-1971, "American National Standard, Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," American Nuclear Standards Institute (1971).
5. W. W. ENGLE, Jr., "A User's Manual for ANISN," K-1693, Oak Ridge Gaseous Diffusion Plant (1967).
6. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAM-2543, Los Alamos Scientific Lab. (1961).

Storage of Unirradiated Fuel in Borated Concrete, D. L. Honkonen (SRP)

Large inventories of enriched uranium-aluminum fuel elements are stored in borated concrete racks at the Savannah River Plant (SRP). This storage method was selected because

1. The racks are virtually indestructible.
2. The boron precludes accidental criticality for all credible accidents including natural disasters.
3. Racks without built-in neutron poisons that provide the same protection are about three times as expensive.

Fuel elements produced at SRP are aluminum-clad, aluminum-enriched uranium alloy fuel tubes up to 9.4-cm o.d. and 410 cm long. Two or three tubes are assembled coaxially to form fuel assemblies for the reactors. The alloy-bearing section of these assemblies is about 380 cm long and contains up to 1300 g ²³⁵U/m.

The largest rack for storing both assemblies and individual tubes consists of borated concrete slabs 412 cm long X 168 cm wide X 15 cm thick, containing 10 storage holes 9.5 cm in diameter. Ninety-one slabs are stacked in a 13-high X 7-wide array, for a total of 910 storage positions (Fig. 1). The concrete is a standard mix except that some of the sand is replaced by borosilicate glass frit (quenched molten glass), so that the concrete contains 0.35 wt% boron. This mixture provides a boron-to-²³⁵U atom ratio of 2.1 when a 1300-g ²³⁵U/m fuel element is placed in each storage hole.

This rack costs \$113,000. A 910-position rack without built-in neutron poison would cost about \$400,000 because it would require widely spaced positions to provide the same protection from accidental criticality with interspersed moderation and so would require six times as much floor area.

The nuclear criticality safety of the 910-position storage rack was confirmed by:

1. determination of the boron content of each slab
2. calculation of the reactivity of each rack loaded with fuel
3. measurement of the reactivity during fuel loading.

The boron content of each slab was determined by measuring the neutron absorption of the concrete at five positions before the slab was stacked. The average boron content for the 91 slabs was 0.35 wt%.

The effective multiplication constant, *k*, of the rack loaded with elements containing up to 1300 g ²³⁵U/m was calculated with MGBS-TGAN¹ and KENO² with Hansen-Roach cross sections.³ The normal condition of no water in the storage holes and the most-reactive condition of full water flooding were calculated. The results of these calculations (Fig. 2) show that the rack loaded with 1300 g ²³⁵U/m fuel would be subcritical with accidental flooding.

Fig. 1. Borated concrete rack for storage of fuel elements.

Fig. 2. Reactivity of 910 element rack.

Correlations with exponential experiments of fuel elements in water performed at the Savannah River Laboratory validate these calculations for this analysis:

1. At similar $H/^{235}\text{U}$ atom ratios, MGBS-TGAN overestimates experimental k values by 7%.
2. KENO calculations agree with nearly critical configurations derived from these same experiments within $\pm 1\%$ k .

The reactivity calculations for the rack were checked during the initial loadings with fuel containing 380, 410, and 600 g $^{235}\text{U}/\text{m}$. The neutron multiplication, M , was measured as fuel was loaded into a rack section and the effective multiplication constant, k , was calculated from

$$k = \frac{M - 1}{M}$$

These k values are plotted in Fig. 2. In all cases, the calculations overestimate the measured values. Therefore, it was concluded that this rack could safely be loaded with elements up to 1300 g $^{235}\text{U}/\text{m}$.

The rack described in this paper was installed in 1966. Subsequently, five additional racks with a total fuel storage capacity of 3300 elements have been constructed at different locations at SRP.

1. H. K. CLARK, "Computer Codes for Nuclear Safety Calculations," DP-1121 (TID-4500), E. I. du Pont de Nemours & Co., Savannah River Lab. (Nov. 1967).
2. G. E. WHITESIDES and N. F. CROSS, "KENO—A Multi-group Monte Carlo Criticality Program," CTC-5, Union Carbide Corporation, Nuclear Division (Sep. 10, 1969).
3. E. G. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543 (Dec. 6, 1961).

Criticality Safety: Its Impact on HTGR Fuel Fabrication Processes, J. D. Jenkins (SC Energy Res Inst), S. R. McNeany (ORNL)

Equipment to fabricate particulate fuel for recycle to high-temperature gas-cooled reactors (HTGRs) has been under development at the Oak Ridge National Laboratory for a number of years. Criticality safety is one of the important design criteria for all fuel processing equipment. In the case of fabrication equipment for HTGR fuels, unique problems were encountered where conflicts between desired material throughput and criticality safety resulted in modifications to the process and process equipment.

An HTGR fuel element is a composite structure. Coated microspheres of enriched uranium are bonded together with fertile microspheres in a carbonaceous matrix to form rods. These rods are then inserted into hexagonal graphite blocks to form the finished element. Both the rods and elements contain fissile material in dilute form and present no special criticality problems. Safety limits can be met by maintaining adequate spacing between units and the fabrication equipment requires no special design features to achieve the desired production rates.

It is in the earlier process steps, carbonization of the enriched fuel microspheres, and application of the coatings where criticality problems arise. The carbonization step is one in which ion-exchange resin beads loaded with uranium are heated to drive off the hydrogenous constituents and convert the particles to microspheres of UO_2 and UC. In the coating process these particles are fluidized in a furnace by a stream of argon and acetylene or hydrogen and methyl-trichlorosilane $(CH_3)_3Cl_3Si$. The molecular gases decompose at high temperature to deposit either a pyrocarbon or a silicon carbide coating on the particles. Effluents from the carbonization and coating furnaces consist of soot and condensable hydrocarbons. These must be processed by an off-gas cleanup system.

The fissile material is highly concentrated (0.5 to 1 g/cm³) in these stages of the process. Processing is done by batch lots and both the carbonization and the coating steps are time consuming. In addition, the process equipment is complex, expensive, and complicated to operate. Economic pressures are therefore to maximize batch sizes in larger equipment to obtain the desired throughput. Herein arises the conflict with criticality constraints.

Most of the process development work at ORNL was done in a 5-in. (12.7-cm)-diam furnace using either natural or ²³⁵U enriched uranium particles. For these materials a 5-in. cylinder is a critically safe configuration and no criticality problems arose when the development furnace was designed and operated with water cooling coils and a kerosene off-gas scrubber.

However, when larger equipment (9-in. or 22.9-cm-diam furnaces) was specified to meet the material production rates required in a commercial ²³³U fuel fabrication facility, criticality safety became a prime issue. A number of potential criticality accidents could be postulated in the larger furnaces if water cooling and kerosene scrubbing were retained in the process. These included:

1. cooling coil rupture and water flooding of the furnace
2. backflow of scrubber liquid (kerosene) into the furnace
3. slug ejection of the fluidized bed of fuel particles into the scrubber catch basin
4. gradual accumulation of particles in the scrubber catch basin

5. water flooding of the entire hot cell (²³³U fuel processing must be done in a contained facility).

The resin carbonization and particle coating processes were modified as a result of these criticality safety concerns. Specific modifications included:

1. substitution of a nonhydrogenous liquid (perchloroethylene, C_2Cl_4) both for furnace cooling and off-gas scrubbing
2. exclusion of hydrogenous liquids and water lines from the cell containing the carbonization and coating equipment
3. relegation of the resin loading operation—a continuous aqueous process where the resin beads are soaked in $UO_2(NO_3)_2$ solution—to a separate cell
4. development and planned incorporation of a hydrogen monitor¹ on the scrubbing liquid recirculation line to ensure that dissolved hydrocarbons in the perc did not exceed a set threshold.

These processes and equipment modifications have been shown² to allow safe operation of the desired 9-in.-diam carbonization and coating furnaces. One minor process problem was encountered as a result of substituting perc for kerosene in the scrubber column, notably leaching of uranium into the first porous pyrocarbon coating as a result of backflow of chlorine into the furnace. This problem was overcome by increasing the argon overpressure in the fluidizing gas stream. Thus, process and equipment modifications were successful in maintaining acceptable criticality safety margins and allowing equipment sizing to meet commercial material production rates for HTGR fuels.

1. S. R. McNEANY and J. D. JENKINS, "Detection of Hydrogen Buildup in Initially Pure Nonhydrogenous Liquids," ORNL/TM 6466 (Dec. 1978).
2. S. R. McNEANY and J. D. JENKINS, "Criticality Considerations for ²³³U Fuels in an HTGR Fuel Refabrication Facility," ORNL/TM-6136 (Jan. 1978).

Enhanced Safety in the Storage of Fissile Materials, G. E. Williams, N. J. Alvares (LLL)

INTRODUCTION

The need for increased storage capacity in existing fissile-material storage vaults is a problem encountered by criticality safety specialists. Unfortunately, construction of additional vault space may be prohibitively expensive because of extensive security and safety standards.¹ Criticality safety calculations indicate storage limits could be increased if a neutron absorber were used to isolate unusually massive units.

MATERIAL FABRICATION

The Plastics Shop at Lawrence Livermore Laboratory (LLL) has fabricated a "plastic-like" supporting material impregnated with a neutron-absorbing agent that is suitable for "lining" the inner surfaces of fissile-material storage containers. This material consists, by weight, of 50% food-grade borax, 25% coal tar, and 25% epoxy resin. It costs about one-tenth as much as commercially available materials, can absorb enough neutrons to isolate units of fissile material, and possesses such structural qualities as flexibility and machinability.

The Plastics Shop mixed the ingredients and appropriate catalysts in a commercial dough mixer to form a liquid with strong thermal-neutron-absorption properties. Then they

poured the liquid into storage containers, slowly rotating each container on its side until each was uniformly lined. After the liner set, they added part of the liquid to the bottom and lid of each container.

ADVANTAGE FOR CRITICALITY SAFETY

Monte Carlo neutron-transport and criticality safety code MORSE-C indicated a much enhanced safety margin for containers lined with borax-loaded epoxy resin neutron absorber. For example, a flooded $4 \times 4 \times 4$ array of 10-kg Orallo spheres packed as closely together as permitted in 0.038-m^3 containers has a multiplication constant, k , of about 0.93, and would exceed the Laboratory convention that a k greater than 0.9 is unacceptable. However, the same array would have an effective k of only about 0.70 if each part were enclosed in a cylinder lined with neutron absorber. This reduction in the state of criticality would allow solid spherical units of up to 26 kg of Orallo or 6 kg of plutonium to be stored in floor storage areas of the Laboratory's Central Vault without exceeding an effective k of 0.9. Without neutron absorber, only 18.5 kg of Orallo or 4.5 kg of plutonium would be permitted.

FIRE-RESISTANT PROPERTIES

General safety considerations demanded the absorber have several properties to ensure it did not increase other hazards while reducing the criticality hazard in fissile-material storage vaults. One significant hazard associated with any hydrocarbon is its flammability. To ensure containers lined with neutron absorber would not appreciably increase the possibility or severity of fire, the Fire Science Group of LLL tested the fire-propagation properties of a freestanding sheet of absorber; the duration and intensity of fire necessary to cause the neutron-absorption properties of a liner container to fail; and smoke-evolution properties.

The fire-propagation tests proved no unsupported burning of absorber occurred regardless of the intensity or duration of exposure to flame.

For endurance testing, the Fire Science Group partially or totally immersed several lined containers in a 160,000 kJ/min flame. After each test, some of which lasted 20 min, they assessed the condition of the liner. Although water trapped in the borax turned to steam, causing the liner to separate from the container wall, the collapse was not severe enough to damage the container's contents or to degrade its neutron-absorption properties.

Before fire testing, the Fire Science Group tested lined containers for their neutron-absorption properties. After exposing each container to flames for 10 to 20 min, they reevaluated its neutron attenuation. For a ^{252}Cf fission spectrum moderated by 25 cm of D_2O , the liner reduced the low-energy neutron flux by an average factor of about 50 before and after fire testing. These tests also showed the neutron-absorption properties of the liner were not greatly affected despite the partial collapse and loss (as much as 8%) of absorber in the form of steam.

The absorber's smoke-evolution properties were largely dependent on the mode of exposure. While slabs of absorber smoked very little when exposed to flames, smoke was prevalent in the absence of flames. Exposure to high-flux radiant heat (2.5 to 5 W/cm²) proved the liner must be considered a heavy smoke producer.

OTHER PROPERTIES

The Laboratory's Criticality Safety Committee subjected the absorber to additional tests to determine its other properties. To determine whether the absorber lost its neutron-

absorption characteristics if wet, they immersed a sample of absorber in water. After 60 h only 1.3% of the absorber dissolved, presumably because the coal tar and resin waterproofed most of the borax, proving sprinkler systems, rain, or accidental flooding should not degrade the absorber's performance.

They found the absorber pliable at room temperature and rigid and breakable at freezing temperatures. Although unsupported sheets of absorber sagged when exposed to the sun and smelled of coal tar (like creosote), the odor lessened with time and was not a serious disadvantage. Fire tests produced a gagging odor and full-face respirators were advisable if entry into the fire test chamber was necessary before ventilation. The toxicity of the absorber is unknown, but respiratory protection is warranted in the event of fire.

CONCLUSION

The Laboratory's neutron absorber for lining fissile-material storage containers is strong and pliable at room temperatures and will survive even the severe conditions of a fire without losing its neutron-absorption properties. It costs less to produce than commercially available materials and is easily adaptable for many uses. Safe for use in fissile-material storage vaults, the absorber preserves enhanced criticality safety in storage arrays, even for unusually massive units.

1. "General Design Criteria, Part II," Energy Research and Development Administration Manual Chapter 6301 (Mar. 1977).

Design Considerations for a Uranium Dissolution Vessel, Alfred C. Dumrose, Thomas P. McLaughlin (LASL)

Highly enriched uranium can be effectively recovered from lean (a few %U) refractory solids by a hydrofluoric-nitric acid leaching process. Due to the corrosive and hazardous characteristics of hydrofluoric acid, it was decided to perform the leaching operation batchwise, in vessels of simple geometry with no moving parts. An annular dissolver (see Fig. 1) of 90-litre capacity was designed for criticality safety for all credible accidental situations, including double batching and leakage of solution into the central volume.

Regular processing to recover highly enriched uranium from various types of unirradiated scrap, but mostly fuel from the Rover space reactor project, produces insoluble, refractory solids containing up to 5% uranium. These solids can be leached with a solution of 10M hydrofluoric acid (HF) and 1.7M nitric acid (HNO_3) to reduce the uranium content to <1000 ppm (0.1%) and discarded.

Two annular vessels were designed, fabricated, and have been in operation for two years. These are composite units made to be HF resistant, with the outer vessel formed of polypropylene, and the inner one of stainless steel coated wit Kynar (vinylidene fluoride).

To operate, a batch of solids weighing 12 kg and containing <1000 g U is added to about 30 litres of HF/ HNO_3 reagent which already has been introduced to the vessel. During operation, the contents are agitated by continuous air sparge. The temperature is maintained between 90 and 100°C by steam sparge for 2½ h. Condensate from the steam increases the volume to about 40 to 50 litres. Immediately prior to vacuum transfer of the dissolver slurry, about 15 litres of aluminum nitrate solution are added to complex excess fluoride so the slurry can be subsequently filtered

Fig. 1. Annular vessel for dissolving uranium solids in HF/HNO₃ reagent.

and solvent extracted in stainless-steel equipment. Total processing time required per dissolution is about 6 h.

Criticality analyses were performed with the two-dimensional, *S_N* computer code, TWOTRAN, in conjunction with the 16-group Hansen-Roach cross-section set.^{1,2} To ensure conservatism in the stepped R-Z cylindrical mockup of the curved dissolver base section, the volume contained in the computational model was ~5% larger than that actually contained therein. For conservatism and calculational convenience the solution was modeled as H₂O and U only. The dissolver is located ~45 cm above a concrete floor and is not closer than this distance to any appreciably reflecting surfaces.

For the normal operating state, with a maximum charge of 1000 g U(93), the multiplication factor is near 0.5. Double batching, i.e., 2000 g U(93), was considered sufficiently likely to require subcriticality in this event, for which $k \lesssim 0.8$. Additionally, the possibility exists that leaks into the central region could occur, which would obviously lead to a more neutronically reactive system. Although a single

batch would not be critical for leakage into an initially voided central cavity, a 2000-g U(93) charge could lead to criticality for such a leak. To ensure subcriticality in this scenario, polyethylene packets containing dry powdered boric acid are loaded in the central cavity, and then to prevent flotation the cavity is covered by a weighted plastic disk.

This annular dissolver is simple, relatively inexpensive (\$3500), largely maintenance free, easy to operate, processes a reasonable quantity of material, and poses no undue criticality hazards.

1. G. E. HANSEN and W. H. ROACH, "Six and Sixteen Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos Scientific Lab. (Nov. 1961).
2. K. D. LATHROP and F. W. BRINKLEY, "TWOTRAN-II: An Interfaced, Exportable Version of the TWOTRAN Code for Two-Dimensional Transport," LA-4848-MS, Los Alamos Scientific Lab. (July 1973).

Usefulness of Operating Reliability Data

Incident Reporting at DOE Nuclear Facilities, *C. A. Burchsted (ORNL)*

Incident reporting is not a new technique and is practiced in many organizations including the U.S. Nuclear Regulatory Commission, the Federal Aviation Administration, and most large industrial companies. In the U.S. Department of Energy (DOE), it is called Unusual Occurrence Reporting (UOR). UORs provide a means for the operating-level manager to clarify his thinking regarding an event and the corrective action on the one hand, and on the other, provides higher authority with a basis for tracking that corrective action, for notifying others of potential hazards or problems revealed by an unusual occurrence, and for uncovering possible trends that may reveal underlying problems. The DOE Nuclear Standards Office (NSO) reviews all UORs issued by DOE contractors, issues a semimonthly summary report of those UORs, and twice a year reviews all incidents that have occurred during the report period, in the aggregate, to extract any useful information of potential interest to contractors or DOE concerning special problems or trends.

Nuclear Fuel Cycle Education Modules

Nuclear Fuel Cycle Education Module on Nuclear Criticality Safety, R. A. Knief (*Univ of NM*), invited

A Nuclear Fuel Cycle Education Module has been developed for the field of nuclear criticality safety. The content of the module is outlined herein.

Nuclear criticality safety is charged with prevention of criticality in all operations with fissile materials outside of reactors. As with other safety-related activities, the balance of risks, costs, and benefits is an important guideline.

The practice of nuclear criticality safety involves integration of a wide range of administrative, calculational, and design principles. Administrative organizations and policy must assume continuing attention to this (and, indeed, all) safety and should establish a system that is as natural and workable as possible. Calculational capabilities must include simple "hand" methods for minor and/or routine facility changes and highly sophisticated computer methods for major design changes in geometrically complex operating areas. Design of processes and process equipment, storage and handling facilities, and transportation containers must assure subcriticality even under the constraint of highly improbable accidents (in fact, the "double-contingency principle" requires two such accidents for criticality). Recommended practice is to use "geometrically favorable" equipment and containers wherever possible. When production or other requirements make such limitations of sizes unacceptable, safety must be based on administrative controls; e.g., limits on mass, volume, concentration, and/or moderation.

Many of the current criticality safety practices have been developed as a result of accident experience. A variety of accidents at government and commercial processing facilities have all suggested that nonroutine operations seem to have the highest risk and deserve special attention.

Effective hand-calculational methods have facilitated analysis of routine operations such as rearrangement of arrays of storage containers or process equipment. The surface-density method considers the effective fissile density of an array as projected on a wall or floor. The solid-angle method is based on correlation of geometric interaction probabilities and multiplication factors. Both methods depend on comparison to and verification with experimental results.

The extremely inhomogeneous geometries in fuel cycle facilities dictate against the use of the diffusion-theory computer codes which are so useful in reactor design. Transport codes can often be applied for scoping calculations; e.g., infinite arrays. More detailed computations employ the Monte Carlo method because of its ability to handle the irregular geometries.

Nuclear criticality safety for an entire facility requires a range of administrative and geometric controls. Case studies are presented for a composite LWR fuel fabrication (slightly enriched uranium) operation and for a spent-fuel reprocessing plant.

Since slightly enriched uranium cannot be made critical without moderation, exclusion of hydrogenous materials is an especially important control in fabrication. Operations with and storage of UO_2 powder, pellets, fuel pins, and finished assemblies must be designed to be subcritical even with complete flooding with water.

Reprocessing facilities may contain spent fuel as assemblies and in acid solution, separated uranium and plutonium solutions, plutonium oxide or metal, and, perhaps, uranium hexafluoride for recycle. Almost every basic criticality control method is employed somewhere within such a facility.

The nuclear criticality education module was designed to describe an important, yet somewhat unrecognized, component of the nuclear fuel cycle. Discussions of both theory and practice are intended to provide the reader with a concise but thorough picture of the field.

Status of Experimental Programs and Research Projects

Regulatory History and Status of Nuclear Criticality Safety Research, *R. H. Odegaarden, D. E. Solberg (NRC)*

NRC (AEC) Regulatory staff members, since 1965, have attempted to confirm, through independent means, the adequacy and safety margins of various calculational methods used to determine the nuclear criticality safety of fuel fabrication facilities, fuel reprocessing facilities, fuel storage pools, fuel storage vaults, and fuel transportation packages. This work led to the publishing of the first technical paper and first confirmatory research by the regulatory staff. Currently, the NRC is sponsoring nuclear criticality safety research at Oak Ridge National Laboratory (ORNL), Rocky Flats Plant (RFP), and Pacific Northwest Laboratory (PNL).

BACKGROUND

The first efforts to confirm the adequacy of a calculation method for nuclear criticality safety were to use an empirical equation based on the density analog method originally proposed by H. C. Paxton. This work was completed in October 1965 and compared the specified density analog method with 58 critical experiments that were performed by A. D. Callihan et al., for metal and solution arrays.

The second effort was in 1968 and resulted in publishing the first technical paper ever presented by the AEC Regulatory staff. The paper was given by G. H. Bidinger and R. H. Odegaarden at the Second International Symposium on Transportation, Gatlinburg, TN. Problems encountered in criticality safety reviews were given, along with outlining hand-calculational methods acceptable to the Regulatory staff.

In January 1969, Regulatory sponsored a project at ORNL in which G. E. Whitesides made the ORNL cross-section processing codes and the KENO code operational at the National Bureau of Standards for Regulatory. Using the 16-group Knight-modified Hansen-Roach cross-section set and KENO, R. L. Stevenson and R. H. Odegaarden published an ANS paper in November 1969 on an empirical relationship between criticality and surface density first proposed by H. C. Paxton.

The first AEC Regulatory-sponsored confirmatory research project was initiated in January 1973 with E. D. Clayton, PNL. The purpose of the work was to verify criticality data on the effectiveness of fixed neutron absorbers (copper and copper-cadmium plates) between fuel blocks consisting of a homogeneous mixture of $\text{PuO}_2\text{-UO}_2$ -polystyrene at H to Pu + U ratios of 30.6 and 2.8 (Ref. 1). Other neutron absorbers (SS, SS-B, U(0.2), Boral, Cd, and Pb) were considered in 1976.²

Beginning in 1972, Regulatory began a series of efforts to improve and build on the nuclear criticality computer

programs in use at ORNL. This work has included improving cross-section processing programs and modifications to KENO.

CURRENT EXPERIMENTAL PROGRAMS

The NRC is sponsoring critical experiments applicable to both fuel element storage and shipping configurations and to bulk handling of fuel.

Pacific Northwest Laboratories

A series of critical experiments is in progress at the PNL Critical Mass Laboratory under the direction of S. R. Bierman, B. M. Durst, and E. D. Clayton. These experiments have determined the critical separation between clusters of optimally moderated fuel rods of 2.35 and 4.29 wt% ^{235}U enriched UO_2 with various neutron poison materials (Boral, SS, SS-B, Cu, Cu-Cd, Al, Zirc-4, Cd) placed between the fuel clusters.^{3,4} In addition, a series of experiments has been completed with these fuel clusters surrounded on two sides by depleted uranium and lead-reflecting walls. Experiments with steel-reflecting walls will be performed in the future.

Experiments are currently in progress for undermoderated clusters of 2.35 and 4.29 wt% ^{235}U enriched fuel rods with neutron poison materials similar to those identified previously between the fuel clusters (1.7 vol% H_2O -to-fuel ratio).

Rocky Flats Plant

Experiments were conducted at RFP under the direction of C. L. Schuske to evaluate critical height of 93% ^{235}U enriched uranyl nitrate solutions.⁵

A series of experiments has also been initiated to establish critical separation between low enriched ^{235}U uranium systems. The U_3O_8 is enriched to 4.5 wt% ^{235}U and experiments performed at each of three values of H/U atomic ratio in the range of 0.75 to 2.0.

CURRENT RESEARCH PROJECTS

Additional nonexperimental criticality research projects have been performed as identified below.

Oak Ridge National Laboratory

In 1975 it was proposed to the Computer Sciences Division (CSD) that they develop a system of unified computer programs for the standardized safety analysis of fuel shipping containers. This work has led to the development of the SCALE (standardized computer analysis for licensing evaluation) program. The criticality safety portion of the program is based on the cross-section processing programs NITAWL (resonance correction) and XSDRNPM

(cell weighting) and Monte Carlo criticality program KENO. SCALE includes several cross-section libraries in its data base; i.e., 123-group GAM-THERMOS, 27- and 218-group ENDF/B-IV, and 16-group Knight-modified Hansen-Roach.

The Nuclear Safety Guide (TID-7016) was recently revised by ORNL⁶ under NRC sponsorship.

A study is in progress to evaluate the areas in which the solid angle and surface density approximations provide nonconservative results and to develop modifications to these methods which will assure subcritical configurations when these methods are applied to all systems.

Pacific Northwest Laboratories

Work was undertaken at PNL to determine if inconsistencies or possible errors exist in the ENDF/B Pu and stainless-steel cross-section values and in the calculated neutron thermal energy values. Results of the study were presented to CSEWG with recommendations.

FUTURE EXPERIMENTAL PROGRAM

NRC proposes to sponsor experiments to evaluate the effects of soluble poisons on ²³⁵U enriched fuel. NRC also proposes to sponsor experimental programs with mixed-oxide fuel rods containing 2 wt% PuO₂. The experiments would be similar to the experiments performed at PNL for the 2.35 wt% ²³⁵U enriched uranium fuel rods. The critical experiments would simulate multielement fuel assemblies in water separated by water and with various neutron absorbers and array reflectors.

1. S. R. BIERMAN and E. D. CLAYTON, "Critical Experiments to Measure the Neutron Poisoning Effects of Cu and Cu-Cd Plates," BNWL-B-338 (Jan. 1974).
2. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Critical Experiments Measuring the Reactivity Worths of Materials Commonly Encountered as Fixed Neutron Poisons," BNWL-2129 (Oct. 1976).
3. S. R. BIERMAN, E. D. CLAYTON, and B. M. DURST, "Critical Separation Between Subcritical Clusters of 2.35 wt% ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," PNL-2438 (Oct. 1977).
4. S. R. BIERMAN, E. D. CLAYTON, and B. M. DURST, "Critical Separation Between Subcritical Clusters of 4.29 wt% of ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," NUREG/CR-0073 (May 1978).
5. ROBERT E. ROTHE and INKI OH, "Benchmark Critical Experiments on Highly Enriched Uranyl Nitrate Solution Systems," NUREG/CR-0041 (Apr. 1978).
6. J. T. THOMAS, Ed., *Nuclear Safety Guide, TID-7016, Revision 2*, NUREG/CR-0095 (June 1978).

Pacific Northwest Laboratory Experimental Criticality Program on LWR Fuel Element Systems, S. R. Bierman (BNWL)

In 1976, the U.S. Nuclear Regulatory Commission (NRC) began funding an experimental research program at the Hanford Critical Mass Laboratory. The objective of this program was, and still is, to obtain experimental criticality data that could be used for verifying criticality calculations on transportation and storage systems involving light-water reactor (LWR)-type fuel elements. The experiments are consequently designed to simulate such systems.

The first experimental data obtained under this program were reported at the 1977 San Francisco Winter Meeting of

the American Nuclear Society.¹ Since that time, three documents²⁻⁴ have been published covering the results of experiments performed to date. A review of the overall program was given in a paper⁵ presented in June 1978 at the American Nuclear Society's Annual Meeting in San Diego, California. The intent of this current paper is to review the overall program again (briefly), and to discuss the work performed since the last review, including experiments planned and proposed for the future.

The criticality experiments are performed with subcritical clusters (simulating fuel bundles) of UO₂ rods in water. Experimental data are obtained at two ²³⁵U enrichments to provide data over the typical LWR enrichment range. These enrichments are 2.35 and 4.29 wt%. Data at each enrichment are obtained, generally, at two fuel-to-moderator ratios—one near optimum neutron moderation, for assessing the most favorable conditions for criticality, and one in the fuel-to-moderator range typical of the pressurized water reactor (PWR)- and the boiling water reactor (BWR)-type fuel elements, for assessing the conditions most likely to be encountered in handling LWR fuel elements. However, some very undermoderated and reduced-density moderated experiments have been proposed. The experiments are designed primarily to measure the reactivity effects of interaction between fuel clusters under a variety of conditions. These experiments and the conditions are summarized in Table II.

As can be seen in Table II, a large portion of the experiments is concerned with obtaining criticality data on systems that use fixed neutron poisons as a means of criticality control. The fixed neutron poisons on which data are being obtained in this program are shown in Table I. As indicated in Table II, data are generally obtained on the effect these neutron poisons have on the critical separation between clusters of fuel rods. The poisons are generally in the form of plates; however, some poison rod experiments are being considered.

The experiments being performed under this program provide data suitable for validating calculational techniques as required by the American National Standard—ANSI N16.1-1969 (Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors). The data from experiments completed have been widely disseminated and used. Also, from all reports to date, they appear to be in a satisfactory and usable form.

TABLE I
Neutron Absorber Materials

- Boral, coarse grain B₄C
- Boral, fine grain B₄C
- Type 304L stainless steel
- 1.1 wt% boron in Type 304L stainless steel
- 1.6 wt% boron in Type 304L stainless steel
- Copper
- 1 wt% Cd in copper
- Aluminum
- Zircaloy-4
- Cadmium
- Boroflex, coarse grain B₄C
- Boroflex, fine grain B₄C

TABLE II
Experiments Performed, Planned, or Proposed
Hanford Criticality Mass Laboratory
NRC Program

• EXPERIMENTS NEAR OPTIMUM NEUTRON MODERATION

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER WITH NEUTRON ABSORBER PLATES BETWEEN FUEL CLUSTERS

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER WITH NON-HYDROGENEOUS REFLECTOR WALLS ON EITHER SIDE OF FUEL CLUSTER ARRAY

DEPLETED URANIUM

LEAD

LOW ALLOY STEEL

B₄C IN COPPER MATRIX^a

BORATED SILICONE RUBBER^a

BORATED EPOXY^a

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER CONTAINING SOLUABLE NEUTRON POISONS^a

MEASUREMENTS TO DETERMINE THE EFFECT OF BORAL AND BOROFLEX B₄C GRAIN SIZE ON CRITICAL SIZE

• EXPERIMENTS IN PWR/ BWR NEUTRON MODERATION RANGE

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER WITH NEUTRON ABSORBER PLATES BETWEEN FUEL CLUSTERS

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER WITH NON-HYDROGENEOUS REFLECTOR WALLS ON EITHER SIDE OF FUEL CLUSTER ARRAY

DEPLETED URANIUM

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B₄C IN COPPER MATRIX^a

BORATED SILICONE RUBBER^a

BORATED EPOXY^a

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS OF FUEL RODS IMMERSED IN WATER CONTAINING SOLUABLE NEUTRON POISONS^a

MEASUREMENTS TO DETERMINE THE EFFECT OF BORAL AND BOROFLEX B₄C GRAIN SIZE ON CRITICAL SIZE^a

CRITICAL SEPARATION BETWEEN SUBCRITICAL CLUSTERS FUEL RODS IMMERSED IN WATER WITH FLUX TRAP BETWEEN FUEL CLUSTERS

CRITICAL SIZE OF FUEL CLUSTER CONTAINING REGULARLY SPACED WATER HOLES AND VOIDS

• EXPERIMENTS WITH REDUCED DENSITY INTERSITIAL MODERATION^a

• EXPERIMENTS IN THE REGION OF VERY LOW NEUTRON MODERATION^a

CRITICAL SIZE OF CLOSE PACKED FUEL RODS IMMERSED IN WATER^a

EFFECTIVENESS OF THERMAL NEUTRON ABSORBERS AS POISONS IN VERY UNDERMODERATED FUEL CLUSTERS^a

^aExperiments proposed, all others either completed or firmly scheduled.

1. S. R. BIERMAN and E. D. CLAYTON, "Critical Separation Between Subcritical Clusters of 2.35 wt% ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," *Trans. Am. Nucl. Soc.*, **27**, 422 (1977).

2. S. R. BIERMAN, E. D. CLAYTON, and B. M. DURST, "Critical Separation Between Subcritical Clusters of 2.35 wt% ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," PNL-2438 (Oct. 1977).

3. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Critical Separation Between Subcritical Clusters of 4.29 wt% ²³⁵U Enriched UO₂ Rods in Water with Fixed Neutron Poisons," NUREG/CR-0073 (May 1978).

4. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Criticality Experiments with Subcritical Clusters of 2.35 wt% and 4.2% ²³⁵U Enriched UO₂ Rods in Water with Uranium or Lead Reflecting Walls," NUREG/CR-0796 (Apr. 1979).

5. B. M. DURST, S. R. BIERMAN, and E. D. CLAYTON, "Critical Program and Data—Battelle-Pacific Northwest Laboratories," *Trans. Am. Nucl. Soc.*, **23**, 300 (1978).
6. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Critical Separation Between Subcritical Clusters of Low Enriched UO_2 Rods in Water with Fixed Neutron Poisons," *Nucl. Technol.*, **42**, 2, 237 (1979).
7. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Criticality Experiments with Subcritical Clusters of Low Enriched UO_2 Rods in Water with Uranium or Lead Reflecting Walls," *Nucl. Technol.*, to be published.

Experimental Programs at Rockwell International's RFP Critical Mass Laboratory, J. D. McCarthy (Rockwell Int)

The Critical Mass Laboratory at the Rocky Flats Plant (RFP) is operated by Rockwell International under contract to the DOE. The laboratory has been in operation since 1965 with the primary objective to provide critical data in support of plant operations. Approximately 1000 critical measurements have been made with high enriched uranium metal, uranyl nitrate, and plutonium metal. In addition, the laboratory accommodates experimental projects for other contractors or agencies, such as the present NRC-funded program.

The NRC Reference Critical Experiments program involves criticality measurements with high enriched uranyl nitrate and low enriched uranium oxide (4.5%) that can be used as benchmarks for computer validation. In the first series, experiments with single cylinders and arrays of uranyl nitrate reflected by various materials were completed and the results published. The present series involves the low-enriched uranium oxide (U_3O_8 at a density of 4.3 g U/cm^3) at $\text{H/U} = 0.75, 1.25$, and 2.0 in concrete, plastic, and minimal-steel reflectors. During the experimental phase associated with the oxide at $\text{H/U} = 0.75$, additional measurements with various moderating and absorbing materials between the oxide units were performed. The interspersed materials were steel, aluminum, and plastic, and each array was reflected by concrete or plastic. These additional measurements are not planned for the higher H/U ratios.

After completion of measurements with the oxide at $\text{H/U} = 1.25$ and 2.0 , attenuation experiments utilizing high enriched uranyl nitrate will be performed. These measurements will consist of arrays of cylinders of solution, and each array will be reflected by concrete or plastic and separated within the array with various thicknesses of the same material.

Subsequent programs, which depend on NRC funding, are being negotiated. Anticipated extension of the benchmark program would include arrays of low-enriched uranium oxide units at $\text{H/U} = 5, 15$, and 30 in concrete or plastic reflectors.

The above proposals are short term; long-term programs will depend on the future political climate and the needs of the industry. As part of the NRC program, a survey is made annually in which input from government agencies and the industry is solicited.

The laboratory is also preparing to perform critical measurements for immediate use in support of plant operations. Rockwell International has given high priority to replacing Raschig-ring-filled tanks used for storing high concentration solutions with safe geometry tanks or other arrangements that resolve some of the concerns associated with Raschig rings. The use of Raschig rings as a neutron

poison has served RFP well; however, the increased emphasis on all aspects of safety has indicated a need to consider alternative methods that would reduce the criticality concerns and the potential for industrial and radiation incidents.

The objective of the tank replacement program is to identify and resolve or minimize the operational and safety concerns. Included in this program are critical measurements on arrays of annular tanks of uranyl nitrate which will be reflected by concrete (Fig. 1). Various parameters will be investigated—number, spacing, neutron poisons—to determine the optimum design from an economic, operational, and safety aspect. The final design or designs identified in this program will be used to fabricate replacement tanks for the present Raschig-ring-filled tanks used for high concentration solutions. In secondary uses, such as collection tanks for filter plenum drains, ring-filled tanks will be permitted.

Program resources are also being allocated to consider the use of a "poisoned tube" heat exchanger for storage of fissile solution (Fig. 2). The neutron poison will be encased in the tubes and the solution will occupy the surrounding volume. The poison will not come in intimate contact with the solution and, therefore, concerns associated with corrosion, leaching, and settling of the poison will be eliminated or minimized. Another significant advantage is the increased storage density over that provided by safe geometry tanks, which will require spacing to control interaction between tanks. The goal will be to determine a poison density that will permit an infinite array of poisoned tube tanks. This method of storage will also require consideration of operational concerns in addition to performing criticality measurements. Annular tanks are anticipated to be a suitable replacement for the present ring-filled tanks; however, the poisoned tube tank may be selected if it appears to be a more viable alternative.

These programs, both NRC and DOE plant funded, are scheduled for the next few years. Longer term proposals

Fig. 1. Annular tank.

Fig. 2. Poisoned tube tank for solution storage.

depend on several factors—political and economic. Although many are anticipated and planned, no definite priorities have been set and no firm commitments have been made at this time. However, the need exists to provide solid benchmark measurements. The nuclear industry cannot afford “surprises”—especially in the field of nuclear criticality safety.

Methods Development for Criticality Safety Analysis, R. M. Westfall (ORNL)

The purpose of this paper is to present a status review on the work at Oak Ridge to develop improved methods for performing multigroup, discrete-ordinates, and Monte Carlo criticality safety analyses. In the area of multigroup cross-section preparation this work entails the testing of ENDF/B-IV-based and other cross-section libraries in the SCALE¹ system, the development of improved cross-section processing methods for the AMPX² system, and the generation of an ENDF/B-V-based library. In the area of systems analysis, this work entails improvements to the one-dimensional discrete-ordinates code XSDRNP-S, the testing of the combinatorial geometry version of KENO, KENO-IV/CG,³ and development of an advanced version of KENO, KENO-V. Also presented is a brief review of the existing criticality safety analytical sequences in the SCALE system, CSAS1 and CSAS2, and development of the advanced analytical sequences CSAS3 and CSAS4.

MULTIGROUP CROSS SECTIONS

The ENDF/B-IV-based cross-section libraries have been validated through the analysis^{4,5} of approximately 130 critical experiments. Recently, 70 experiments designed to simulate nuclear fuel shipping casks and storage pools have been analyzed with the 27-group ENDF/B-IV, the 123-group GAM-THERMOS, and the 16-group Hansen-Roach-Bondarenko cross-section libraries. The SCALE system card-image input for these experiments has been compiled into a data bank. This was done with the CESAR program which intersperses the input sets with descriptive information and provides the user with the capability to search on the basis of fuel type, enrichment, and geometry.

New modules being developed for the AMPX system to process ENDF/B data into multigroup cross-section libraries have been named TUMS and PRUDE-TABOO. TUMS develops transfer arrays from thermal scattering kernel information by performing a double integration over a fine energy-pointwise mesh. PRUDE-TABOO develops resonance shielding factors for the unresolved resonance energy range.

The generation of a new multigroup library based on ENDF/B-V data is under way. The reference library will contain 227 energy groups. This group structure includes the boundaries for the 218-group structure, plus additional groups above 6 MeV and below 0.01 eV. The high-energy groups have been inserted to better account for reaction

thresholds in the analyses of concrete shields. The low-energy groups were inserted to determine their importance in calculating the neutron absorption rate in gadolinium. The weight functions applied in generating the multigroup data are slightly different than those applied in generating the 218-group library. The 227-group ENDF/B-V library will be collapsed to the 27-energy-group structure. This new 27-group library will be validated through the analysis of the same set of critical experiments used to validate the ENDF/B-IV library.

SYSTEMS-REACTIVITY ANALYSIS

The XSDRNPM-S module is used to perform cell averaging in problem-dependent cross-section processing and to calculate the multiplication factor for those systems that can be adequately represented by a one-dimensional model. A new B_n calculation for the transverse leakage has been implemented into XSDRNPM-S. For systems in which the transverse leakage is important (short cylinders and slabs), the new treatment provides a more accurate leakage estimation than that given by a geometric buckling factor. Also, a new coarse-mesh rebalance scheme is being developed for XSDRNPM-S. This scheme is designed to treat bands of energy groups in performing the rebalance. The motivation here is to accelerate the convergence of problems involving thermal upscatter.

The KENO-IV/CG module has been developed to combine the geometric modeling capabilities of the combinatorial and the KENO mixed-box geometry packages. The KENO mixed-box geometry is particularly efficient in performing fuel pin lattice calculations, while the combinatorial geometry offers more flexibility in the variety and relative position of the geometric bodies that can be modeled. The use of the two geometries in conjunction in KENO-IV/CG is a complicated procedure requiring some familiarity with the input for each geometry package and experience with the interfacing mechanisms in KENO-IV/CG. The program has received extensive in-house use, particularly in the analysis of the split-table critical experiments being performed at Rocky Flats. Presently, the program is being modified to reduce numerical precision problems occurring in the combinatorial tracking routines. The program will be released for distribution after these modifications are completed.

The KENO-V module is being developed for application in the advanced criticality safety analytical sequences in the SCALE system. This module features a supergrouping scheme for minimizing the computer memory requirements associated with the application of cross-section libraries with large-group structures. Unlike KENO-IV, KENO-V treats anisotropic scattering to the full expansion order represented in the cross-section transfer arrays. An advanced geometry package is being developed for KENO-V which will relax some of the restrictions of KENO mixed-box geometry while retaining most of its efficiency.

SCALE SYSTEM ANALYTICAL SEQUENCES

The CSAS1 and CSAS2 sequences have been developed to perform problem-dependent cross-section processing followed by the calculation of the system multiplication factor by either a one-dimensional discrete-ordinates analysis in XSDRNPM-S or by a multidimensional Monte Carlo analysis with KENO-IV. The CSAS3 and CSAS4 sequences are being developed to provide reactivity searches. For a given system geometry and fuel content, CSAS3 will determine the most reactive system as a function of the content of an admixed material. The CSAS4 sequence employs KENO-V in determination of the most reactive fuel-pin lattice pitch in a shipping cask or storage pool environment.

Along with the additional analytical sequences, a CRT interactive input processor and a geometry graphics package are being developed for the SCALE system.

1. R. M. WESTFALL et al., "SCALE, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200 (1979).
2. N. M. GREENE et al., "AMPX, A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B," ORNL-TM-3706 (1976).
3. J. T. WEST et al., "KENO-IV/CG, The Combinatorial Version of the KENO Monte Carlo Criticality Safety Program," NUREG/CR-0709 (1979).
4. N. F. CROSS et al., "Validation of Criticality Safety Broad-Group Library Using Uranium Systems," *Trans. Am. Nucl. Soc.*, **27**, 406 (1977).
5. R. M. WESTFALL, "Design Criteria for the 218-Group Criticality Safety Reference Library," ORNL/RSIC-41 (1978).

Dissolution and Storage Experimental Program with U[4.75]O₂ Rods, J. C. Manaranche, D. Mangin, L. Maubert, G. Colomb, G. Poullot (CEA-France)

At the criticality facility of Valduc (CEA-France), the experimental program is continuing with the same 4.75 wt% ²³⁵U enriched UO₂ rods as those already described.¹

This paper presents results of two series of recent experiments. The purpose of the first series is to provide data for determining criticality safety conditions of the dissolution of fuel rods in nitric acid. The results of experiments are relative to regular lattices of fuel rods with square pitches, immersed in water plus NaNO₃.

The purpose of the second series is to simulate accidental moderation of nonirradiated fuel rod storage. The results of experiments are relative to four clusters of 18 X 18 rods with interposition of various hydrogenous compounds.

Critical configurations were carried out with regular rod lattices for several pitches (12.6, 16.0, 21.0, 25.2 mm) immersed in aqueous NaNO₃ solutions. Experimental and theoretical results are shown in Table I, and the experimental arrangement is shown in the figure of Ref. 1. Benchmark calculations, utilizing the MORET² computer code and cross sections (for homogenized lattice equivalent medium) generated and contracted in 16 energy groups (Hansen-Roach cutting) by the APOLLO code,³ agree with the experiment.

The experimental arrangement of four fuel clusters, with interposition of hydrogenous compounds, is shown in Fig. 1.

Waterproof boxes, in crossing style, with 3-mm aluminum walls, with various thickness—25, 50, 100 mm—were interposed between four 18 X 18 rod fuel clusters (square pitch 13.5 mm). Several hydrogenous compounds were put inside the boxes: air, expanded polystyrene (C₈H₈)_n, powder and small balls of polyethylene (CH₂)_n, and water. The critical approach parameter is the height of water that immerses the whole. These compounds are at the same level as water in clusters.

Other experiments were carried out with these same four clusters in water without a box. Experimental results are mentioned in Table II as well as theoretical results using the previous calculation method.

The curves that connect the critical height to the hydrogen concentration C(H) for each distance Δ(H) between

TABLE I
Lattice of U (4.75) O₂ Rods in H₂O + NaNO₃

Lattice		Solution			Apollo Results		Apollo-Moret Results	
Square Pitch (mm)	Configuration	Density (g/cm ³)	Concentration NaNO ₃ (g/cm ³)	H _c ^a (mm)	B _m ² (cm ⁻²)	K _∞	K _{eff} ± 2σ	G _f ^b
12.6	24 × 25	1.1521	0.247	809.0 ± 0.6	0.01036	1.447	1.005 ± 0.010	14.33
16.0	18 × 18	1.1503	0.243	761.1 ± 0.6	0.01243	1.473	0.998 ± 0.011	15.00
	17 × 17	1.1523	0.246	762.0 ± 0.6	0.00981	1.357	0.993 ± 0.010	15.32
21.0	16 × 16	1.1094	0.178	895.3 ± 0.6	---	---	---	---
	15 × 16	1.0565	0.090	848.2 ± 0.7	0.01094	1.379	0.996 ± 0.011	15.34
25.2	20 × 21	1.1523	0.246	858.9 ± 0.6	0.00592	1.215	1.003 ± 0.009	15.43

^aWater critical height, level zero: bottom of fuel part.

$$G_f = \left[\sum_{i=1}^{16} i \times F(i) \right] / \left[\sum_{i=1}^{16} F(i) \right]; [F(i) \text{ is the total number of fission neutrons in group } i].$$

Fig. 1. Experimental setup.

clusters are rather attractive. They state a located minimum when $\Delta(H) \times C(H) \approx \text{constant} \approx 0.22 \text{ g H} \cdot \text{cm}^{-2}$. So, for values higher than 10 cm, the minimum is located for very weak concentrations in hydrogen ($<0.02 \text{ g cm}^{-3}$). This latter result is very important because it confirms the hypotheses that have to be taken into account for calculating PWR new fuel storage; in this case, the theoretical conditions of

optimal moderations are needed. Otherwise the theoretical results agree with the experiment.

1. J. C. MANARANCHE, D. MANGIN, L. MAUBERT, G. COLOMB, and G. POULLOT, "Critical Experiments with Lattices of 4.75 wt% ²³⁵U Enriched UO₂ Rods in Water," *Trans. Am. Nucl. Soc.*, **27**, 302 (1978).

TABLE II

Interposition of Hydrogenous Compounds Between Four Clusters of 18 X 18 Rods, Square Pitch 13.5 mm

Δ^a (cm)	Compounds			Water Critical Height (mm)	Apollo-Moret $k_{eff} \pm 2\sigma$
	Nature	Density (g/cm ³)	"H" Concentration (g/cm ³)		
0.0	water	1.0	0.1119	238.0 $\pm$ 0.6	1.010 $\pm$ 0.011
2.5	box + air	0	0	290.3 $\pm$ 0.9	1.005 $\pm$ 0.010
	box + (C ₈ H ₈) _n	0.0323	0.0025	286.1 $\pm$ 0.8	0.987 $\pm$ 0.010
	box+(CH ₂) _n (powder)	0.2879	0.0414	269.8 $\pm$ 0.6	-
	box+(CH ₂) _n (balls)	0.5540	0.0800	255.4 $\pm$ 0.6	0.995 $\pm$ 0.010
	box + water	1.0	0.1119	256.6 $\pm$ 0.7	-
	water	1.0	0.1119	244.8 $\pm$ 0.6	1.006 $\pm$ 0.011
5.0	box + air	0	0	344.8 $\pm$ 0.7	1.000 $\pm$ 0.012
	box + (C ₈ H ₈) _n	0.0262	0.0020	343.9 $\pm$ 0.8	
	box+(CH ₂) _n (powder)	0.3335	0.0480	301.6 $\pm$ 0.6	
	box+(CH ₂) _n (balls)	0.5796	0.0833	307.3 $\pm$ 0.8	
	box + water	1.0	0.1119	327.8 $\pm$ 0.8	
	water	1.0	0.1119	314.7 $\pm$ 0.6	
10.0	box + air	0	0	460.8 $\pm$ 0.7	0.996 $\pm$ 0.010
	box + (C ₈ H ₈) _n	0.0288	0.0022	456.2 $\pm$ 0.8	0.987 $\pm$ 0.010
	box+(CH ₂) _n (powder)	0.3216	0.0464	420.5 $\pm$ 0.6	1.011 $\pm$ 0.012
	box+(CH ₂) _n (balls)	0.568	0.0816	499.4 $\pm$ 0.6	1.010 $\pm$ 0.010
	box + water	1.0	0.1119	641.2 $\pm$ 0.9	0.998 $\pm$ 0.011
	water	1.0	0.1119	643.4 $\pm$ 0.8	

^aSeparation between fuel clusters (see Fig. 1); thickness of hydrogenous compounds: $\Delta(H) = \Delta - 0.6$ (cm).

2. R. CAIZERGUES, J. P. MARIOT, J. MORET-BAILLY, G. POULLOT, and J. R. TEILLET MORET, "Un Programme Monte-Carlo pour le Calcul Rapide de Coefficients de Multiplication Effectifs de Milieux Fissiles de Géométrie Complexe," CEA-R 4752 (1976).

3. A. HOFFMANN, F. JEANPIERRE, A. KAVENOKY, M. LIVOLANT, and H. LORRAIN, "APOLLO—Code Multigroupe de Résolution de l'Équation du Transport pour les Neutrons Thermiques et Rapides," CEA-N 1610 (1973).

Utilization of B&W Critical Facility in Support of Spent-Fuel Storage, G. S. Hoovler, M. N. Baldwin, R. L. Eng (B&W)

Close-packed storage of LWR fuel assemblies is needed to expand the capacity of existing underwater storage pools. This increased capacity is required to accommodate the large volume of spent fuel that arises from prolonged on-site storage. To provide benchmark criticality data in support of this effort, the Babcock & Wilcox Company was awarded

Fig. 1. Benchmark core with B_4C pins.

Fig. 2. Benchmark core with metal isolation sheets.

a contract by the U.S. Energy Research and Development Administration (now the U.S. Department of Energy) to conduct a series of critical experiments that simulate close-packed storage of LWR-type fuel. Criticality calculations using the Monte Carlo code KENO-IV were performed to provide an analytical basis of comparison with the experimental data.

The experiments were performed at B&W's Lynchburg Research Center in the Company's CX-10 critical facility. CX-10 is a tank-type critical facility licensed for the performance of critical experiments with water-moderated UO_2 and mixed oxide (Th-U) fuel lattices. Although the facility is licensed to use either light water or heavy water as a moderator, only light water was used in this series of experiments.

Low enriched (2½%) UO_2 fuel pins were used to construct 20 critical loadings that simulated a variety of close-packed LWR fuel storage configurations. Each critical loading consisted of nine LWR-type fuel assemblies (clusters) grouped in a 3 X 3 array. Both the spacing and material between the fuel clusters were varied to provide numerous critical configurations. The spacing between the fuel clusters was varied from 0 to about 6.5 cm. The materials used to isolate the fuel clusters included (besides water) B_4C pins, stainless-steel sheets, and borated aluminum sheets. Photographs of cores using B_4C pins and metal isolation sheets are shown in Figs. 1 and 2, respectively.

These experiments were completed in March 1979, and provide a body of benchmark data suitable for validating calculational methods used to predict K_{eff} of nuclear fuel stored in close-packed, heavily poisoned arrays. All pertinent data for each critical assembly are documented in sufficient detail to permit their use in validating calculational methods according to ANSI standard N16.9-1975. These data are being made available publicly through DOE's Technical Information Center.

As a followup to this work, B&W has proposed and received a letter of intent for a contract extension by DOE that would broaden the current study to include tightly packed fuel pin storage. Such storage would result from

Nonmoderator to moderator ratio = 9.7

Fig. 3. Module with fuel rods touching on triangular pitch.

dismantling LWR-type fuel assemblies and reassembling the pins from several assemblies into tightly packed bundles. Figure 3 shows how such a tightly packed bundle might appear. During this program extension, we will study critical configuration of fuel bundles consisting of low enriched UO_2 fuel pins packed one against the other on a triangular and square pitch. Maximum fuel storage density can be attained with this tightly packed configuration.

Preparation for this series of experiments is now under way. The experimental work will require about one year to complete, and is planned to commence at the beginning of FY 1980.

Experimental Programs at the Hanford N Reactor in Support of Spent-Fuel Storage, H. Toffer (UNC Nucl Ind)

With the shutdown of the Purex separations plant, deactivation of the single-pass reactors, and operation of N Reactor with well-established and fixed fuel designs, the need for additional critical mass data has been limited to the support of spent-fuel storage. The limitation of available storage for spent N Reactor fuel has placed large incentives on optimized use of existing storage facilities. Therefore, the experimental effort in recent years and in the immediate future has and will be in the direct support of safe and efficient storage of spent fuels. The experimental work can be divided into five categories.

1. benchmark subcritical measurements with spent fuel
2. *in situ* reactivity assessment of storage and transport configurations
3. *in situ* verification of fixed neutron absorber in a spent-fuel pool
4. axial fuel burnup assessment and isotopic analysis of spent fuel
5. monitoring of basin sludges and filter deposits for fissile isotope concentration.

A benchmark subcritical measurement was performed with spent and unirradiated fuel of the same design under identical conditions. The measurements served as a confirmation of calculations predicting a critical mass ratio of 2 between spent and unirradiated fuel.¹ The experimental results provided a basis for the use of spent-fuel critical mass values in establishing safety margins for spent-fuel storage.² Any significant future fuel assembly design or enrichment modification would necessitate a repeat of the measurements.

In situ pulsed-neutron measurements have been performed on shipping cask loadings with spent fuel and are planned for special high-density storage configurations in the N Reactor supplemental storage basin. The measurements establish the subcriticality of the particular configuration. In the case of the spent-fuel cask loading, based on the measurements, the number of shipments required was reduced from 6 to 2 (Ref. 3). The measurements in the storage basin will confirm calculations permitting an increase in the density of stored fuel by 130%.

In situ neutron gauging measurements have been carried out in the N Reactor fuel storage basin to confirm the presence of boron in the borated concrete cubicle walls. The cubicles were installed in the N Reactor spent-fuel basin over ten years ago to provide high-density fuel storage within the required nuclear criticality safety criteria.⁴ The neutron transmission measurements using a ^{252}Cf source and a fission chamber in selected cubicles confirmed adequate presence of boron in the cubicle walls for criticality control. In the

future, additional neutron gauging is planned in the basin and of a cubicle wall removed from the basin.

Axial fuel burnup measurements are performed routinely on selected fuel charges following discharge from the reactor. The information derived from these measurements is essential to assigning a safe exposure value to fuel assemblies handled and stored under spent-fuel critical mass parameters. The measurements consist of axial gross gamma scans of fuel assemblies making up an individual fuel charge. Efforts are under way to develop techniques that will yield axial fissile isotope distributions in spent fuel in addition to the gross gamma counts.

The fuel used at N Reactor is metallic uranium; any fuel that comes in contact with water will oxidize. Damage to cladding can result in fuel material loss in prolonged storage. The material is transported by the water away from the fuel and can end up as a sediment on the basin floor or as a deposit in filter cartridges, in sand filter media, or in ion-exchange resins. A surveillance program is in place monitoring fissile material concentrations in the basin water, sediments, and filter media. Effects on criticality control are evaluated using the observed concentrations.

Measurement programs in all five areas discussed aid the nuclear criticality safety efforts in assuring safe storage and handling of fuel. At the same time, the results obtained provide for maximum use of available storage space by the incorporation of spent-fuel critical mass parameters in establishing safety margins.

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Data and Analysis for Nuclear Criticality Safety

Scale System Cross-Section Validation with Shipping-Cask Critical Experiments, R. M. Westfall, J. R. Knight (ORNL)

This paper presents results obtained in the analysis of seventy recently performed critical experiments using three cross-section libraries available in the SCALE system.¹ These experiments²⁻⁴ were designed to provide criticality data on systems simulating conditions associated with LWR fuel element shipping casks and storage pools.

Four series of experiments were analyzed. Each of the experiments involved the determination of the critical spacing between three clusters of UO_2 rods arranged in a linear array. The first series of experiments was performed with 2.35 wt% enriched UO_2 rods at a lattice pitch of 2.032 cm. Neutron absorber plates were inserted between the central cluster and the two outer clusters. Parametric variations included the thickness and composition of the absorber plates and their spacing relative to the central cluster. The second series of experiments involved the absorber plates inserted between clusters of 4.29 wt% enriched UO_2 rods at a lattice pitch of 2.54 cm. The third and fourth series of experiments involved each type of fuel rod lattice positioned between lateral reflecting walls simulating biological shields of lead and depleted uranium. Here, the parametric variations were the type of reflecting wall and the water-filled gap between the wall and the side of the linear array.

The analyses were performed with the CSAS2 analytical sequence in the SCALE system. This analytical sequence involves the execution of the BONAMI-S and NITAWL-S modules for problem-dependent cross-section processing and the KENO-IV module for a Monte Carlo determination of the system multiplication factor.

The three cross-section libraries utilized in the analyses were the 27-group ENDF/B-IV library, the 123-group GAM-THERMOS library, and the 16-group Hansen-Roach library.⁵ The 27-group library is a subset of the 218-group Criticality Safety Reference Library.⁶ The 27-energy-group structure was determined through an extensive series of model calculations.⁷ The 123-group GAM-THERMOS library is an older data set based on the GAM-II⁸ library and early versions of ENDF/B. The 16-group Hansen-Roach library has been cast in the form of tabulated values for the cross sections at infinite dilution and Bondarenko shielding factors. The background cross sections for resonance shielding were determined on a groupwise basis in the BONAMI-S module. The BONAMI-S formulation includes a correction to the background cross section for heterogeneity effects. Resonance shielding for the 27- and 123-group libraries was performed with the Nordheim method in the NITAWL-S module.

The results of the analyses are given in Tables I and II.^a Each of the 210 analyses involved the calculation of approximately 30,000 neutron histories. Four major conclusions can be drawn from these results.

1. For the 2.35 wt% enriched UO_2 rod lattices, the average of the calculated multiplication factors was 0.994 for the 27-group ENDF/B-IV library, 1.002 for the 123-group GAM-THERMOS library, and 1.004 for the 16-group Hansen-Roach library.

2. For the 4.29 wt% enriched UO_2 rod lattices, the average of the calculated multiplication factors was 0.988 for the 27-group ENDF/B-IV library, 0.989 for the 123-group GAM-THERMOS library, and 0.993 for the 16-group Hansen-Roach library.

3. Each of the six sets of 35 analyses is sufficiently large that a normal distribution of the results can be observed. Virtually all of the values lie within three standard deviations of the average values. Furthermore, a significant number of the results differ from the average values by more than two standard deviations.

4. Each of the three libraries is adequate for performing criticality safety analyses for systems similar to the ones studied. The analytical biases to be applied in the use of these libraries are functions of both fuel enrichment and neutron moderation level.

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3. S. R. BIERMAN et al., "Critical Separation Between Subcritical Clusters of 4.29 wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," NUREG/CR-0073 (1978).
4. S. R. BIERMAN et al., "Criticality Experiments with Subcritical Clusters of 2.35 wt% and 4.29 wt% ^{235}U Enriched UO_2 Rods in Water with Uranium or Lead Reflecting Walls," NUREG/CR-0796, PNL-2827 (1979).
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6. W. E. FORD III et al., "A 218 Neutron Group Master Cross Section Library for Criticality Safety Studies," ORNL/TM-4 (1976).
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8. G. D. JOANOU et al., "GAM-II, A B_3 Code for the Calculation of Fast-Neutron Spectra and Associated Multigroup Constants," GA-4265 (1963).

Reactivity Effect of Cross-Section Processing for Moist Bulk-Oxide Criticals, J. A. Bucholz, R. M. Westfall (ORNL)

Good multigroup cross-section data are obviously necessary if one is to perform accurate analyses of critical or slightly subcritical systems. Generation of this multigroup data from the point data requires careful weighting of all processes over the resonance range. The purpose of this paper is to show how the flux spectrum as calculated by the Nordheim method¹ and by a more exact method can ultimately affect the calculated multiplication and/or critical mass of slightly moderated bulk-oxide systems.

^aSee p. 369 for Table I and p. 370 for Table II.

TABLE I (Paper 1)

CSAS2 Analyses of 2.35 wt% Enriched UO₂ Three-Cluster Lattices

Exp. No.	Plate Material	Plate Thickness (cm)	Array Size	Plate-to-Center Cluster Gap (cm)	Critical Separation Between Clusters (cm)	Calculated Multiplication Factor		
						27-Group NDFB4	123-Group GAMTH	HANSEN-ROACH
15	--	--	20 x 17	--	11.92	0.992 ± 0.004	1.008 ± 0.004	0.995 ± 0.004
5	--	--	20 x 16	--	8.39	0.990 ± 0.004	1.005 ± 0.004	1.000 ± 0.004
27	SS304L	0.302	20 x 16	0.645	7.42	0.999 ± 0.004	0.997 ± 0.004	1.005 ± 0.004
26	SS304L	0.302	20 x 16	4.042	7.76	1.004 ± 0.004	0.999 ± 0.004	1.011 ± 0.004
28	SS304L	0.485	20 x 16	0.645	6.68	0.991 ± 0.004	1.002 ± 0.004	1.007 ± 0.004
29	SS304L	0.485	20 x 16	4.042	7.51	0.991 ± 0.004	0.994 ± 0.004	1.002 ± 0.004
24	6061 Al	0.625	20 x 16	0.645	8.67	0.996 ± 0.005	0.998 ± 0.004	1.009 ± 0.004
48	6061 Al	0.625	20 x 16	4.042	8.78	0.992 ± 0.004	0.997 ± 0.004	1.005 ± 0.004
46	Zircalloy-4	0.652	20 x 16	0.645	8.79	0.990 ± 0.004	0.996 ± 0.004	1.003 ± 0.004
47	Zircalloy-4	0.652	20 x 16	4.042	8.78	0.994 ± 0.004	0.998 ± 0.004	0.996 ± 0.005
31	Copper	0.646	20 x 16	0.645	6.62	0.987 ± 0.004	1.001 ± 0.005	1.008 ± 0.005
12	Copper	0.646	20 x 16	4.442	7.51	0.992 ± 0.004	1.000 ± 0.004	0.999 ± 0.004
43	Copper	0.337	20 x 15	0.645	6.88	0.995 ± 0.004	1.004 ± 0.004	1.016 ± 0.004
44	Copper	0.337	20 x 15	4.042	7.00	0.986 ± 0.004	1.007 ± 0.004	1.006 ± 0.004
41	Cu(0.989 Cd)	0.357	20 x 15	0.645	5.15	0.992 ± 0.004	1.014 ± 0.004	1.015 ± 0.004
36	Cd	0.061	20 x 17	0.645	6.74	0.991 ± 0.004	1.005 ± 0.004	1.009 ± 0.004
37	Cd	0.061	20 x 17	4.042	9.37	0.996 ± 0.004	1.006 ± 0.004	1.005 ± 0.004
50	Cd	0.0291	20 x 17	1.482	7.87	0.991 ± 0.004	0.998 ± 0.004	1.012 ± 0.004
54	Cd	0.061	20 x 17	1.482	7.60	0.987 ± 0.004	1.007 ± 0.004	1.001 ± 0.004
52	Cd	0.0901	20 x 17	1.482	7.54	0.989 ± 0.004	1.009 ± 0.004	1.005 ± 0.004
20	Boral	0.713	20 x 17	0.645	6.34	0.997 ± 0.004	1.010 ± 0.004	1.006 ± 0.004
16	Boral	0.713	20 x 17	4.442	9.03	0.986 ± 0.004	1.007 ± 0.005	1.008 ± 0.004
32	SS304L (1.05B)	0.298	20 x 17	0.645	7.56	0.994 ± 0.004	1.004 ± 0.004	1.003 ± 0.004
33	SS304L (1.05B)	0.298	20 x 17	4.042	9.62	0.996 ± 0.004	0.999 ± 0.004	1.011 ± 0.004
38	SS304L (1.62B)	0.298	20 x 17	0.645	7.36	0.992 ± 0.004	1.000 ± 0.004	0.999 ± 0.004
39	SS304L (1.62B)	0.298	20 x 17	4.042	9.52	0.994 ± 0.004	1.002 ± 0.004	1.014 ± 0.004
	Reflector Material	Reflector Thickness (cm)		Reflector-to-Fuel Cell Edge Gap (cm)				
1	U (0.19) Metal	7.65	19 x 16	0	11.83	0.998 ± 0.004	1.001 ± 0.004	1.001 ± 0.004
2	U (0.19) Metal	7.65	19 x 16	1.321	13.93	0.992 ± 0.004	1.011 ± 0.004	1.005 ± 0.004
3	U (0.19) Metal	7.65	19 x 16	1.956	14.11	1.004 ± 0.004	1.000 ± 0.003	1.002 ± 0.004
4	U (0.19) Metal	7.65	19 x 16	2.616	13.70	1.000 ± 0.004	0.999 ± 0.004	1.002 ± 0.004
5	U (0.19) Metal	7.65	19 x 16	5.405	10.69	0.990 ± 0.004	0.998 ± 0.004	0.997 ± 0.003
6	U (0.19) Metal	7.65	19 x 16	10.676	8.56	0.988 ± 0.004	0.999 ± 0.004	0.995 ± 0.005
9	Pb	10.2	19 x 16	0	13.84	0.999 ± 0.004	1.008 ± 0.004	1.008 ± 0.004
11	Pb	10.2	19 x 16	2.616	11.25	0.987 ± 0.004	1.006 ± 0.004	1.006 ± 0.004
12	--	--	19 x 16	--	8.31	0.990 ± 0.004	0.992 ± 0.004	0.992 ± 0.004

TABLE II (Paper 1)
CSAS2 Analyses of a 4.29 wt% Enriched UO_2 Three-Cluster Lattices

Exp. No.	Plate Material	Plate Thickness (cm)	Array Size	Plate-to-Center Cluster Gap (cm)	Critical Separation Between Clusters (cm)	Calculated Multiplication Factor		
						27-Group NDFB4	123-Group GMTH	HANSEN-ROACH
32	--	--	15 x 8	--	10.60	0.989 $\pm$ 0.005	0.984 $\pm$ 0.004	0.997 $\pm$ 0.004
14	SS304L	0.485	15 x 8	0.245	8.58	0.997 $\pm$ 0.004	0.989 $\pm$ 0.006	0.993 $\pm$ 0.005
13	SS304L	0.485	15 x 8	3.277	9.65	0.989 $\pm$ 0.005	0.991 $\pm$ 0.005	0.998 $\pm$ 0.005
8	SS304L	0.302	15 x 8	0.428	9.22	0.986 $\pm$ 0.005	0.993 $\pm$ 0.005	0.988 $\pm$ 0.005
7	SS304L	0.302	15 x 8	3.277	9.76	0.989 $\pm$ 0.005	0.991 $\pm$ 0.005	0.996 $\pm$ 0.005
10R	SS304L (1.05B)	0.298	15 x 8	0.432	6.10	0.982 $\pm$ 0.004	0.987 $\pm$ 0.004	1.001 $\pm$ 0.004
9	SS204L (1.05B)	0.298	15 x 8	3.277	8.08	0.991 $\pm$ 0.004	0.986 $\pm$ 0.004	0.988 $\pm$ 0.004
12	SS304L (1.62B)	0.298	15 x 8	0.432	5.76	0.982 $\pm$ 0.004	0.985 $\pm$ 0.004	0.987 $\pm$ 0.004
11	SS304L (1.62B)	0.298	15 x 8	3.277	7.90	0.988 $\pm$ 0.004	0.992 $\pm$ 0.005	0.996 $\pm$ 0.004
6	6061 Al	0.625	15 x 8	0.105	10.72	0.990 $\pm$ 0.005	0.983 $\pm$ 0.006	1.003 $\pm$ 0.004
5	6061 Al	0.625	15 x 8	3.277	10.77	0.985 $\pm$ 0.005	0.987 $\pm$ 0.005	0.997 $\pm$ 0.005
30	Zircalloy-4	0.652	15 x 8	0.78	10.92	0.987 $\pm$ 0.005	0.984 $\pm$ 0.005	0.990 $\pm$ 0.004
29	Zircalloy-4	0.652	15 x 8	3.277	10.86	0.986 $\pm$ 0.005	0.979 $\pm$ 0.005	0.991 $\pm$ 0.004
16	Copper	0.646	15 x 8	0.084	8.15	0.989 $\pm$ 0.005	0.990 $\pm$ 0.005	0.999 $\pm$ 0.004
15	Copper	0.646	15 x 8	3.277	9.42	0.987 $\pm$ 0.005	0.992 $\pm$ 0.005	0.994 $\pm$ 0.005
18	Copper	0.337	15 x 8	-0.057 ^a	8.48	0.985 $\pm$ 0.005	0.991 $\pm$ 0.004	0.983 $\pm$ 0.004
17	Copper	0.337	15 x 8	4.241	9.62	0.979 $\pm$ 0.005	0.989 $\pm$ 0.004	0.988 $\pm$ 0.005
20	Cu (0.989 Cd)	0.357	15 x 8	-0.057 ^a	6.66	0.987 $\pm$ 0.004	0.986 $\pm$ 0.005	0.992 $\pm$ 0.004
19	Cu (0.989 Cd)	0.357	15 x 8	4.241	8.35	0.990 $\pm$ 0.005	0.992 $\pm$ 0.004	0.989 $\pm$ 0.004
26	Cd	0.0291	15 x 8	0.7009	5.93	0.988 $\pm$ 0.004	0.987 $\pm$ 0.005	0.994 $\pm$ 0.005
25	Cd	0.0291	15 x 8	3.277	7.42	0.986 $\pm$ 0.004	0.994 $\pm$ 0.004	1.001 $\pm$ 0.004
28	Cd	0.0610	15 x 8	0.669	5.96	0.993 $\pm$ 0.004	0.993 $\pm$ 0.004	0.991 $\pm$ 0.004
27	Cd	0.0610	15 x 8	3.277	7.42	0.984 $\pm$ 0.004	0.985 $\pm$ 0.004	0.998 $\pm$ 0.004
22	Cd	0.0901	15 x 8	0.640	5.87	0.988 $\pm$ 0.005	0.986 $\pm$ 0.004	0.991 $\pm$ 0.004
21	Cd	0.0901	15 x 8	3.277	7.38	0.992 $\pm$ 0.005	0.995 $\pm$ 0.004	0.993 $\pm$ 0.004
24	Cd	0.2006	15 x 8	0.529	5.63	0.988 $\pm$ 0.004	0.990 $\pm$ 0.005	1.000 $\pm$ 0.005
23	Cd	0.2006	15 x 8	3.277	7.28	0.990 $\pm$ 0.005	0.992 $\pm$ 0.005	0.986 $\pm$ 0.004
31	BORAL	0.713	15 x 8	3.277	6.72	0.997 $\pm$ 0.005	0.996 $\pm$ 0.005	0.989 $\pm$ 0.005
	Reflector Material	Reflector Thickness (cm)		Reflector-to-Fuel Cell Edge Gap (cm)				
14	U (0.19) Metal	7.65	13 x 8	0.0	15.38	0.983 $\pm$ 0.004	1.013 $\pm$ 0.004	0.990 $\pm$ 0.004
16	U (0.19) Metal	7.65	13 x 8	3.912	18.05	0.990 $\pm$ 0.005	0.990 $\pm$ 0.004	1.003 $\pm$ 0.004
17	U (0.19) Metal	7.65	13 x 8	5.405	13.49	0.985 $\pm$ 0.004	0.990 $\pm$ 0.004	0.992 $\pm$ 0.004
19	Pb	10.2	13 x 8	0.0	20.62	0.993 $\pm$ 0.004	1.001 $\pm$ 0.004	0.996 $\pm$ 0.005
21	Pb	10.2	13 x 8	1.321	19.04	0.989 $\pm$ 0.005	0.993 $\pm$ 0.005	1.006 $\pm$ 0.004
22	Pb	10.2	13 x 8	5.405	10.30	0.979 $\pm$ 0.006	0.977 $\pm$ 0.005	0.995 $\pm$ 0.005
23	--	--	13 x 8	--	8.24	0.974 $\pm$ 0.005	0.978 $\pm$ 0.005	0.986 $\pm$ 0.004

^aGap lies within lattice cell boundary.

In the present work, self-shielded multigroup cross sections were generated using the Nordheim method as implemented in the NITAWL code² and by a more exact method as implemented in the ROLAIDS code.³ To eliminate geometric effects, both codes were used to generate cross sections for an infinite homogeneous media of U_3O_8 and water ($U = 4.65\% \text{ }^{235}U$).

The slowing-down treatment in ROLAIDS is based on the kinematics of elastic scattering. Within the resolved energy range, 20,000 to 50,000 energy points may be used to represent the actual cross section of the mixture. Assuming a linear variation of the scattering density, $\phi(E)\Sigma_s(E)$, between these ultrafine-mesh energy points, the slowing down integrals shown in Eq. (1) may be evaluated

$$\phi(E)\Sigma_T(E) = \sum_i^{\text{all nuclides}} \left(\frac{1}{1 - \alpha_i} \right) \int_E^{E/\alpha_i} \phi(E')\Sigma_{s,i}(E') \frac{dE'}{E'} \quad (1)$$

analytically by marching from high to low energies. The resulting flux spectrum $\phi(E)$ may then be used to weight the multigroup cross-section data:

$$\sigma^E = \left(\frac{1}{\phi^E} \right) \int_E^{E^E-1} \phi(E')\sigma(E')dE' \quad (2)$$

Note that the cross-section data for each nuclide in the mixture is weighted (i.e., shielded) by the same flux spectrum, $\phi(E)$.

The Nordheim method as implemented in the NITAWL code proceeds in much the same fashion, with several key exceptions.

1. A separate slowing down/self-shielding calculation is performed for each resonance nuclide in a mixture. In this case, one such calculation would be performed with ^{235}U as the absorber and another would be performed with ^{238}U as the absorber.

2. Each resonance is treated separately and assumed to be isolated from the others.

3. Nonresonance nuclides are not self-shielded and retain the same $1/E$ or $1/E\Sigma_T$ weighting they had when the material-independent multigroup library was created.

4. For heterogeneous systems (not considered here), one has additional approximations.

TABLE I

k_{∞} and Critical Mass for Several Moist Mixtures of Uranium Oxide, as Calculated Using NITAWL and ROLAIDS Data

Type of Data Used for Each Nuclide				k_{∞}		
^{235}U	^{238}U	H	O	H/U = 0.75	H/U = 1.25	H/U = 2.00
R	R	R	R	1.02523	1.12021	1.21493
N	N	N	N	1.04431	1.13908	1.23157
$\Delta\%$				1.9% ^a	1.7%	1.4%
R	R	R	R	1.02523	1.12021	1.21493
R	R	R	N	1.02807	1.12151	1.21537
$\Delta\%$				0.3%	0.1%	0.04%
R	R	R	R	1.02523	1.12021	1.21493
R	R	N	R	1.03327	1.12745	1.22070
$\Delta\%$				0.8%	0.6%	0.5%
R	R	R	R	1.02523	1.12021	1.21493
R	N	R	R	1.03574	1.13269	1.22705
$\Delta\%$				1.0%	1.1%	1.0%
R	R	R	R	1.02523	1.12021	1.21493
N	R	R	R	1.02257	1.11794	1.21323
$\Delta\%$				-0.3%	-0.2%	-0.1%
Type of Calculation		Cross-Section Data Used		Critical/Near-Critical Mass ^b		
				H/U = 0.75	H/U = 1.25	H/U = 2.00
Critical mass (Diffusion theory)		All ROLAIDS		263.821	14.2217	3.16506
		All NITAWL		111.507	11.4291	2.83235
		$\Delta\%$		137%	24.4%	11.7%
Critical mass (S_{16} calculation)		All ROLAIDS		258.661	13.6691	2.98573
		All NITAWL		108.231	10.9674	2.66553
		$\Delta\%$		139%	24.6%	12.0%
Subcritical mass with $k = 0.950$ (S_{16} calculation)		All ROLAIDS		42.4870	7.03152	1.84166
		All NITAWL		30.3458	6.01979	1.68419
		$\Delta\%$		40.0%	16.8%	9.3%

^aDifferences of $\sim 2\%$ in k_{eff} have also been observed in analyzing the first set of bulk-oxide add-on experiments at Rocky Flats. For configuration 1B described in report No. RFP-2868,⁴ KENO yielded a k_{eff} of 1.016 ± 0.005 using the NITAWL data and a k_{eff} of 0.998 ± 0.005 using the ROLAIDS data.

^bMass in metric tons; $M = \frac{4}{3}\pi\rho R^3$, $\rho = 4.7 \text{ g/cm}^3$.

To determine the importance of these differences, two self-shielded 27-group libraries were prepared for each of three mixtures of U_3O_8 and water with H/U ratios of 0.75, 1.25, and 2.00. These mixtures closely represent those used in the bulk-oxide critical experiments now being conducted at Rocky Flats.⁴ A number of k_{∞} calculations were then performed using the NITAWL data, the ROLAIDS data, and various combinations of the two. These results are shown in Table I. Assuming that the ROLAIDS data yielded the more exact result, several conclusions may be drawn:

1. The NITAWL data generally overpredicted k_{∞} by $1\frac{1}{2}$ to 2%.
2. Results using the NITAWL data became more accurate as the moderation level was increased.
3. The key differences were due to ^{238}U and H.

It was particularly interesting to find that the Δk due to the shielding of the hydrogen data was almost as large as that due to the improved self-shielding technique used for ^{238}U . To see why the transfer matrices should be shielded (i.e., why they depend on $\phi(E)$), one must note that

$$\begin{aligned}\sigma_R(g' \rightarrow g) &= \left(\frac{1}{\phi g'}\right) \int_{Eg}^{Eg'-1} \int_{Eg'}^{Eg'-1} \phi(E') \sigma_R(E' \rightarrow E) dE' dE \quad (3) \\ &= \left(\frac{1}{\phi g'}\right) \left(\frac{1}{2A}\right) \int_{Eg'}^{Eg'-1} \phi(E') \sigma(E') \\ &\quad \times \left(\int_{\mu_c(E', E_{L,A})}^{\mu_c(E', E_{U,A})} P_R\{\mu_L[E', E(\mu_c), A]\} d\mu_c \right) dE'\end{aligned}$$

Whereas $\sigma(E')$ may be independent of E' over a broad range of energies, the quantity in brackets still depends on E' . ROLAIDS analytically evaluates this expression⁵ and appropriately shields each term in the transfer matrix.

To further illustrate how such approximations in the cross-section generation scheme can affect the multiplication and/or critical mass of slightly moderated bulk-oxide systems, the XSDRNPM code² was used to find the critical radius corresponding to each of the three mixtures. Similar calculations were also performed to determine the size of systems with a k_{eff} of 0.95. In all cases, the ratio of the predicted masses varied roughly as

$$\frac{M_{ROLAIDS}}{M_{NITAWL}} \approx \left(\frac{k_{\infty}^{NITAWL} - k_{eff}}{k_{\infty}^{ROLAIDS} - k_{eff}} \right)^{3/2} \quad (4)$$

The actual results given in Table I show that the NITAWL data underpredicted the critical masses of these systems by as much as 12, 25, and 139%.

To summarize, we have found that subtle differences in currently used cross-section processing techniques can have considerable impact on the conclusions drawn from subsequent multigroup criticality calculations involving slightly moderated mixtures of low-enriched uranium oxide. As noted in footnote a of Table I, these differences appear to be nonnegligible in the analysis of the bulk-oxide critical experiments now being conducted at Rocky Flats.

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Improved Criticality Search Techniques for Low- and High-Enriched Systems, M. J. Lorek, H. L. Dodds (Univ of Tenn), L. M. Petrie, R. M. Westfall (ORNL)

To improve the computational efficiency (i.e., accuracy versus computing cost) of performing criticality searches on low- and high-enriched systems with codes such as ANISN¹ and KENO-IV², a new automated search technique has been developed. This technique searches for the value of a parameter (e.g., lattice pitch) that corresponds to a fixed value of K-effective, or the value of a parameter that maximizes K-effective.

The technique employs a least-squares fit to a cubic polynomial on parameter values that have been previously generated either by the extended mean value theorem³ (EMVT) or by previous curve fits. The solution of the cubic for its roots at the desired value of K-effective completes one pass for the fixed value search while the solution of its derivative provides information about maximum values. This new search technique has been implemented in a FORTRAN routine called OPTMIZ which will eventually be part of a module in the SCALE⁴ system.

For the fixed value search, the user supplies an initial guess, a fixed value of K-effective, and a tolerance about K-effective. OPTMIZ then proceeds to generate four points by the EMVT method and then performs a least-squares fit to a cubic, followed by the analytical solution of the cubic for its real roots at the desired value of K-effective.

For the parameter search to maximize K-effective, the user supplies an initial guess. OPTMIZ generates four equally spaced points, performs a least-squares fit, and takes the derivative of the cubic. The resulting quadratic is solved for its roots.

Specifically, the EMVT is simply the mean value theorem³ (MVT) plus another term of Taylor's series

$$F(x) = f(a) + f'(a)(x-a) + f''(a) \frac{(x-a)^2}{2} \quad (1)$$

A few assumptions and some manipulation of Eq. (1) yield the following modifying equation:

$$R_{new} = R_{old} - \frac{\Delta R}{\Delta K} + \frac{\Delta R}{\Delta K} (1 + |K_{cal} - K_d|)^{1/2}, \quad (2)$$

where

$$\Delta R = R_i - R_{i-1}, \Delta K = K_i - K_{i-1}, K_{cal} = K_i$$

K_d = desired value of K-eff

R_{old} = previous parameter value

R_{new} = new parameter value.

Fig. 1. A comparison of the MVT, EMVT, and OPTIMZ results for a 2 X 2 X 2 array of 93.2% uranium metal cylinders.

Fig. 2. A comparison of the KENO-IV MVT results with OPTIMZ results for a 3 X 3 water-reflected PWR array.

Equation (2) is a modification of the existing linear search routine that is built into KENO-IV. This routine is based on the MVT. The MVT modifying equation is

$$R_{\text{new}} = R_{\text{old}} + \frac{\Delta R}{\Delta K} (K_d - K_{\text{cal}}),$$

where ΔR , ΔK , K_d , K_{cal} , R_{new} , and R_{old} are defined above.

Results obtained using this new technique with KENO-IV to perform the criticality calculations are presented in Figs. 1 and 2. Figure 1 compares the MVT, EMVT, and OPTMIZ results for a high enriched system of uranium metal cylinders. The experimentally determined value of the critical lattice pitch for this system² is also shown in Fig. 1. Figure 2 compares the MVT with OPTMIZ for a low enriched array of PWR fuel assemblies in water.

These results indicate that OPTMIZ is a considerable improvement in both accuracy and CPU time over the linear MVT method (the method currently used in KENO-IV). In three out of eight cases, the MVT method failed, while OPTMIZ provided answers for all cases. The reason for this improvement is that the K-effective versus pitch data for these problems fits extremely well to a cubic.

In conclusion, the new methodology described herein for performing criticality searches provides improved accuracy with reduced computing cost relative to the MVT technique. Also, this new technique provides unambiguous results, as shown in Fig. 2, where two values of pitch exist for a K-effective of 1.26. For an initial guess of 10.00 and 15.00 cm, the MVT method found only one value and quit, while OPTMIZ found both values.

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Critical Experiments Supporting Close-Packed Storage of Power Reactor Fuel, G. S. Hoovler, M. N. Baldwin, R. L. Eng (B&W)

Close-packed storage of LWR fuel assemblies is needed to expand the capacity of existing underwater storage pools. This increased capacity is required to accommodate the large volume of spent fuel that arises from prolonged on-site storage. To provide benchmark criticality data in support of this effort, the Babcock & Wilcox Company was awarded a contract by the U.S. Energy Research and Development Administration (now the U.S. Department of Energy) to conduct a series of critical experiments that simulate close-packed storage of LWR-type fuel.

Fig. 1. Benchmark core with B₄C pins.

TABLE I
Calculated and Experimental K_{eff} for Close-Packed Fuel Configuration

Core #	Fuel Cluster Separation, Pin Pitches	Isolation Material* (Besides Water)	Boron Concentration, ppm	Calculated by KENO	Measured	Calculated Minus Measured
I	-	---	0	0.998 ± 0.006	1.0002 ± 0.0005	-0.002
II	0	---	1037	1.007 ± 0.004	1.0001 ± 0.0005	+0.007
III	1	None	769	0.999 ± 0.004	1.0000 ± 0.0006	-0.001
IV	1	B ₄ C Pins	0	1.004 ± 0.007	0.9999 ± 0.0006	+0.004
XI	1	SS Sheets	510	1.015 ± 0.004	1.0000 ± 0.0006	+0.015
XIII	1	B/Al Set #5	15	1.008 ± 0.005	1.0000 ± 0.0010	+0.008
XIV	1	B/Al Set #4	92	1.003 ± 0.004	1.0001 ± 0.0010	+0.003
XV	1	B/Al Set #3	395	0.995 ± 0.005	0.9998 ± 0.0016	-0.005
XVII	1	B/Al Set #2	486	0.993 ± 0.005	1.0000 ± 0.0010	-0.007
XIX	1	B/Al Set #1	634	0.991 ± 0.004	1.0002 ± 0.0010	-0.009
V	2	B ₄ C Pins	0	1.005 ± 0.005	1.0000 ± 0.0007	+0.005
VI	2	B ₄ C Pins	0	0.998 ± 0.004	1.0097 ± 0.0012	-0.012
XII	2	SS Sheets	217	0.991 ± 0.005	1.0000 ± 0.0007	-0.009
XVI	2	B/Al Set #3	121	0.990 ± 0.005	1.0001 ± 0.0019	-0.010
XVIII	2	B/Al Set #2	197	1.005 ± 0.005	1.0002 ± 0.0011	+0.005
XX	2	B/Al Set #1	320	0.997 ± 0.005	1.0003 ± 0.0011	-0.003
VII	3	B ₄ C Pins	0	0.994 ± 0.005	0.9998 ± 0.0009	-0.006
VIII	3	B ₄ C Pins	0	1.003 ± 0.005	1.0083 ± 0.0012	-0.005
X	3	None	143	0.988 ± 0.004	1.0001 ± 0.0009	-0.012
XXI	3	B/Al Set #1	72	0.981 ± 0.004	0.9997 ± 0.0015	-0.019
IX	4	None	0	0.987 ± 0.005	1.0030 ± 0.0009	-0.016

*B/Al Set #1: 0.10 wt% nat. B
 B/Al Set #2: 0.24 wt% nat. B
 B/Al Set #3: 0.40 wt% nat. B
 B/Al Set #4: 1.26 wt% nat. B
 B/Al Set #5: 1.61 wt% nat. B

Fig. 2. Benchmark core with metal isolation sheets.

Low-enriched (2½%) UO_2 fuel pins in a light-water-moderated lattice were used to construct 20 critical loadings that simulated a variety of close-packed LWR fuel storage configurations. The critical loadings consisted of nine LWR-type fuel assemblies (clusters) grouped in a 3×3 array. Each fuel cluster consisted of a 14×14 matrix of 1.21-cm-o.d. low-enriched UO_2 fuel pins spaced on 1.64-cm centers. The fuel pins are clad with aluminum and have an active length of about 1.6 m.

Both the spacing and material between the fuel clusters were varied to provide numerous critical loadings. The spacing between the fuel clusters was varied from 0 to about 6.5 cm. The type and position of intervening material and the boric acid concentration in the moderator were adjusted to retain a critical configuration as the spacing was varied. The materials used to isolate the fuel clusters included (besides water) B_4C pins, stainless-steel sheets, and borated aluminum sheets. Photographs of cores using B_4C pins and metal isolation sheets are shown in Figs. 1 and 2, respectively. All pertinent data for each critical assembly were documented in sufficient detail to permit use of these data in validating calculational methods according to ANSI Standard N16.9-1979. These data will be available publicly through DOE's Technical Information Center.

Criticality calculations were performed to provide an analytical basis of comparison with the experimental data.

The neutron multiplication factor (K_{eff}) of each benchmark core was calculated using the Monte Carlo criticality code KENO IV.¹ The 123-group XSDRN cross-section set was used.² Resonance self-shielding was accounted for in the ^{238}U isotope only. Self-shielding corrections were made with the NITAWL code from the AMPX package.³

A comparison between the calculated and experimental values for K_{eff} is summarized in Table I. The error bars represent one standard deviation. These data show that the calculational model underestimates K_{eff} when separation between fuel clusters is greater than one pin pitch (1.64 cm), and that the degree of underestimation increases with wider spacing.

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Criticality of Intermediate Uranium Enrichments, R. A. Libby, E. D. Clayton (BNWL)

Data on the criticality of uranium in the intermediate enrichment range (from ~6 to ~93 wt% ^{235}U) from which to deduce subcritical limits for criticality control are limited. In addition, this enrichment range is not covered in the standard for Fissionable Materials Outside of Reactors¹ and only limited guidance is provided in other standard criticality safety references.^{2,3} The transition in minimum critical dimensions from solid metal systems at 93 wt% enrichment to solution and heterogeneous systems at lower enrichments has not been clearly understood. Calculations have now been completed for both uranium metal and uranium dioxide systems which provide an improved basis for criticality control in the intermediate enrichment range.

The calculations used the computer code EGGNIT⁴ with an ENDF/B-IV library to generate 18-group cross sections. The diffusion theory code HFN⁵ and the transport theory code DTF-IV⁶ used these cross sections to determine the minimum critical size for spheres, as infinite cylinders and infinite slabs. Uniform lattice spacings or solution concentrations were assumed in the fuel region and full water reflection was placed outside the fuel region.

For uranium metal systems, the critical mass for uranium enriched in ^{235}U to 6 wt% or less is lower for a heterogeneous system than a homogeneous system; the critical volume also is smaller for the heterogeneous system. For enrichments above 6 wt%, however, the minimum critical mass for uranium in an aqueous solution will be less than the minimum achievable for a lattice of rods immersed in water. In this case the smallest critical mass is found to occur with rods of vanishingly small diameter, or of zero diameter. On the

other hand, although the minimum critical mass may be less for the homogeneous case (for enrichments above 6 wt%), the minimum critical volume will not be. It is then possible to achieve criticality in a smaller volume with a heterogeneous system, for example with fuel rods in water, but the critical mass will be greater than the minimum value for uranium in a homogeneous aqueous solution at the same uranium enrichment. Finally, at uranium enriched to greater than about 30 wt% ^{235}U the smallest volume will occur with a solid metal sphere at the theoretical density of 18.9 kg U/litre. The minimum critical spherical volume versus enrichment for full water-reflected solid metal, heterogeneous, and homogeneous systems is shown in Fig. 1.

Since solid metal is one extreme for both homogeneous and heterogeneous systems, at enrichments greater than the points at which the solid metal curves cross the homogeneous or heterogeneous curves the latter curves become only relative rather than absolute minima. The absolute minima occur for solid metal above these points. Adding water to solid metal at enrichments above ~30 wt% will initially increase the critical volume (to uranium densities of about 2 kg U/litre) and then the critical volume will decrease slightly before it increases again with further dilution. At the crossover point the well-moderated minimum equals the solid metal minimum and for enrichments slightly less than 30 wt% the relative metal minimum is higher than the absolute minimum for the moderated case. At even lower enrichments there will be no relative minimum for the metal.

Similar behavior occurs for the cylinder diameter and slab thickness shown in Figs. 2 and 3. The minimum critical parameters for uranium dioxide-water systems behave analogously except that the crossover point between theoretical density oxide and moderate oxide occurs at an enrichment greater than 50 wt% ^{235}U .

Fig. 1. Minimum critical volume as a function of ^{235}U enrichment in water-moderated and -reflected systems. (Symbols indicate calculated data points.)

Fig. 2. Minimum critical cylinder diameter as a function of ^{235}U enrichment in water-moderated and -reflected systems. (Symbols indicate calculated data points.)

Fig. 3. Minimum critical slab thickness as a function of ^{235}U enrichment in water-moderated and -reflected systems. (Symbols indicate calculated data points.)

These calculations provide a basis for setting criticality safety limits for uranium enriched to between 6 and 93 wt% ^{235}U . The data in Figs. 1 through 3, and similar curves to be presented for $\text{UO}_2\text{-H}_2\text{O}$ systems, will be discussed together with relevant validations of calculations.

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A k_{∞} Meter for Spent Nuclear Fuel Storage Pools, S. H. Levine, M. A. Schultz, D. Chang (Penn State Univ)

It is the objective of this program to develop a device to measure the k_{∞} of a spent-fuel assembly used in light-water

reactors. It is proposed that by measuring the k_{∞} of each fuel assembly before it is actually placed into a fuel storage rack, greater reliability is attained in knowing the k_{eff} of the fuel array in the fuel storage racks. The k_{∞} meter is designed to measure the k_{∞} of a fuel assembly prior to placing it into the storage rack.

Elaborate close-spaced fuel storage racks are being designed to permit close physical packing of the fuel assemblies from a nuclear power plant. The limiting factor is the fact that the used fuel assemblies still contain fissionable material capable of producing a supercritical condition if the spacing of the array is too close. Calculations can be made on the k_{eff} of the array to prevent criticality provided the maximum k_{eff} allowed for the array is equal to or less than 0.95. Using the k_{∞} meter to measure the k_{∞} of each fuel assembly prior to placing it in the storage rack provides greater confidence in the reactivity calculations of the subcritical array.

The k_{∞} meter is a subcritical assembly into which a fuel assembly is placed and its k_{∞} determined by the change it produces in the multiplication of the subcritical assembly. The subcritical assembly has a cross configuration as shown in Fig. 1 and the central position of the subcritical assembly is used for placing the fuel assembly of the unknown k_{∞} . Nuclear analyses of this configuration have been performed to permit establishing the multiplication response equations suitable for solution by a microcomputer. Neutron detectors, connected to the microcomputer, can provide data to the computer for analysis. The computer determines the k_{∞} of the central subassembly automatically and displays the result either on a scope, meter, printout, or all three or a combination thereof.

The basic equation used to relate the count rate, CR, to the parameters of the subcritical assembly is

$$\text{CR} = \frac{\alpha_p S_p + \alpha_0 S_0}{(1 - k_{\text{eff}})} \quad (1)$$

Fig. 1. Subcritical assembly for k_{∞} meter.

where

α_p = geometric factor which relates the neutron source strength in the peripheral fuel assemblies to the count rate

S_p = neutron source strength in the peripheral fuel assembly

α_0 = geometric factor which relates the source strength in the central fuel assembly to the count rate.

S_0 = neutron strength in the central fuel assembly.

Initial measurements must be made with the initial subcritical assembly to determine the k_{∞} and source strength of both the peripheral fuel assemblies and the central fuel assembly. The count rate, CR_0 , of the initial subcritical assembly is measured and the ratio of any subsequent measurement, CR_1/CR_0 , is related by

$$\frac{CR_1}{CR_0} = \frac{(1 - k_{eff,0}) (\alpha_p S_p + \alpha_0 S_0)}{(1 - k_{eff,1}) (\alpha_p S_p + \alpha_0 S_0)} \quad (2)$$

where the

0 subscript refers to the initial fuel assembly in the central position

1 subscript refers to the unknown fuel assembly in the central position.

Solving Eq. (2) for $k_{eff,1}$ yields

$$k_{eff,1} = 1 - (1 - k_{eff,0}) \left(\frac{CR_0}{CR_1} \right) \frac{(\alpha_p S_p + \alpha_0 S_0)}{(\alpha_p S_p + \alpha_0 S_0)} \quad (3)$$

Using a Taylor series expansion about $k_{eff,0}$ gives

$$k_{eff,1} = k_{eff,0} + \left(\frac{\partial k_{eff}}{\partial k_{\infty c}} \right)_0 (k_{\infty 1} - k_{\infty 0}) \quad (4)$$

where the $k_{\infty c}$ refers to the central fuel assembly only. $k_{\infty 1}$ and $k_{\infty 0}$ also refer to the central fuel assembly; however, the c is left off to simplify the notation. Solving for $k_{\infty 1}$ yields

$$k_{\infty 1} = k_{\infty 0} + (k_{eff,1} - k_{eff,0}) / \left(\frac{\partial k_{eff}}{\partial k_{\infty c}} \right)_0 \quad (5)$$

Nucleonic codes are used to establish the ratio (CR_1/CR_0) and k_{eff} as a function of a range of values for S_p , S_0 , $k_{\infty p}$, and $k_{\infty c}$. In addition, the α 's and $(\partial k_{eff}/\partial k_{\infty c})_0$ are determined as a function of $k_{\infty p}$ and $k_{\infty c}$. The subscripts p and c are used to identify the peripheral and central fuel assemblies, respectively. These functions are used to calculate k_{eff} from the CR_0/CR_1 ratio [using Eq. (3)] and then determine the unknown $k_{\infty 1}$, using Eq. (5). In practice, the α_p 's and α_0 's are functions of $k_{\infty c}$ and $k_{\infty p}$; hence, a first guess of the $k_{\infty 1}$ is made from the ratio CR_0/CR_1 . The program then iterates between Eq. (3) and Eq. (5) until the calculation converges.

Data have been used to check out the program by using a given input CR_0/CR_1 to calculate $k_{\infty 1}$. A wide range of CR_0/CR_1 values has been input into the computer program to test it. In all cases, the program converges to the correct $k_{\infty 1}$.

Measurement and Calculation of Reactor Parameters

A Benchmark Critical Experiment Program for 600- to 700-MW(e) LMFBRs, L. G. LeSage, M. J. Lineberry, H. F. McFarlane, S. G. Carpenter, P. I. Amundson (ANL-Idaho)

A series of fast reactor critical experiments at the zero-power plutonium reactor (ZPPR) is providing benchmark physics data for both conventional and heterogeneous LMFBRs in the 600- to 700-MW(e) size class. This data base is similar to one established by an earlier series related to fast reactors of 350-MW(e) size.¹ This paper discusses the scope of the program, outlines the measurements being emphasized, and presents some results from the conventional core experiments.

Briefly, the reactor configurations included in this program are:

ZPPR-9: A clean two-zone conventional core with cylindrical boundaries and no control rod positions (CRPs). Initial criticality was in May 1978.

ZPPR-10A: A two-zone conventional core with active core volume similar to ZPPR-9, but with a hexagonal core layout and 19 sodium-filled CRPs. In 10B, seven of the CRPs were filled with mockup control rods (CRs). Date of criticality for 10A was November 1978.

ZPPR-10C,10D: Again, two-zone cores with hexagonal layouts, but core volume expanded by 30%, simulating either a 900-MW(e) reactor (or a more conservative reactor of lesser rating). Nineteen CRPs were present in 10C, thirty-one in 10D.

ZPPR-11A,11B: Radially heterogeneous reactors prototypic of a Westinghouse reactor design.² They have a central blanket and three fuel rings separated by blanket zones. Twenty-four sodium-filled CRPs will be present. The core fuel volume fraction of 0.39 in 11A will be reduced to 0.32 in 11B. Criticality of 11A is expected in November 1979.

ZPPR-11C,11D: Radially heterogeneous reactors with cylindrical boundaries. These systems have central blankets, and two fuel rings separated by a blanket ring. The blanket ring thickness is 165 mm in 11C and will increase to 330 mm in 11D. The program is expected to be completed in February 1981.

Some parameters of interest for these reactor configurations are shown in Table I.

Of the conventional cores, only ZPPR-9 was constructed for basic physics data, and such parameters as power distribution, reaction rate ratios, sodium-void reactivities, material and Doppler reactivity worths, and control rod worths were

TABLE I
Key Reactor Parameters in the Benchmark Program

	Geometrical Layout	No. of CRPs/CRs	Core Volume (ℓ)	Core Height (m)	Fuel Volume Fraction
<u>Conventional Cores</u>					
ZPPR-9	cylindrical	none	4600	1.02	0.40
-10A	hexagonal	19/0	"	"	"
-10B	"	19/7	"	"	"
-10C	"	19/0	6000	"	"
-10D	"	31/0	"	"	"
<u>Radially-Heterogeneous Cores</u>					
ZPPR-11A	hexagonal	24/0	3780	0.92	0.39
-11B	"	"	"	"	0.32
-11C	cylindrical	none	4110	"	"
-11D	"	"	3910	"	"

TABLE II
Sensitivity Coefficients*

Core	k Sensitivities				Flux Shape Sensitivities ^a			
	²³⁸ U(n,γ)	²³⁸ U(n,f)	²³⁹ Pu(n,f)	radial D	²³⁸ U(n,γ)	²³⁸ U(n,f)	²³⁹ Pu(n,f)	radial D
ZPPR-2	-0.20	+0.07	+0.58	-0.09	-0.04	~ 0	~ 0	+0.03
-9	-0.25	+0.09	+0.52	-0.05	-0.17	-0.02	~ 0	+0.23
ZPPR-11C	-0.30	+0.07	+0.56	-0.09	-0.38	+0.06	-0.06	+0.45
-11D	-0.28	+0.06	+0.57	-0.12	-0.56	+0.08	-0.09	+0.66
NEACRP	-0.25	+0.08	+0.48	-0.03	-0.24	+0.02	+0.01	+0.23

*defined as $\frac{1}{\rho} \frac{\delta \rho}{\delta \sigma}$, where ρ is the parameter (k or flux shape), and σ is the varied quantity. All ZPPR-11 calculations were with one group; ZPPR-2, -9, and NEACRP results were from 12-group calculations.

^aFlux shape is the ratio of the average flux in the innermost fuel ring (or inner core) to the average flux in the outermost fuel ring (or outer core).

studied in detail. The four ZPPR-10 cores were more representative of power reactor designs, and power distributions, sodium-void effects, and control rod bank worths were emphasized.

The first two heterogeneous cores, ZPPR-11A and 11B, are to provide data for a representative power reactor design. Since optimum fuel pin size is still an issue in these types of reactors, the main difference between these two cores will be in the fuel volume fraction. Emphasis is to be placed on power distributions and spatial flux sensitivity to perturbations, control rod worths and interactions, and sodium-void and Doppler reactivities. The latter two configurations of this series, ZPPR-11C and 11D, will supply basic physics data on heterogeneous core behavior with two different degrees of neutronic coupling. One criteria for 11D was that it be as loosely coupled as the current industry consensus will permit.³ Separation of the fundamental and first harmonic eigenvalues is relatively small (2.7%), and the sensitivity of the flux shape to perturbation appears to be quite high.

Comparisons of experiments and calculations of the conventional cores with ENDF/B-IV data generally agreed with such comparisons for smaller cores, in spite of an increase in sensitivity of some parameters to nuclear data uncertainties. The calculated k_{eff} for ZPPR-9 of 0.985 was essentially the same, power distributions were predicted to within 2% in core zones, and sodium-void calculated-to-experimental ratios were similar to values observed before. Large control rod interactions were observed; up to 50% for six outer-ring rods compared with 20% in the smaller cores, but were calculated to within a few percent. Sensitivity analyses were made for 350-, 700-, and 1250-MW(e) core sizes. This has led to a quantitative understanding of the way in which bias factors and uncertainties obtained from

the ZPPR-9 and -10 experiments could be extrapolated to larger core sizes. An example of sensitivities of k and flux shape is shown in Table II, where ZPPR-2 represents the 350-MW(e) size, and the recent NEACRP benchmark reactor⁴ was used for the larger size.

When the configuration of the next-generation LMFBR is known better, results from this program of critical experiments will permit initial assessment of physics feasibility, and will guide planning for experiments on more prototypic configurations.

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Recriticality due to Clad-Free Pellet Rearrangement in a PWR, D. Cokinos, D. J. Diamond (BNL)

As a result of the recent incident at the Three Mile Island-2 (TMI) reactor there was concern that sufficient fuel rod cladding damage had occurred so that many of the fuel pellets were loose in the vessel. These pellets might arrange themselves in a configuration that would lead to recriticality unless sufficient soluble boron was in the system. The present calculations were undertaken to determine what boron concentration would be necessary to preclude recriticality under a number of assumed conditions.

The conditions for which results were obtained are generally conservative. It is assumed that a large fraction of the pellets in the core could come together in a minimum leakage (spherical) configuration. This would imply that there is large-scale cladding damage as well as damage to structural parts of the core. Other possible configurations were not considered.

To calculate the multiplication factor or eigenvalue (k) for the system, the HAMMER¹ multigroup, integral transport theory code was used. The code calculates the space-dependent spectrum in an infinite lattice. It then applies a leakage correction to the spectrum based on an input buckling. This input buckling is also used in the calculation of k for the system.

The pellets are assumed to be randomly oriented but uniformly spaced throughout the medium. Hence, the spectrum calculation is best represented by an array of spheres. Since this geometry is not available in HAMMER, the lattice calculation used one-dimensional (1-D) cylindrical geometry (i.e., cylinders of infinite length). In an attempt to obtain the correct resonance reaction rate, the surface-to-mass ratio (S/M) for the fuel cylinder in the HAMMER representation was made equal to that of an isolated pellet. Although this approach helps to account for the important effect of heterogeneity on resonance reaction rates, it does not properly account for other effects. Hence, parameters such as the disadvantage factor or Dancoff factor will have uncertainties associated with them.

Fig. 1. Multiplication factors as a function of U/W ratio for 1000, 2000, 3000 ppm for the 30, 50, and 100% configuration.

Calculations were done with a UO_2 -to- H_2O volume ratio (U/W) of 0.5-to-1.5. This range spans the optimum moderation for the pellets. The ^{235}U enrichment used was 2.6 wt%—consistent with the loading of fuel in the inner core regions. The temperature used was 250°F which was consistent with measured values at the time of these calculations. Three concentrations of soluble boron (1000, 2000, and 3000 ppm) have been considered. Leakage was taken into account by using a buckling corresponding to a sphere that includes 30, 50, or 100% of the UO_2 inventory.

Fig. 2. Minimum boron requirement as a function of U/W ratio.

Multiplication factors as a function of U/W ratio for the three concentrations are plotted in Fig. 1 for the 30 and 50% UO_2 configurations. The result of a single calculation at 3000 ppm for the entire UO_2 inventory and a U/W ratio of 1.25 is also shown. This figure also shows the radii R_{30} , R_{50} , and R_{100} at the corresponding U/W values, for the 30, 50, and 100% configurations, respectively.

From the variation of the multiplication factors with boron concentration for each U/W value, one can obtain the boron concentration which corresponds to a critical eigenvalue. Hence, one has a minimum boron requirement curve for each of the fractional fuel inventories considered; in essence, a criterion for subcriticality. Figure 2 shows these results and suggests that a minimum boron concentration of ~2720 ppm would ensure that the 30% configuration will be subcritical, while a concentration of ~2900 ppm would prevent a 50% configuration from reaching criticality. For the extreme case of the reassembling of the entire fuel inventory (100% of the UO_2) the minimum boron concentration is ~3060 ppm. These results suggest that it would be possible to keep a reconfigured core under extreme accident conditions in a subcritical state with reasonable boron concentrations.

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Thermal Reactor Benchmark Calculations with the MCNP Monte Carlo Code, Richard E. Prael (LASL)

A number of thermal reactor benchmark calculations¹ have been performed using the continuous-energy Monte Carlo code MCNP² with ENDF/B-IV data. The principal motivations for these calculations were:

1. to document the thermal scattering model and the treatment of $S(\alpha, \beta)$ scattering law data as implemented in the current production version of MCNP
2. to provide a reference point for continuing development work in the representation of bound molecular scattering for use in future versions of MCNP
3. to contribute to the current CSWEG thermal data testing efforts, using a thermal scattering method for $S(\alpha, \beta)$ independent of that used in comparable codes.

The model used for the calculations presented below is a coupled energy-angle scattering treatment using data derived from the ENDF/B $S(\alpha, \beta)$ scattering law data with the NJOY nuclear data processing system.³ The basic features of the model are as follows:

1. Over a grid of incident-neutron energies, a set of 32 equally probable discrete final energies is supplied for each incident energy.

2. For each incident energy and final energy, the probability distribution for the scattering angle is represented by four equally probable cosine bins, with boundaries chosen to preserve μ .

3. To obtain the results of a neutron scattering at a given energy, the final energy and angle distributions are sampled with the same random numbers for tabulated energies above and below the current neutron energy and interpolated linearly to obtain the outcome of the scattering event.

The above method is used for hydrogen scattering below 4 eV, with the available $S(\alpha, \beta)$ data extended as necessary with a short collision time approximation using an effective temperature of 1396.8 K. For these calculations, hydrogen scattering in the energy range from 10 to 4 eV and scattering by other light isotopes below 10 eV were represented by free gas scattering at 300 K. In addition, for these benchmark calculations, special data sets for ²³⁸U and ²⁴⁰Pu were prepared using NJOY³ from ENDF/B-IV with an extremely accurate resonance representation below 200 eV.

The results are shown in Table I for infinite lattice parameters for H₂O-moderated uranium lattices TRX-1 through -4 and H₂O-moderated uranium oxide lattices BAPL-UO₂-1 through -3. The parameters shown are as conventionally defined.⁴ Table II shows results for the benchmarks ORNL-1 through -4, spheres of uranyl nitrate. Table II also shows

TABLE I
Infinite Lattice Parameter Computations

Assembly	k (infinity)	ρ_{28}	δ_{25}	C*
TRX-1	1.1680 ± .0021	1.348 ± .008	.0972 ± .0005	.793 ± .004
	1.1696 ± .0009	1.344 ± .004	.0978 ± .0004	.793 ± .001
TRX-2	1.1533 ± .0024	.844 ± .006	.0605 ± .0004	.639 ± .003
	1.1586 ± .0015	.839 ± .003	.0596 ± .0003	.639 ± .001
TRX-3	1.0488 ± .0018	3.126 ± .017	.2379 ± .0013	1.265 ± .006
	1.0532 ± .0011	3.081 ± .007	.2357 ± .0006	1.257 ± .002
TRX-4	1.0186 ± .0024	.508 ± .005	.0353 ± .0003	.532 ± .003
	1.0156 ± .0016	.505 ± .003	.0351 ± .0002	.532 ± .001
BAPL-UO ₂ -1	1.1284 ± .0024	1.437 ± .008	.0835 ± .0005	.815 ± .004
BAPL-UO ₂ -2	1.1360 ± .0023	1.170 ± .006	.0675 ± .0004	.734 ± .003
BAPL-UO ₂ -3	1.1220 ± .0026	.924 ± .005	.0520 ± .0003	.657 ± .003

*BAPL calculations using RCP01 from Ref. 5.

TABLE II
Computations for Solutions

Assembly	k	ρ_{28}	δ_{25}	Leakage
ORNL-1	.9942 ± .0030	4.30 ± .08	.0215 ± .0001	.1720 ± .0012
	1.0007 ± .0025			.1709
ORNL-2	.9886 ± .0034	5.05 ± .09	.0254 ± .0001	.1716 ± .0012
	.9993 ± .0036			.1690
ORNL-3	.9913 ± .0030	5.80 ± .10	.0288 ± .0001	.1669 ± .0012
ORNL-4	.9937 ± .0030	6.16 ± .11	.0305 ± .0002	.1681 ± .0012
Assembly	k	ρ_{40}	²⁴⁰ Pu Capture	Leakage
PNL-2	1.0087 ± .0035	9.98 ± .14	.0480 ± .0006	.3826 ± .0015
	1.0109 ± .0030			.3849
PNL-3	.9919 ± .0033	1.98 ± .03	.0151 ± .0001	.3172 ± .0015
	.9961 ± .0030			.3163
PNL-5	1.0097 ± .0035	3.53 ± .05	.0216 ± .0002	.3632 ± .0015
	1.0088 ± .0033			.3689

*BNL calculations using RECAP from Ref. 4.

results for several benchmarks with plutonium nitrate solutions; in this table, δ_{40} is the ratio of epithermal-to-thermal capture in ^{240}Pu . The statistical uncertainties shown represent one standard deviation in the estimate; for computed ratios, the uncertainties are inferred approximately from the uncertainties in the numerator and denominator.

In conclusion, the infinite lattice calculations for the TRX lattices are in generally good agreement with comparable Monte Carlo calculations.^{4,5} The eigenvalue calculations for the solutions appear generally low compared to other computations,^{4,6} and computed leakages may be somewhat higher. The sensitivity of these calculations to various features of the scattering model is currently under investigation; preliminary results for ORNL-2 using a more detailed scattering representation indicate an eigenvalue of 1.0002 ± 0.0030 and a leakage of 0.1697 ± 0.0012 .

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Data and Analysis for Nuclear Criticality Safety

History, Facilities, and Programs of the Los Alamos Critical Assemblies Laboratory, Richard E. Malenfant (LASL)

Recently, the personnel of the Los Alamos Critical Assemblies Laboratory were recognized for thirty years of operation—1,900,000 man-hours—without a disabling injury. This achievement not only acknowledges the care implicit in the operation, but also the foresight that went into the design of the facility and the attitudes of the operating staff. It was the second fatality—Louis Slotin in 1947—that indicated the urgent need for a remotely controlled laboratory. It is also a pointed commentary on history that the Slotin accident occurred at the site of the present laboratory. We are one of the few facilities to be acknowledged in fact (e.g., "Day of Trinity," by Lansing Lamont and "Brighter Than a Thousand Suns," by Robert Jungk) and historical fiction (e.g., "The Accident," by Dexter Masters) as well as by an exhibit in the Smithsonian Institution (a model of Lady Godiva, the first bare sphere ^{235}U assembly, was on display for some years; the model is now in the Hall of Science at Flushing, New York). The present facilities of the laboratory and some current activities are described.

The Los Alamos Critical Experiments Facility consists of three remote-control laboratories, known as Kivas, which are located at some distance from the main laboratory building that houses individual control rooms, one for each Kiva.

Several general-purpose assembly machines occupy each Kiva. However, material, and even entire machines, may be moved from one Kiva to another. This is perhaps the only facility in which all fissile species (^{233}U , ^{235}U , ^{239}Pu) are routinely handled in a multitude of configurations involving all physical forms (solid, liquid, and gaseous). Machines have been run at or near critical with loadings from a few hundred grams to hundreds of kilograms, bare (as the 17-kg Jezebel) to massively reflected (as the 10 MT Big Ten), and at power levels from fractions of a milliwatt to near 100,000 MW (for a few μs).

KIVA I

Honeycomb

The honeycomb critical assembly is a universal split table machine containing a 1.83-m (6-ft)-cubical matrix of 76-mm (3 in.)-square aluminum tubes. It is designed to serve as a flexible system for initial mockup studies for basic critical parameter investigations. Fuel inventory consists of various fissile species such as 330 kg of 0.05-mm-thick $\text{U}(93)$ foils with widths and lengths appropriate to the aluminum matrix tubes. Control and safety rods utilize sections of the core or reflector materials for their function and major disassembly is provided by the movable section of the table. Honeycomb is presently stacked with a $\text{UO}_2\text{-Mo}$ (core), Be (reflector) mockup of a space power reactor.

Mars

The plasma core assembly (PCA) makes use of our Mars critical assembly machine, which is basically a heavy-duty stand with a hydraulic cylinder recessed in the floor to

provide primary assembly motion. PCA consists of a 1-m-long $\times$ 1-m-diam cavity surrounded on all surfaces by a $\frac{1}{2}$ -m-thick beryllium reflector. Recent experiments (funded by NASA) were directed at determining critical conditions for a UF_6 gas core.

KIVA II

Big Ten

Big Ten is a cylindrical assembly comprising a $^{235}\text{U}(10\%)$ core surrounded by a depleted uranium reflector. Overall, the cylinder is 0.84 m in diameter by 0.96 m long. The reflector is 0.15 m thick peripherally and 0.21 m thick at the ends. Major disassembly is provided by a movable section 0.39 m long; the stationary section is 0.57 m long.

The fissile core is composed of a central section of homogeneous $^{235}\text{U}(10\%)$ surrounded by a region of alternating 3-mm-thick $^{235}\text{U}(93\%)$ and 27-mm-thick natural uranium plates simulating 10% material. The total ^{235}U inventory in the core is 226 kg. Big Ten is presently being used for a series of replacement measurements of transuranic isotopes and for diagnostic irradiations.

Comet

The Comet machine is a general-purpose assembly machine that has been used for many critical determinations and nuclear safety studies. It is basically a support stand with means for bringing two parts of a configuration together. Secondary vernier motion on top of this table is accomplished by means of a screw-driven platform powered by a stepping motor, with accurate position readout in the control room. The forte in the use of Comet is that a flexible facility is available for addressing a variety of experiments on short notice. During 1979, Comet was employed in a series of measurements to evaluate ^{242}Pu cross sections and for various stackings of interest to criticality safety.

Flattop

Flattop is at the opposite extreme from our unreflected critical assemblies, inasmuch as it provides a thick natural uranium reflector around spherical cores of uranium (93.2% ^{235}U and 98.1% ^{233}U) and plutonium (94.9% ^{239}Pu). A complete set of data has been obtained for the neutronic characteristics of each of the three core materials. Critical masses are 6.0 kg for the ^{239}Pu (94.9%) ball, 5.7 kg for the ^{233}U (98.1%), and 17.8 kg for the ^{235}U (93.2%) ball after correction to a completely spherical geometry. Flattop is used for replacement measurements and service irradiations.

Jezebel

Major features of Jezebel, a plutonium sphere clad with 0.13-mm-thick nickel, include a stationary central section, upper and lower movable blocks, and a single plutonium control rod. The reactivity of Jezebel may be adjusted in increments by means of plutonium filler buttons that fit into cavities machined into the top surfaces of the lower block and the central section. Jezebel is mounted on a portable aluminum stand and extension cables allow operation even outside the building. In fact, measurements have

been made with Jezebel supported 10 m from the ground to avoid neutron room-return effects. Jezebel is presently being recast and reassembled for precise replacement measurements.

KIVA III

Godiva IV

Godiva IV is the latest in the line of unreflected enriched uranium critical assemblies. The original Godiva (Lady Godiva), spherical in shape, was used to establish an accurate value for the critical mass of a ^{235}U unreflected sphere. The reactivity self-quenching features of metal critical assemblies were also established with Godiva I.

Godiva IV was designed to give maximum pulse yield and made use of a U-Mo(1.5%) alloy for increased tensile strength and resistance to deformation or cracking under the operating thermal shock conditions. Typical bursts of 4×10^{16} fissions with a 50- μs peak width at half-maximum produce a temperature rise of 200°C.

During 1979, Godiva IV was used in a series of exposures to evaluate nuclear excitation of lasers.

SKUA

SKUA, a burst-mode critical assembly under construction, will be an approximate unit-height-to-diameter (0.318 m) cylinder of annular U (93%) fuel rings, internally moderated by $\text{ZrH}_{1.6}$ and externally reflected by Cu. When complete, SKUA will be able to vaporize a ^{235}U foil in the flux trap during a burst.

PARKA

PARKA is a Kiwi-type reactor with a core made up of hexagonal, one-hole, uranium-graphite fuel elements. The core is reflected by 50 mm of graphite and 114 mm of beryllium; the ends are essentially unreflected. Control is by 12 rotating Be drums, each with a ^{10}B -Al vane covering 129 deg of the surface. The control drums are rotated by stepping motors whose angular position is read out precisely in the control room. Measurements to evaluate fuel motion diagnostics were made in Parka during 1979.

The Critical Assemblies Laboratory provides a unique capability for LASL and other DOE contractors—no comparable facility exists in the country. Material, assembly machines, remotely controlled laboratories, and expertise are readily available here—and nowhere else—to provide immediate support for nuclear data and design, certification, and development for civilian and military applications.

The Sham Ram-Scram: A Positive Reactivity Change on Disassembly, E. R. Martin, G. A. Jarvis, R. E. Malenfant, E. A. Plassmann (LASL)

In the course of the confined UF_6 plasma core reactor experiment,¹ an interesting criticality safety question arose. The resolution of the question clearly indicated a possible pitfall in the control and safety systems of ram-operated critical assemblies.

The reactor configuration, shown in Fig. 1, consists of a large outer cylindrical reflector shell of beryllium, into which the fuel (in the form of a cylindrical shell of uranium carbide fuel elements) and an inner beryllium reflector are assembled by a ram which lifts them into the reflector. Originally, the reactor control and safety systems consisted of rotary drums with a 120-deg boron sector running the full length of the core, which resided permanently in the outer beryllium reflector and were spring loaded for fail-safe control.

Fig. 1. Reactor configuration.

To obtain adequate reactivity control of the reactor for the gaseous UF_6 experiment, these rotary control elements were replaced by linear boron vanes running vertically, which could be inserted into the beryllium reflector from the top and acted as a reactivity poison on insertion. Eight such vanes were provided; four could be individually controlled for reactivity measurements and four ganged together and were controlled by an air-driven solenoid to act as safeties in case of a scram shutdown. Each of these vanes was worth about 2\$,² so the total shutdown from the four safeties alone was nearly 8\$. Since our operating plan demands two independent safety scram mechanisms,² the ram drop was also used as an additional scram independent of the boron safety vanes.

This operating procedure was duly approved and we had operated the system for many hours calibrating control vane and fuel worths. Before actual operation with the gaseous UF_6 in the reactor core, we decided to measure the two scram shutdown mechanisms independently to resolve an internal discussion as to the worth of the ram dropping. Some of us felt the ram would provide at least 50¢/s shutdown, others said it would be much less than this, perhaps as little as 10¢/s, while a few argued it might even go positive. It was important to resolve this question since the experiment is conducted with the control vanes on automatic control, recording the reactivity change as UF_6 is introduced into the system. On automatic control, the vanes could change reactivity at 5¢/s, so we needed to know that the ram drop could override this much reactivity change in case of a scram shutdown where the safety vanes failed.

We had not previously tested the scram mechanisms separately, but the logic circuitry lent itself well to simply removing the safety vanes from the scram shutdown chain. We then measured the reactivity change as a function of time by computer scaling of the ionization chamber outputs during a scram without the vane safeties. The data obtained are typically shown in Fig. 2. It is clear that with certain fuel

²The dollar of reactivity is defined as the increment between delayed critical and prompt critical. It is subdivided into 100¢.

Fig. 2. Ram drop power history.

loadings (the worst case is with an intermediate fuel loading, since this causes the control vanes to be halfway in), the reactivity actually increased for the first 3 s after scram. By repeating the test with the automatic controller on, and knowing that this would compensate for 5¢/s reactivity change, we were able to estimate the total increase at 36¢; an unfortunate situation for a scram mechanism!

In retrospect, we can easily understand this reactivity history. The initial ram dropping causes fuel to be removed from the boron control vane shielding and placed into a more reactive configuration faster than removal of the fuel assembly from the reflective upper beryllium plug decreases reactivity. It should also be emphasized that with the safety

vanes operational, their insertion of 8\$ negative reactivity within the first half-second of scram makes the system quite safe in any case, but this example of increasing reactivity with ram withdrawal does illustrate the need to examine reactivity conditions carefully when any change is made in the control system of a reactor.

Subsequently, we removed the ram drop as a scram mechanism and made each of the safety vanes independent with its own air supply. Ram drop is initiated only following full insertion of at least one safety rod.

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Using Soluble Neutron Absorbers for Criticality Safety, R. C. Lloyd, S. R. Bierman, E. D. Clayton, B. M. Durst (BNWL)

Experimental criticality data are needed for soluble neutron absorbers (poisons) to determine their effectiveness in the criticality prevention and control of solutions bearing fissile material. These data are needed to validate criticality control calculational techniques that are used in the analysis of poisoned systems. This paper summarizes the results of recent experiments performed at the Battelle-Pacific Northwest Critical Mass Laboratory on uranium rod lattices in uranyl nitrate solution containing gadolinium. (The use of soluble neutron absorbers can be a useful safety measure in the prevention of nuclear criticality in the processing and storage of nuclear material.) Other experiments, not discussed in this paper, have been reported for plutonium

FUEL TUBE CANNED LENGTH*	(34.421 in.)	874.29 mm
FUEL ROD CORE LENGTH	(32.394 in.)	822.81 mm
FUEL PELLET LENGTH	(0.625 in.)	15.88 mm
CLAD - OD	(0.567 in.)	14.40 mm
CLAD THICKNESS (304L SS)	(0.032 in.)	0.813 mm
FUEL COLUMN DIAMETER	(0.498 in.)	12.649 mm
UO ₂ DENSITY (% THEORETICAL)	94.9 ± 0.6	
ENRICHMENT	4.29 w/o ²³⁵ U	

* MEASURED FROM BOTTOM OF END SPADE TO BOTTOM OF TOP SPADE

Fig. 1. Description of 4.3 wt% ²³⁵U enriched fuel rods.

nitrate solution containing Gd, assemblies of mixed-oxide rods in plutonium-uranium nitrate solutions containing Gd, plutonium solutions in Raschig ring filled vessels, and lattice assemblies of low enriched uranium rods in water containing Gd, Cd, and B as soluble absorbers.¹⁻⁴

The experiments were oriented toward providing data on the effectiveness of Gd as a soluble absorber for use in nuclear criticality prevention. Data on integral critical experiments are presented against which calculational techniques can be checked.

Three series of criticality experiments were made with fuel rods in uranyl nitrate solution with Gd. The fuel rods were made up of UO₂ pellets (4.3 wt% ²³⁵U enrichment) contained in stainless-steel tubes with a cladding o.d. of 14.4 mm (cladding thickness of 0.813 mm). A detailed description of the fuel rods is shown in Fig. 1. The measurements were made in three stainless-steel (304L) cylindrical vessels with inside diameters of 557.5, 658.0, and 761.6 mm, containing lattice templates having center-to-center fuel rod spacings of 22.9, 27.9, and 33.0 mm, respectively. These vessels were reflected with water on the sides and bottom. A

detailed schematic diagram of the experimental assembly is shown in Fig. 2.

For each case above, the vessel was filled with a fixed number of fuel rods. The uranyl nitrate solution was then remotely pumped into the vessel. The critical height of the uranyl nitrate solution was then determined for three different concentrations of gadolinium.

The assembly with the 22.9-mm lattice spacing contained 451 fuel rods in a vessel having 557.5-mm i.d. The critical height was determined to be 373 mm when the uranyl nitrate did not contain gadolinium and 752 mm when the solution contained 0.184 g/litre gadolinium. The assembly with the 27.9-mm lattice spacing contained 433 fuel rods in a vessel having a 658.0-mm i.d. The critical height of the uranyl nitrate solution without gadolinium was 309 mm. Added gadolinium of 0.124 g/litre increased the critical height to 565 mm. The assembly with the 33.0-mm lattice spacing had an i.d. of 761.6 mm and contained 421 fuel rods. The system was determined to be critical with a uranyl nitrate height of 333 mm without gadolinium and a height of 577 mm when the solution contained 0.074 g Gd/litre.

TABLE I
Effect of Soluble Gadolinium in Uranyl Nitrate Solution on
Criticality of Lattice Assemblies

NUMBER OF FUEL RODS	TRIANGULAR LATTICE SPACING (mm)	GADOLINIUM CONCENTRATION (g/litre)	CRITICAL SOLUTION HEIGHT (mm)	CALCULATED k_{eff}^a
451	22.86	0	373	1.002 ± 0.009
		0.124	546	0.992 ± 0.008
		0.148	612	0.990 ± 0.009
		0.184	752	0.987 ± 0.009
433	27.94	0	309	1.001 ± 0.009
		0.074	413	0.995 ± 0.009
		0.099	481	1.010 ± 0.006
		0.124	565	0.981 ± 0.006
421	33.00	0	333	1.010 ± 0.007
		0.025	388	1.017 ± 0.006
		0.049	460	1.006 ± 0.008
		0.074	577	1.007 ± 0.007
				Av. = 1.000
				RANGE OF 0.981 TO 1.017
URANYL NITRATE COMPOSITION			ISOTOPIC COMPOSITION OF URANIUM	
U. CONCENTRATION	195.8 g/litre		238	95.65 wt%
Sp. GRAVITY	1.254		236	0.022
FREE ACID MOLARITY	1.52		235	4.306
			234	0.022

^aCALCULATED USING KENO IV. EIGHTEEN MULTIGROUP CROSS SECTIONS WERE GENERATED USING EGGNIT (17 EPITHERMAL, 1 THERMAL) WITH ENDF/B-IV DATA INPUT. THE THERMAL GROUP CROSS SECTIONS WERE GENERATED BY THE TEMPEST MODULE IN EGGNIT.

Fig. 2.

The effect of gadolinium additions in the three systems can be seen in Table I, where the critical heights are given for each gadolinium concentration. Table I also includes the chemical analysis data and the calculated k_{eff} values which correlate well with experiment.

As a result of these experiments, critical data now exist for low-enriched uranium rod systems immersed in uranyl nitrate solutions containing gadolinium. These data can be used to benchmark commonly used calculational methods.

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Criticality of Water-Moderated and -Reflected Plutonium Arrays, B. L. Koponen (LLL)

At Lawrence Livermore Laboratory (LLL) we traditionally establish fissile material mass limits in two categories: solid and solution/dispersible. A question has arisen,

however, as to the unit size below which solid units must be regarded as dispersible; i.e., interunit moderation must be considered in the criticality controls. When answering this question for plutonium metal units, we had a shortage of experimental data and had to use validated computational methods. In this paper we present the results of critical mass calculations for plutonium units that enable us to determine in which of the two categories units of a particular size belong.

The LLL version of the MORSE¹ neutron Monte Carlo code was used to generate plutonium metal-array-criticality data. We used a 92-group neutron cross-section set derived from Howerton's compilation.² Thermal data for water were derived by the FLANGE code.³ Validation of the computational methods was done by calculating (a) a single water-reflected metal sphere,⁴ (b) the limiting critical concentration of plutonium in water,⁵ (c) the arrays of 3- and 6-kg plutonium cylinders,⁶ and (d) the lattices of Pu-Al rods in water.⁷

Parameters that were varied in the calculations include the (a) number of units, (b) unit mass, (c) unit spacing, (d) unit density, (e) unit shape, and (f) water moderator density. The arrays were cubic and were reflected by a 15-cm layer of water.

A useful display of the data is a plot of the critical, average plutonium density (plutonium unit mass divided by array cell volume) vs the mass of the plutonium unit. In Fig. 1 is an example of a 5^3 array of full-density metal spheres, both with and without a full-density water moderator. Data for moderated infinite arrays are also shown. The family of slanted lines, representing unit spacings, assist in determining the effects of various parameters on array criticality. Limits appear on Fig. 1 at 10.34 g Pu/cm³, corresponding to units in contact (higher densities are possible only with nonspherical units), and at the critical masses for single isolated units (5.5 kg reflected and 10 kg unreflected). Between these limits are three unique regions for the effect of water flooding; these are labeled A, B, and C in Fig. 1. In region A, water flooding causes a decrease in critical mass for a given unit spacing. In region B, the effect of water flooding is opposite to that of region A. In region C, water flooding causes a decrease in critical unit mass, simply because the critical unit mass of a dry array exceeds that of a single water-reflected unit.

Fig. 1. Criticality of arrays of plutonium metal spheres. The family of slanted lines represents selected unit spacing.

Fig. 2. Effect of water-moderator density on criticality of water-moderated and -reflected 10^3 arrays of plutonium metal spheres. The mass calculated for critical isolated units is 5.5 kg with water reflection and 10 kg without reflection.

Results of varying the other parameters mentioned above are also presented. For example, in Fig. 2 we see the results of critical mass calculations for a 10^3 array with several densities of a uniformly distributed water moderator. The shapes of the curves are similar for the various moderator densities although the crossover points between the regions and the moderator effectiveness vary with moderator density. Another varied parameter was the length-to-diameter ratio of a cylinder. A full density water-moderated and -reflected 10^3 array of plutonium metal cylinders, $D/L = 4$, has a critical mass about 30% less than a comparably moderated and -reflected array of spheres with masses ranging from 10 to 300 g.

One outgrowth of this, which we found useful in our solid or solution/dispersible problem, is that arrays of compact plutonium units, dry or fully immersed in water, are less reactive than an array of the same number of units in contact if the mass of each unit is 10 g or more. Therefore, mass limits set for solid units can be applied to assemblages of smaller metal units if each exceeds 10 g, regardless of the degree of assemblage moderation.

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Criticality Analyses for Safety Evaluation of Dry Storage of Spent-Fuel Assemblies, M. H. Lipner, J. M. Ravets (W-AES)

This paper presents the results of the nuclear analyses utilized in the criticality safety evaluation for the Spent-Fuel Handling and Packaging Program Demonstration at the Nevada Test Site.¹ This demonstration is being conducted at the Engine Maintenance Assembly and Disassembly (E-MAD) Facility in Area 25, which is operated by the Westinghouse Advanced Energy Systems Division for the Nevada Operations Office of the Department of Energy (DOE). The configurations that were considered are associated with encapsulating the spent-fuel assemblies and subsequently storing them in surface and near-surface dry storage configurations.

The lag storage facility pit shown in Fig. 1 is utilized to store unsealed or seal-welded canisters containing spent-fuel assemblies prior to seal welding or during the interval between seal welding and final storage emplacement, respectively. The pit is below the E-MAD hot bay floor and has a storage capacity of 24 canisters arranged in three separate 2 X 4 arrays.

The fuel assembly used in the numerical analysis is an unirradiated, standard 17 X 17 lattice array, Westinghouse PWR fuel assembly containing 0.4615 MTU (3.5 wt% ^{235}U). Although only spent-fuel assemblies are handled at E-MAD, a conservative analytical approach was taken by performing calculations of the effective multiplication factor (k_{eff}) and evaluating all postulated accident conditions assuming unirradiated fuel. If a given configuration of fuel assemblies is shown to be subcritical for unirradiated fuel, then the same configuration of spent-fuel assemblies would be even more subcritical.

The nuclear criticality analyses employed the current version of the AMPX modular code system,² which includes a 218-group cross-section library,³ and the KENO-IV Monte Carlo criticality program,⁴ all developed by the Oak Ridge National Laboratory. The cross-section set and codes were thoroughly qualified by the Westinghouse Nuclear Fuel Division by comparing predicted results with critical experiments.

Table I presents a summary of the results of the KENO-IV Monte Carlo nuclear criticality calculations for the facilities and accidents considered. It should be noted that the KENO-IV results given in Table I must be adjusted upward by adding the method uncertainty (U) (Refs. 5, 6). U is obtained using the formula $U = [(K_1 S_1)^2 + (K_2 S_2)^2]^{1/2}$, where K_1 is the one-sided 95-95 tolerance limit factor of the benchmark criticality calculations, S_1 is the standard deviation of the benchmark calculations, K_2 (~ 1.7) is the one-sided 95-95 tolerance limit factor for the individual KENO calculations performed in this study, and S_2 is the standard deviation of the individual KENO calculation. U is thus composed of two parts. The first ($K_1 S_1$) is due to the spread in the individual values of k_{eff} calculated for the benchmark criticals. The second ($K_2 S_2$) is due to the statistics of the specific KENO calculation, which results from the fact that less than an infinite sample of neutron histories is calculated. There is a 95% probability with 95% confidence that the true k -effective will be less than or equal to the adjusted value given in Table I.

Fig. 1.

In addition to the adjustment of the KENO-IV results by the method uncertainty, the calculated k_{eff} for the lag storage pit must also be adjusted upward approximately 0.03 Δk to account for the fact that the water-filled spaces in the fuel assembly, which result from the removal of instrumentation and control rods, were not represented explicitly in the calculational model. The 0.03 Δk correction factor is based on independent KENO-IV calculations performed by the Westinghouse Nuclear Fuel Division.

The k -effectives were calculated to be less than unity for the various postulated accident conditions. Using these

calculated results, as well as other published data, the criticality safety evaluation showed that no single credible accident could result in criticality during either fuel handling or fuel storage operations. Similarly, there are no double accidents having any reasonable probability of occurrence which would lead to a critical configuration.

1. "Safety Assessment Document for the Spent Fuel Handling and Packaging Program Demonstration at the Nevada Test Site," NVO-198, U.S. Department of Energy (Dec. 1978).

TABLE I
Calculated k-Effectives of Configurations Associated with the
Storage of Unirradiated PWR Fuel Assemblies

Configuration	Accident	KENO-IV k-Effective	Adjusted k-Effective
Lag Storage Pit	Canisters and Pit Flooded ^a	0.8506 ± 0.0047 ^b	0.90
	Canisters Flooded and Pit Dry	0.8704 ± 0.0042	0.92
Physical Rearrangements	Slumped ^c Fuel in a Flooded Canister, 127-cm-thick Concrete Reflector	0.9425 ± 0.0044	0.96
	Slumped Fuel in a Flooded Canister, 51-cm-thick Water Reflector	0.9110 ± 0.0048	0.93
	Slumped Fuel in a Flooded Canister, No Reflector	0.8233 ± 0.0044	0.84
	Slumped Fuel in a Flooded Canister, Bubble Formation, ^d 51-cm-thick Water Reflector	0.8619 ± 0.0048	0.88
	Homogeneous Fuel-Water Mixture in a Flooded Canister, 51-cm-thick Water Reflector	0.8593 ± 0.0045	0.87

^aThis is equivalent to a single, isolated, flooded and reflected canistered fuel assembly.

^bS₂, the standard deviation of the individual KENO calculation.

^cComplete cladding failure leading to agglomeration of fuel pellets in the bottom of the canister is referred to as a fuel slump. The assumed volume fraction of fuel pellets in the slumped fuel region was 0.532 based on experimental measurements using standard Westinghouse fuel pellets

^dAssumed 90% of normal water density.

2. N. M. GREENE et al., "AMPX—A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B." ORNL/TM-3706 (Mar. 1976).
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Radius of Coverage of Nuclear Criticality Detectors, E. L. Maceda (B&W), D. K. Sgarlata (B&W-Apollo)

This paper describes a method to calculate the radius of coverage of criticality alarms when there are concrete walls that must be accounted for. CFR.70.24 (a) (1) states that "the monitoring system shall be capable of detecting a criticality that produces an absorbed dose in soft tissue of

20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute. Coverage of all areas shall be provided by two detectors." The calculations were made to avoid mounting redundant detectors to afford double coverage. At Babcock & Wilcox's plutonium facility at Apollo, Pennsylvania, there are high- and low-density concrete walls between different areas in the plant. Criticality alarm coverage must take these walls into account to provide adequate protection.

The radius of coverage in air of a criticality detector can be found in the American National Standard for Criticality Accident Alarm System.¹ It can be shown that the dose rate at a detector due to a criticality is proportional to

$$\sum_{i=1}^f \frac{D_i B_i}{R^2 \exp(\alpha_i \rho_c X_1 + \beta_i \rho_a X_2)} \quad (1)$$

where the energy spectrum² of fission gamma rays has been divided into seven groups and

i = energy group index

D_i = dose rate for group i

B_i = dose buildup factor due to intervening concrete for group i

R = distance between source and detector

α_i = mass attenuation coefficient in concrete for group i

ρ_c = concrete density

X_j = concrete thickness

β_i = mass attenuation coefficient in air of group i

ρ_a = density of air (0.001293 g/cm³)

$X_2 = R - X_1$

The dose rates were obtained from data in Ref. 3 and the buildup factors were obtained from Ref. 4 where aluminum was used in place of concrete for buildup since the effects are approximately equivalent. Mass attenuation coefficients for concrete and air were obtained from Ref. 5.

To calculate the reduction in the radius of coverage due to concrete at a given detector trip point, Eq. (1) should be evaluated, where X_1 equals zero and $R - R_D$, the radius of coverage for a steady-state excursion at the detector trip point. R_D can be obtained from Appendix B of the proposed standard on criticality alarm systems.¹ The result of these substitutions gives

$$AD = \sum_{i=1}^f \frac{D_i B_i}{R_D^2 \exp(\beta_i \rho_a R_D)} \quad (2)$$

Then to determine the effect of concrete on the radius of coverage, Eqs. (1) and (2) can be combined to give

$$\sum_{i=1}^f \frac{D_i B_i}{R^2 \exp(\alpha_i \rho_c X_1 + \beta_i \rho_a X_2)} \geq AD \quad (3)$$

This equation means the dose at the detector must be greater than or equal to the dose the detector sees at the radius of coverage in air given by the standard. At Apollo,

Fig. 1. Distance of air vs wall thickness detector trip of 50 mR/h.

both hollow and solid concrete block are used; the densities used in these calculations for each type of block are 2.64 g/cm³ for solid and 1.88 g/cm³ for hollow block. These densities are at least 20% greater than those used in the plant and therefore give conservative results. Equation (3) was solved with a computer program for various wall thicknesses of solid and hollow concrete block at several detector trip points.

Figure 1 is a plot of these results for a detector trip point of 50 mR/h. The area between the two curves represents the coverage allowed by a combination of solid and hollow block walls. As an example for using this graph to determine the radius of coverage when there is both solid and hollow walls, assume there is 10 in. of hollow and 10 in. of solid block present. From Fig. 1, 10 in. of solid block allows a radius of coverage of 300 ft. Proceed horizontally over to the hollow block curve where this is equivalent to 14 in. of hollow block. Add 10 in. to this and find the radius of coverage at 24 in. of hollow block, which is 185 ft. Using this method, the radius of coverage for any combination of hollow and solid concrete walls can be determined, and thus ensure non-redundant placement of detectors to ensure double coverage of criticality alarms within a plant.

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5. *Radiological Health Handbook*, U.S. Dept. of HEW (1970).

CSRL-V: An ENDF/B-V 227-Group Cross-Section Library for Criticality Safety Studies, W. E. Ford, III, R. M. Westfall, B. R. Diggs, C. C. Webster, J. R. Knight (UCC-ND)

The AMPX system^{1,a} was used to generate a P₃ 227-neutron-group master cross-section library^b containing data for all materials in the ENDF/B-V general-purpose file.² (See Table I for a list of the master data sets, the ENDF/B-V materials processed, and the data sets that contain resonance parameters.) The library, designated CSRL-V, is a data base for the subsequent generation of problem-dependent fine- and/or broad-group cross sections for shipping cask calculations and other criticality safety analyses. The problem-dependent data can be used with codes such as KENO IV, ANISN, XSDRNPM, VENTURE, DOT, MORSE, etc. CSRL-V data can be coupled with photon-production and photon-interaction multigroup data produced with the

^aThe AMPX system is distributed by the Radiation Shielding Information Center (RSIC) in a code package identified as PSR-63/AMPX-II.

^bAn AMPX master cross-section library is the most comprehensive processed library in the AMPX system. The library contains multigroup cross sections, transfer matrices, resonance parameters, weighting functions, etc., has provisions for neutron, photon-production, and photon-interaction data, has provisions for temperature dependence on thermal scattering kernels, etc.

TABLE I
CSRL-V 227-Neutron-Group Master Cross-Section Library

Nuclide	ENDF/B-V MAT No.	Master Library ID No.	Resolved Resonance Data	Weight Function ^a	Nuclide	ENDF/B-V MAT No.	Master Library ID No.	Resolved Resonance Data	Weight Function ^a
H-1	1301	1001	No	A	Xe-132	1352	54132	Yes	B
H-2	1302	1002	No	A	Xe-134	1354	54134	Yes	B
H-3	1169	1003	No	A	Xe-135	1294	54135	No	B
He-3	1146	2003	No	A	Xe-136	1356	54136	No	B
He-4	1270	2004	No	A	Cs-133	1355	55133	Yes	B
Li-6	1303	3006	No	A	Ba-138	1353	56138	No	B
Li-7	1272	3007	No	A	Sm-149	1319	62149	Yes	B
Be-9	1304	4009	No	A	Eu-151	1357	63151	Yes	B
B-10	1305	5010	No	A	Eu-152	1292	63151	Yes	B
B-11	1160	5011	No	A	Eu-153	1359	63151	Yes	B
C-12	1306	6012	No	A	Eu-154	1293	63151	Yes	B
N-14	1275	7014	No	A	Gd-152	1362	64152	Yes	B
N-15	1307	7015	No	A	Gd-154	1364	64154	Yes	B
O-16	1276	8016	No	A	Gd-155	1365	64155	Yes	B
O-17	1317	8017	No	A	Gd-156	1366	64156	Yes	B
F-19	1309	9019	No	A	Gd-157	1367	64157	Yes	B
Na-23	1311	11023	Yes	B	Gd-158	1368	64158	Yes	B
Mg	1312	12000	No	A	Gd-160	1370	64160	Yes	B
Al-27	1313	13027	No	A	Dy-164	1031	66164	Yes	B
Si	1314	14000	No	A	Lu-175	1032	71175	Yes	B
P-31	1315	15031	No	A	Lu-176	1033	71176	Yes	B
S-32	1316	16032	No	A	Hf	1372	72000	Yes	B
Cl	1149	17000	No	A	Hf-174	1374	72174	Yes	B
Ca	1320	20000	No	A	Hf-176	1376	72176	Yes	B
Ti	1322	22000	No	A	Hf-177	1377	72177	Yes	B
V	1323	23000	No	B	Hf-178	1378	72178	Yes	B
Cr	1324	24000	Yes	B	Hf-179	1383	72179	Yes	B
Mn-55	1325	25055	Yes	B	Hf-180	1384	72180	Yes	B
Fe	1326	26000	Yes	B	Ta-181	1285	73181	Yes	B
Co-59	1327	27059	Yes	B	Ta-182	1127	73182	Yes	B
Ni	1328	28000	Yes	B	W-182	1128	74182	Yes	B
Cu	1329	29000	Yes	B	W-183	1129	74183	Yes	B
Kr-78	1330	36078	Yes	B	W-184	1130	74184	Yes	B
Kr-80	1331	36080	Yes	B	W-186	1131	74186	Yes	B
Kr-82	1332	36082	Yes	B	Re-185	1083	75185	Yes	B
Kr-83	1333	36083	Yes	B	Re-187	1084	75187	Yes	B
Kr-84	1334	36084	Yes	B	Au-197	1379	79197	No	B
Kr-86	1336	36086	Yes	B	Pb	1382	82000	No	A
Zr	1340	40000	Yes	B	Th-232	1390	90232	Yes	C
Zr-90	1385	40090	Yes	B	Pa-233	1391	91233	Yes	C
Zr-91	1386	40091	Yes	B	U-233	1393	91233	No	C
Zr-92	1387	40092	Yes	B	U-234	1394	92234	Yes	C
Zr-94	1388	40094	Yes	B	U-235	1395	92235	Yes	C
Zr-96	1389	40096	Yes	B	U-236	1396	92236	Yes	C
Nb-99	1189	41099	Yes	B	U-238	1398	92238	Yes	C
Mo	1321	42000	Yes	B	Np-237	1337	93237	Yes	C
Tc-99	1308	43099	Yes	B	Pu-238	1338	94238	Yes	C
Rh-103	1310	45103	Yes	B	Pu-239	1399	94239	Yes	C
Ag-107	1371	47107	Yes	B	Pu-240	1380	94240	Yes	C
Ag-109	1373	47109	Yes	B	Pu-241	1381	94241	Yes	C
Cd	1281	48000	No	A	Pu-242	1342	94242	Yes	C
Cd-113	1318	48113	Yes	B	Am-241	1361	95241	Yes	C
Xe-124	1335	54124	Yes	B	Am-242	1369	95242	Yes	C
Xe-126	1339	54126	Yes	B	Am-243	1363	95243	Yes	C
Xe-128	1348	54128	Yes	B	Cm-243	1343	96243	Yes	C
Xe-129	1349	54129	Yes	B	Cm-244	1344	96244	Yes	C
Xe-130	1350	54130	Yes	B	Cm-245	1345	96245	Yes	C
Xe-131	1351	54131	Yes	B	Cm-246	1346	96246	Yes	C

^aThe following code is used to identify the weight functions:

Code	Weight Function
A	10^{-5} -0.1265 eV Maxwellian; 0.1265 eV-1.4 MeV 1/E; 1.4-20 MeV fission.
B	10^{-5} -0.1265 eV Maxwellian; 0.1265 eV-0.75 MeV 1/E; 0.75-20 MeV fission.
C	10^{-5} -0.1265 eV Maxwellian; 0.1265 eV-0.1 MeV 1/E; 0.1-20 MeV fission.

AMPX system to produce coupled neutron-gamma cross-section libraries.

Experiences gained with the ENDF/B-IV 218-group master cross-section library for criticality safety studies^{3,4,c} influenced the selection of the group structure and the point-to-multigroup weighting functions used to generate the subject library. When selecting the energy structure for the 3-eV to 20-MeV energy range, consideration was given to the resonance structure of prominent nuclei, the thresholds of important reactions, and various fission spectra. Additional groups were added in the 6- to 20-MeV range to facilitate the calculation of deep penetration in concrete. For $10^{-5} \leq E_n \leq 3$ eV, 79 closely spaced thermal groups were chosen to examine the effects of low-energy resonance and thermal-neutron upscatter. Two groups were added in the 10^{-5} to 10^{-2} eV range to "better treat" capture by gadolinium. The 218-group CSRL, the 27-group SCALE, and the 16-group Hansen-Roach energy structures are subsets of the 227-group structure. As identified in Table I, various weighting spectra were used to prepare the multigroup cross sections for the actinides, the resonance structural materials, and the nonresonance materials.

Data in the CSRL-V library were checked for first-order consistencies with the RADE module of the AMPX system.

^cThe ENDF/B-IV 218-group library (CSRL) is distributed by RSIC in a code package identified as DLC-43.

The validity of the selected data sets from the library was tested in performance parameter calculations for a series of benchmark critical experiments. Where available, results using CSRL-V data were compared with results using ENDF/B-IV processed data.

The CSRL-V library is available on magnetic tape from RSIC. An AMPX-generated pointwise cross-section library was produced as an intermediate step in the preparation of the CSRL-V library. The pointwise library, containing energy-cross-section pairs for the fission, capture, elastic scattering, and total reactions, is also available from RSIC.

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Measurement and Calculation of Reactor Parameters

Comparison of Calculations and Integral Measurements for a Uranium/Iron Benchmark Critical Assembly, W. R. Robinson, G. J. Dilorio, J. A. Morman, R. W. Schaefer (ANL)

A series of critical experiments on fully (93%) enriched uranium/iron compositions has been made to support Safety Test Facility (STF) core design efforts.^{1,2} The first several assemblies of the series contained loadings of low fissile density and employed a relatively complex unit cell.^{3,4} The uranium/iron benchmark assembly reported here employed a higher fissile density (U^{235}/Fe atom ratio of 0.016) and was constructed with a simple, one-drawer unit cell, making this assembly suitable for use as a data testing benchmark. The purpose of this paper is to compare measured and calculated results of the critical mass, mockup boron-carbide control rod worth, reaction rate ratios, and reaction rate distributions for the benchmark assembly.

The as-built assembly consisted of a single-zone core of enriched uranium and iron (radius of 62.73 cm) with stainless-steel axial (28-cm-thick) and radial (33-cm-thick) reflectors and an outer radial reflector of depleted uranium (12 cm thick). The unit cell consisted of a single $\frac{1}{8}$ -in.-thick column of fully enriched uranium nearly centered within a 2-in. thickness of iron bars and stainless-steel plates.

All calculations were based on 29-group diffusion theory using ENDF/B-IV data which had been collapsed through the MC-2-II and SDX modules of the ARC system.⁵ Platewise resonance self-shielding of both the intermediate mass and isotopes was treated explicitly.

Criticality was achieved with 991.24 kg of ^{235}U in the reference as-built assembly. Table I compares the measured k_{eff} result with an R-Z diffusion theory eigenvalue calculation. The calculation underpredicts the measurement by 1.4%, a result similar to past experience with LMFBR compositions.

The reactivity worth of a mockup boron-carbide control rod composed of a $\frac{1}{2}$ -in.-wide column of stainless-steel-clad natural B_4C -filled cans centered within a 2-in. thickness of

Fig. 1. Radial fission and capture rate distributions in the uranium/iron benchmark critical assembly.

iron bars was measured relative to an identical composition except for the B_4C itself. The reactivity change was measured by three different techniques⁶; the weighted average is given in Table I. The eigenvalue difference calculation, which used infinitely dilute boron cross sections in the mockup control rod, resulted in a worth C/E of 0.975. This is similar to results obtained in LMFBR compositions.

Reaction rate ratios of ^{238}U fission and ^{238}U capture per atom to ^{235}U fission per atom were determined by stacking thin metallic enriched and depleted uranium foils through a hole in a fuel plate located near the core center and measuring their activation by absolute counting techniques. Fission rate traverses were flat through the $\frac{1}{8}$ -in.-thick fuel plate, within the statistical uncertainty of 1%, while the ^{238}U capture rate decreased at the plate center by about 2%. The plate integrated measurements are compared directly to calculated cell-average reaction rates in Table I. The C/E of 1.1 for c^{238}/f^{235} is in the range previously observed in LMFBR compositions.

TABLE I
Integral Physics Parameter Results from the Uranium/Iron Benchmark Critical Assembly

Parameter	Measured	Calculated	C/E
k_{eff}	1.00203	0.98875	0.987
B_4C control rod worth, lh^a	-299.0 $\pm$ 2.3	-293.5	0.982
Cell-average reaction rate ratios:			
c^{238}/f^{235}	0.1517 $\pm$ 0.0015	0.1659	1.094
f^{238}/f^{235}	0.0203 $\pm$ 0.0003	0.0216	1.059
Fission counter (cavity) reaction rate ratios:			
c^{238}/f^{235}	0.178 $\pm$ 0.002	0.193	1.08
f^{238}/f^{235}	0.0154 $\pm$ 0.0002	0.0131	0.85
f^{40}/f^{235}	1.05 $\pm$ 0.01	0.989	0.94
f^{23}/f^{235}	1.53 $\pm$ 0.03	1.47	0.96
f^{40}/f^{235}	0.184 $\pm$ 0.002	0.182	0.99
f^{41}/f^{235}	1.29 $\pm$ 0.02	1.32	1.02
f^{42}/f^{235}	0.118 $\pm$ 0.005	0.133	1.13

^a1% $\Delta k/k = 423.442$ lh .

Uncertainties assigned to the reaction rate ratios are purely statistical.

Fission counter measurements are also reported in Table I. The measurements involved placing absolute fission counters, each with a "principal isotope" deposit, in a 2-in. cubic central cavity. Capture rates were made by placing foils on the face of one of the counters and subsequently absolutely counting the foil activation. The associated calculations use the cell-average spectrum at the detector location and specifically generated detector cross sections which contain resonance self-shielding, but no plate self-shielding. The uncertainties assigned to the measurement results in Table I are purely statistical and do not reflect the spectral distortion introduced by the cavity.⁷

Radial and axial reaction rate traverses were measured by placing mapping foils adjacent to the fuel column along rays in both directions. The radial traverse results for the principal reaction rates are shown in Fig. 1. The calculated distributions also shown in Fig. 1 used the cell averaged flux and cross sections generated specifically for foils in the reduced fissile density STF 3/4-fuel loading assembly.⁴ The differences between the calculated and measured distributions for all three reaction rates in the reflector region indicate that the calculated spectrum is too soft. The collapsing spectrum used for cross-section generation is significantly different from the spectrum actually present in the reflector region of the uranium/iron assembly and this may be the cause of the overprediction of the calculated capture rates in the reflector region; however, the uncertainties in the foil cross sections can not fully explain the observed differences.

In summary, the C/E ratios observed in the uranium/iron benchmark for k_{eff} , c^{28}/f^{25} , and control rod worth are in the range of previous experience with LMFBR compositions, while f^{28}/f^{25} appears to be substantially overpredicted. The spectrum appears to be calculated too soft in the iron reflector.

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Nuclear-Pumped Lasers

Critical UF_6 Cavity Reactor Experiments,
J. F. Jaminet (UTRC), D. M. Barton (LASL)

Experiments have been conducted to investigate the possibility of producing continuous power from fissile fuel in the gaseous state. Successful development of gaseous fuel reactor technology could open new options for meeting future terrestrial energy needs and, in addition, could have space application as a high-thrust, high-specific-impulse propulsion device.

Experiments utilizing a beryllium reflector-moderator cavity-critical assembly have been performed at Los Alamos Scientific Laboratory (LASL) to establish the feasibility of fissioning UF_6 gas reactors and/or uranium plasma reactors.¹ Cavity configurations of systematically increasing complexity were used to progress from simply distributed uranium foil experiments to more advanced tests involving the vortex confinement of gaseous UF_6 fuel. Accomplishment of the gaseous UF_6 critical experiments at LASL required the development of reliable techniques for handling heated UF_6 throughout extended ranges of temperature, pressure, and flow rate. The present work involved development and testing of laboratory-scale flow systems which utilized gaseous UF_6 at temperatures up to 500 K, pressures up to 4 atm, and continuous flow rates up to 25 g/s. Also, advanced designs of a 200- to 2500-MW self-critical mixed-flow reactor have been studied.

Initially, a UF_6 system was fabricated for static critical tests at LASL.² The system was designed to supply UF_6 at temperatures of 300 to 400 K and pressures up to 4 atm to a double-walled aluminum reactor core canister assembly. The dynamic reactivity characteristics of the system were determined by imposing step changes in temperature and pressure.

A second UF_6 system was designed and fabricated to provide a circulating flow of up to 25 g/s of gaseous UF_6 in a closed-loop through a similar double-walled aluminum re-

actor core canister assembly at various temperatures and pressures. This system was used in cavity reactor experiments at LASL in which a UF_6 /helium mixture flowed through an active cavity reactor core region. A unique feature of the flowing system was the capability of separating UF_6 by desublimation for subsequent reuse from the UF_6 /helium mixture while maintaining a sealed system. Dynamic reactivity measurements were made in the LASL cavity critical assembly at pressures up to 4 atm and with 400 g of 93%-enriched UF_6 within the canister.³

Recently, in-reactor testing of a UF_6 flow system was completed, which provided a UF_6 /He fuel mixture fluid dynamically confined away from the inner wall of the cavity reactor core canister. A continuous closed-loop heated argon flow was used to generate the confining radial-inflow vortex within the core canister. Following separation of the other flow constituents, the argon was stored and pumped as a cryogen. UF_6 was continuously mixed with helium to obtain the proper mixture density for injection into the vortex flow. The continuous open-loop heated helium flow was separated and vented while the argon was being condensed. Gaseous UF_6 was supplied for durations of several minutes from stainless-steel bottles heated in an oven. The UF_6 was separated from the vortex effluent gases by desublimation in a heat exchanger. The double-walled aluminum core canister was designed to allow for seven separate in/out flow paths at one end, with minimum disruption of the cavity reflector-moderator.

Preliminary flow tests were conducted at pressures up to 2.5 atm with UF_6 /He fuel flow rates up to 5 g/s and argon flow rates up to 70 g/s. Noncritical tests of fuel density and fuel distribution using a light absorption method determined that the fuel was confined within 70% of the canister inner radius. Critical tests were conducted at LASL at pressures up to 2.8 atm with UF_6 /helium fuel flow rates up to 2.9 g/s and argon flow rates up to 55 g/s. In one test, a UF_6 exhaust temperature rise of 5 K was recorded at a total reactor power

TABLE I
Conditions for Gaseous Reactor Tests at LASL

A. Gas Flow Conditions:	
Ar Injection Mass Flow	50 g/s
Ar Injection Temperature	70 C
Total Pressure in Canister	2.8 std atm
He Mass Flow	0.2 g/s
Ar Residence Time in Vortex	6.2 s
UF_6 (93% enriched) Residence Time	19 s
UF_6 Flow	1.8 g/s to 2.7 g/s
B. Reactor Conditions:	
Mass of U (93%) in U-C Fuel Pins	7.70 kg
Maximum Reactivity Addition	5 cents/s
Power of Operation (during flow conditions)	1 kW to 30 kW
Total Time of Tests	430 s
C. Experimental Results:	
Reactivity Changes - At 1.8 g/s,	62.6 cents
At 2.7 g/s,	89.5 cents
(Worth of 1 g UF_6 in vortex ~ 1.79 cents)	
Temperature Changes: (i) Ar Buffer Gas -	0.6 C
(ii) PCA Driver Fuel -	79 C
(iii) Center of Exit Port (UF_6 +He+Ar) -	5 C

TABLE II
Operating Characteristics of Mixed-Flow Gaseous Nuclear Reactor

Fuel	UF ₆ + He
Buffer Gas	None
Power, MW(th)	206
Cavity Pressure, atm	100
Average Fuel Temperature, deg K	1200
Cavity Diameter, m	1.0
Cavity Length, m	4.35
Number of Cavities	7
Critical Mass, kg	47
Energy Extraction Mode	Convection
Working Fluid	UF ₆ + He
Working Fluid Temperature, deg K	1200
Working Fluid Flow Rate, kg/s	68
Power Conversion Cycle	Combined He gas turbine & steam turbine
Cycle Efficiency	0.36

of 30 kW with a confined mass of UF₆ of 35 g. The results of these tests are listed in Table I.

The experiments conducted at LASL have demonstrated that a flowing gaseous fission reactor is indeed feasible. The advantages of a gaseous reactor over present reactors are substantial: (a) an inherent low inventory of fissile material; (b) minimal transport of radioactive material due to on-site reprocessing of fuel; (c) potentially high cycle efficiencies due to higher operating temperatures; (d) low doubling times of one to three years for a breeder configuration; and (e) increased safety and greater nuclear fuel resource utilization with significantly reduced proliferation risk.

A conceptual design study has been made* of an ~200-MW mixed-flow gaseous nuclear reactor. Some of the results of this study are presented in Table II.

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Criticality Safety and Three Mile Island

Criticality Safety Considerations for TMI-2, F. M. Alcorn, M. R. Gudorf (B&W)

This paper presents a summary of evaluations made by the Babcock & Wilcox Company (B&W) in support of its customers, General Public Utilities (GPU), to assess the potential for recriticality of the damaged Three Mile Island Unit 2 Reactor (TMI-2) core and to provide a basis for recommendations to prevent the occurrence of such an event. The TMI-2 incident occurred on March 28, 1979; on March 31, B&W formed a Criticality Analysis Task Force composed of personnel from the Company's reactor physics unit and their nuclear criticality safety unit. The Task Force at maximum strength utilized ten individuals and completed all pertinent evaluations in two weeks; advisory activity continued for several months.

Two concurrent evaluations were made for all probable (and a few improbable) fuel configurations that could be identified. One was made by reactor physics personnel using reactor design codes, while the other was made by nuclear criticality safety specialists using criticality safety codes. In addition to evaluating specific fuel configurations, basic data such as K-infinite vs fuel-to-water ratios were tabulated for different enrichments, fuel particle sizes, water temperatures, boron levels, etc.

Physics computer codes used by the Task Force included: PDQ07,¹ KENO4,² NULIF,³ and AMPX.⁴ NULIF is a B&W-developed code which generates input to PDQ07 and calculates infinite-medium nuclear parameters; NULIF uses a mixed cross-section set based on both the ENDF/B-III cross-section set and company-evaluated cross sections. AMPX is a computer code package from which the cross-section handling code NITAWL and the one-dimensional transport code XSDRNPM was used. All AMPX/KENO runs were made using the XSDRN 123-energy-group cross-section library.

Since a description of the damaged core was not available, the Task Force was asked to recommend a boron content for the primary system water to prevent the possibility of recriticality. Infinite-medium calculations were performed using both NULIF and XSDRNPM for UO_2 pellets at varying volume fractions of fuel and soluble boron content. Auxiliary studies were made to identify the effects of fuel particle size and coolant temperature. No credit was taken for fuel depletion or for the presence of any neutron poison material; an unlimited quantity of UO_2 was assumed. In general, XSDRNPM predicted K_{∞} 's about 2% $\Delta K/K$ higher than NULIF. XSDRNPM results indicate that 4800-ppm boron would be required to maintain $K_{\infty} < 1.0$ for the average core enrichment of 2.6 wt%; NULIF results would indicate that 4000-ppm boron would be required for the same condition.

Refinements in the above calculations to take credit for both depletion and the actual amount of UO_2 present demon-

strated that the TMI-2 fuel at an average enrichment could be held slightly subcritical at 3000-ppm boron assuming the fuel to be in a reflected sphere at the optimum fuel-to-water volume ratio (about 63% fuel in borated water). Again, no credit is taken for any structural, cladding, or neutron absorber material. Based on these calculations the Task Force recommended a minimum boron concentration of 3000 ppm subject to other plant effects and control over reactor state changes; a level of 4000 ppm was recommended to cover all hypothetical out-of-core conditions. The boron level of TMI-2 has been maintained at >3500 ppm since the incident.

A great number of calculations were made for postulated damaged core configurations. The most credible situation evaluated is one in which some portion at the top of the core is damaged and has partially collapsed while the lower portion of the core is still intact. To conservatively simulate this, a calculation was made with KENO in which it is assumed all fuel above the fourth grid has slumped on the fourth grid and all fuel below this grid is standing in its undamaged configuration. This is shown conceptually in

K_{eff} BY KENO4 = 0.998 ± .007

ASSUMING AN INFINITE X-Y MEDIUM

3,000 ppm BORON

280°F

FRESH FUEL, AVG. ENRICHMENT (2.6)

NO STRUCTURAL MATERIAL OR RODS

K_{eff} ADJUSTED FOR LEAKAGE AND FUEL DEPLETION

NO RODS 0.957

WITH RODS 0.928

Fig. 1. Damaged core—credible but conservative.

Fig. 1. This calculation assumed an infinite array in the x-y direction and took no credit for fuel depletion, structural material, or neutron poison material. Adjusting the "raw" KENO value of 0.998 for fuel depletion, radial leakage, an increase in boron concentration to 3560-ppm boron, a decrease in coolant temperature to 160°F, and assuming 60% of the control rod worth to still be available yields a conservative estimate for K_{eff} of the TMI-2 core of 0.93. This value is conservative in that the degree of damage is considered to be much more severe than actually present.

In conclusion, the TMI-2 core is subcritical with a more than adequate safety margin. Subcriticality can be easily maintained through control over the amount of boron dissolved in the primary coolant system water.

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TMI-2 Post-Accident Criticality Analyses, J. P. Colletti, J. A. Easley, J. D. Luoma (GPU Serv Corp)

INTRODUCTION

Avoiding a return to criticality, whether localized or global, in the present TMI-2 core is of primary importance in maintaining a long-term cold shutdown state and the subsequent safe disposal of the fuel. To this end, a series of reactivity analyses was performed at GPU to supplement those existing analyses available shortly after the accident, and to encompass all plausible post-accident coolant/core conditions.¹⁻⁶ All studies were aimed at finding what fuel configurations would be most reactive for given core conditions, and what soluble boron concentration would be required to keep the configuration subcritical. Also, reac-

tivity dependence on coolant temperature, pressure, and density was studied. In all GPU studies, no credit was taken for the negative effect of the control rods and burnable poisons which were in the TMI-2 core. However, core leakage and burnup were considered in recommending a minimum safe boron concentration.

DISCUSSION

Fuel configurations were separated into two states: undamaged and damaged. For the undamaged state, the reactivity of an infinite array of intact TMI-2 fuel assemblies was calculated for various coolant pressures, moderator temperatures, moderator densities, boron concentrations, and fuel pin pitches. These and all infinite media calculations were performed using the computer code XPOSE,⁷ an Exxon Nuclear revised version of LEOPARD.

The diffusion code PDQ07 (Ref. 8) was used to calculate the reactivity of an entire core of undamaged fuel assemblies for the worst case of reduced moderator density. For each fuel assembly type, cross sections were generated at the water density which had yielded the highest k_{∞} in the infinite media studies.

Damaged fuel calculations were motivated by conjecture that fragmented UO_2 pellets may have been freed from the fuel cladding and are loose in the reactor coolant. For the damaged fuel, reactivity calculations were made for infinite arrays of pure UO_2 and boroated water. A homogeneous mixture was modeled by placing both fuel and water in all three regions (pellet, cladding, and moderator) of the unit fuel cell. Heterogeneous mixtures were modeled by infinite arrays of fuel pins in water with no cladding or structural material. Fragmenting of the pellets was approximated by varying the fuel pellet diameter and adjusting the pitch to achieve different fuel volume fractions. For all the damaged fuel studies, the middle core enrichment of 2.644 wt% was used.

RESULTS AND CONCLUSIONS

Table I gives the conditions which yielded the highest reactivity for each of the fuel configurations studied. The main conclusions reached as a result of the GPU analyses are as follows:

1. The damaged fuel assemblies, wherein fuel pellets and boroated water are mixed in absence of other material, would be more reactive than the nominal configuration of undamaged assemblies under the same subcooled coolant conditions.

TABLE I
Summary of Worst Cases for Conditions Analyzed by GPU

Fuel Condition	Geometry Condition	Enrichment (wt%)	UO_2 Fuel Volume Fraction	Water Density (g/cm ³)	Coolant Temperature (°F)	Press. (psi)	Highest Boron (ppm)	K_{∞}
Undamaged Fuel	Infinite Media (XPOSE)	2.965	Nominal Pin Pitch (0.568 in.)	0.39	215	Boiling	4000	1.0014
Undamaged Fuel	Finite Media (PDQ)	Core Geometry	Nominal Pin Pitch	Worst case per wt% Cross Section	215	Boiling	3400	0.940
Damaged Fuel	Infinite Media Homogeneous (XPOSE)	2.644	0.57	Nominal	280	1000	4000	0.942
Damaged Fuel	Infinite Media Heterogeneous (XPOSE)	2.644	0.59 (full-size pellets)	Nominal	50	15	4000	1.0106

2. A whole pellet is more of the reactivity problem than fragments of fuel pellets. Also, a fuel volume fraction in boric water of 0.55 to 0.65 would be the most reactive for an average enrichment slurry. A concentration above 4000-ppm boron would be required to keep this fresh, nonpoisoned mixture subcritical.

3. As moderator temperature is reduced, the system becomes *more* reactive, everything else being the same for $\text{UO}_2/\text{H}_2\text{O}$ mixtures ($\alpha_T \approx -1.0 \times 10^{-4} \Delta\rho/^\circ\text{F}$).

4. As moderator temperature is reduced, the system becomes *less* reactive, everything else being the same, for a nominal undamaged fuel assembly configuration ($\alpha_T \approx +0.8 \times 10^{-4} \Delta\rho/^\circ\text{F}$).

5. For undamaged fuel, as water density is reduced, reactivity peaks for water densities between 0.3 and 0.5 g/cm^3 . Also, a lower coolant temperature will create a less reactive system at high boron concentrations.

6. The most reactive fuel/cladding configuration at a high boron concentration is the nominal pitch case for a low-density coolant.

7. Reactivity changes are negligible as pressure is changed for a subcooled moderator.

Since the core configuration is not known now and could take many forms before head removal occurs (~2 years), the possibility exists that an undesirable fuel state could materialize. Considering the worst cases mentioned above and the reactivity effects of fuel burnup ($\sim 2.5\% \Delta k/k$) and neutron leakage ($\sim 1.6\% \Delta k/k$) along with a boron worth of $\sim 0.41 \times 10^{-4} \Delta\rho/\text{ppm}$, maintenance of a 3500-ppm boron concentration is recommended for TMI-2.

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6. "Reactor Core Analysis," TDR-048, GPU Service Corp. (Sep. 28, 1979).
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Recriticality Potential of TMI-2 Core, C. R. Marotta (NRC)

The 61 movable (Ag-In-Cd) control rods successfully scrambled the TMI-2 reactor in the first few seconds of the event. Subsequent core uncovering over extended periods of time led to fuel temperatures in excess of 1750°C and thus left uncertain the status of these control rods as well as the burnable poisons ($\text{B}_4\text{C}-\text{Al}_2\text{O}_3$) contained in 68 fuel assemblies. Hence, a potential for recriticality existed for the damaged reactor (possibly until dismantled) relying ex-

clusively on the boron (boric acid) concentration in the moderator/coolant to guarantee subcriticality. This paper presents the methodology and results of the NRC criticality studies performed during the early stages of the accident to establish necessary boron concentrations for a variety of assumed damaged core conditions.

The objectives of the analyses that were successfully addressed were:

1. Establish the minimum boron concentration for:
 - a. a fully intact undamaged latticed core without control rods or burnable poisons
 - b. the most-reactive intact damaged latticed core without control rods or burnable poisons
 - c. a fully pelletized damaged core without control rods or burnable poisons.
2. Establish sensitivity of reactivity to varying boron concentrations of most reactive intact latticed core as a function of radial regions; i.e., consider gross boron hideout as a function of radial core volumes.
3. Establish conditions (if any) of a possible local criticality.

The mathematical-criticality analyses were performed using the KENO¹ Monte Carlo computer program together with the 123-group GAM THERMOS² neutron cross-section set. Fuel pin cell k_{∞} calculations were performed to establish the most reactive lattice pitch as a function of ^{235}U enrichment and boron concentrations. Three-dimensional (3-D) KENO k_{eff} calculations were used to establish the reactivity of the reactor under a variety of states. Since the TMI-2 coolant will eventually reach room temperature, all criticality analyses (with the exception of the hot, clean benchmark TMI-2 configuration) were performed at this most reactive (neutronically) temperature. The moderator density was taken as 1.0 g/cm^3 and the fuel (UO_2) density was assumed as 95% theoretical.

The core was modeled (containing only latticed fuel pellet-cladding-moderator) geometrically in 3-D, quarter symmetry, explicitly describing every fuel rod (pin diameter of 0.9398 cm, active height 366 cm, Zr-clad 0.0673 cm thick) at the pitch under consideration. Only fuel rods exist in this model; all water holes contain a fuel rod. This model assumes 36,864 (versus 36,816 actual) fuel rods, giving only 0.1% more fuel than actually in the core. The central "checkerboard mixture" of 1.98 and 2.64% enriched fuel assemblies were modeled as assemblies having an effective enrichment of 2.31%, occupying two distinct regions: a central square (33% of core) surrounded by a square annulus (33% of core) of identical rods. The outer portion, containing 2.96% enriched fuel assemblies, forms the last square annulus (34% of core). A different boron concentration can be specified for each of these three regions. A two-foot unborated water, all-around reflector surrounds the above-described core. This latter modeling representing external regions of the core introduces approximately a +0.5% conservatism in k .

Some confidence is established in the above calculational procedure for the configurations of interest noting the successful KENO run for the critical TMI-2 core (zero power, hot 530°F , clean, all rods out). This critical configuration had a boron level of 1500 ppm. The k_{eff} calculated for this configuration at temperature was 1.050 ± 0.004 . Since the fixed B_4C burnable poison rods were estimated to have worth of 4.5% in k_{eff} and the modeling assumed these rods to be lost, the agreement can be considered excellent. The 0.3% core volume occupied by stainless steel, which is

also neglected in the model, is not expected to change the final k_{eff} since the Δk will be less than the order of the uncertainty in the Monte Carlo calculations.

The above modeling gave the following answers to the above-stated objectives.

1. a. k_{∞} cell calculations for an infinite undamaged array of TMI-2 fuel assemblies at highest enrichment

of 2.96%, would be subcritical with 2350-ppm boron in the coolant.

- b. k_{∞} cell calculations performed as a function of rod pitch, boron concentration, and ^{235}U enrichment indicate a maximum reactive lattice to exist for a rod spacing of 1.26 cm (versus the 1.44 cm as-built). Results are given in Fig. 1 and clearly indicate the

Fig. 1. KENO k_{∞} of TMI-2 pin cell as a function of enrichment, pitch (W/F), and boron concentration.

TABLE I

k_{eff} of TMI-2 Core as Function of ppm Boron in Water
(No Control Rods, Burnable Poisons, or Fission Products)
(Room Temp)

As-Built Pitch 1.44 cm ppm Boron					Most Reac Pitch 1.26 cm ppm Boron				
Zr-Clad	A(1)	B(2)	C(3)	k_{eff}^a	Zr-Clad	A(1)	B(2)	C(3)	k_{eff}^a
YES (cold)	1500	1500	1500	1.040	YES	3000	3000	3000	0.944
YES (hot)	1500	1500	1500	1.050					
YES	3000	3000	3000	0.883	YES	3000	3000	2000	0.954
NO	3000	3000	3000	0.857	YES	3000	3000	1500	0.989
					YES	3000	3000	1000	0.992
					NO	3000	3000	3000	0.936
					NO	2500	2500	2500	0.977
					NO	3000	2500	2000	1.000

- (1) Region A is central cuboid, containing 12,100 fuel rods at 2.31% enrichment (no water holes).
 (2) Region B is annular cuboid surrounding A, containing 12,236 fuel rods at 2.31% enrichment (no water holes).
 (3) Region C is outer annular cuboid surrounding B, containing 12,528 fuel rods at 2.96% enrichment (no water holes).
 Regions A, B, and C are 12 ft in height; Regions are surrounded by 2 ft unborated water all around reflector.

^aAll k_{eff} calc. by KENO -123 Gps, using 15,000 neutron histories and all within ± 0.004 in k_{eff} for 1 standard deviation.

reverse trend of k vs W/F for standard LWR under-moderated fuel assemblies in unborated water.

- c. Unclad UO_2 spheres resulting from cladding rupture of all rods would be the most reactive configuration being kept subcritical with at least 3500-ppm boron.

2. Table I gives k_{eff} 's calculated for TMI-2 full-core modeling under a variety of boron concentrations in the three radial core regions indicating the controlling aspects of the outer highest enriched region.

3. For a local criticality, four contiguous fuel assemblies with highest enrichment at the most reactive pitch were calculated to go critical for zero boron concentration.

The major conclusions and recommendations from the analyses undertaken are:

1. A 3500-ppm boron level guarantees subcriticality for all conceivable abnormal states of the TMI-2 core.

2. The peripheral highest enriched region of the core is shown to be the most sensitive and controlling to boron concentrations.

3. Regardless of the boron concentration throughout the core, a slug of completely unborated water passing through a minimum of four contiguous (in a square) fuel assemblies, the full length of the core, would cause a criticality.

4. Special note should be made of reactivity increases for reduced pitches for borated lattices as indicated in Fig. 1.

This would especially be applicable for guidance during core dismantling where intact pitches may have to be reduced for assembly removal.

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TMI-2 Criticality Analysis—Analytical Models and Methods, J. T. West, R. M. Westfall, G. E. Whitesides, J. T. Thomas (UCC-ND)

The purpose of this paper is to describe the calculational models and analytical methods employed in a comprehensive study of reactivity effects due to hypothetical core disruptive mechanisms associated with the accident at Three Mile Island Unit 2. The TMI core design is described with regard to the heterogeneity arising from various fuel assembly configurations. Then the geometric lattice modeling capabilities of MORSE-SGC/S¹ and KENO-IV² are explained. After discussing the calculational difficulties and the analytical tools, the development of the disrupted core models is presented along with the procedure utilized in generating macroscopic cross sections. Verification of the analytical calculations includes a discussion on prior experience with the 27-neutron-group cross-section library³ and results of a benchmark

calculation of the reactor at the hot, zero-power startup condition.

The TMI core (see Fig. 1) is a Babcock & Wilcox 177 fuel assembly plant containing three fuel enrichments; 1.98, 2.64, and 2.96 wt%. Each assembly may contain one of three types of lump burnable poison rods, two types of control rods, or stainless-steel orifice rods. Preservation of the assembly loading pattern was considered to be essential in studying reactivity changes due to disruptive core mechanisms.

The multiple-array lattice capability of MORSE-SGC/S with the new MARS geometry system and the single-array capability of KENO-IV with the MIXED BOX geometry system allowed pin-by-pin modeling of the TMI design core and several postulated disrupted core models. The most complicated model was done with MARS geometry which allows multiple-array modeling, unlimited array nesting, arbitrary vacancies in arrays, and array repetition by both translation and rotation. The simpler models were done with the KENO MIXED BOX geometry which allows modeling a single KENO array without repetition. The KENO analysis was performed to validate and back up the MORSE-SGC/S effort. It also gave significant additional insights into the reactivity effects of specific core disruptive mechanisms.

The Three Jump Slump Model studied with MORSE-SGC/S was designed to incorporate all of the core disruptive

mechanisms in an internally consistent manner; fuel swelling in lower-half of core and conical fuel slumping in upper-half of core with a transition region in the middle of the core. This model (see Fig. 2) contained 990 short assemblies of 15 X 15 X 1 pins each. The Three Jump Slump core was modeled as a 15 X 15 X 7 array of subarrays and bulk material regions using the MARS geometry system.

The Displaced Fuel Slump Model studied with KENO-IV features uniform fuel slumping without fuel swelling. The In-Place Fuel Slump Model features pin swelling without any fuel density changes or cladding failure. The Displaced Fuel Slump Model and the In-Place Fuel Slump Model were each based on a 120 X 120 X 1 KENO MIXED BOX orientation array representing all pin lattice locations in the quarter-core geometry of Fig. 1. The program MAKARAY was written to generate the KENO MIXED BOX input.

All of the analyses utilized the SCALE 27 neutron-group cross-section library based on ENDF-IV. The resonance isotopes were corrected for resonance self-shielding via the Nordheim Integral treatment⁴ in the NITAWL code. Prior criticality studies⁵ using this library on several EPRI critical experiments⁶ indicated that for the coolant-to-fuel volume ratio and corresponding lattice pitch of the TMI design core, the K-effective calculated would range from 0.98 and 0.99.

Pin-by-pin models of the TMI design core (as-built) were analyzed with MORSE-SGC/S and KENO-IV. As anticipated

*BATCH A IS DISCHARGED AT THE END OF THE FIRST CYCLE

Fig. 1. Cycle One fuel loading scheme Three Mile Island Nuclear Station Unit 2.

Fig. 2. MORSE-SGC/S Three Jump Slump Core Model.

with the 27-group library, the MORSE-computed K-effective was 0.987 ± 0.003 and the KENO-computed K-effective was 0.983 ± 0.006 . Calculated control rod worths agreed well with measured values. This hot, zero-power startup benchmark analysis provided a verification of the analytical methods and modeling techniques for the subsequent disrupted core analyses.

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TMI-2 Criticality Analysis-Parametric Studies and Overall Results, R. M. Westfall, J. T. West, G. E. Whitesides, J. T. Thomas (UCC-ND)

Three hypothetical disrupted core models were analyzed for the President's Commission on the Accident at Three Mile Island Unit 2. The analytical models and methods applied in this study are described in a companion paper.¹ The purpose of this paper is to provide a summary of the results obtained from preliminary parametric studies and

from the analyses of the disrupted core models. Hopefully, the implications from these results are useful in planning plant recovery operations. However, the scope of this study did not include a quantitative determination of the probable condition of the reactor core or, on the other end, the development of recommendations for specific actions to assure the criticality safety of the plant. A detailed description of this study is given in Ref. 2.

One-dimensional, discrete-ordinates analyses were performed to study the reactivity effects of geometric changes in the fuel pins and the interstitial coolant. The core disruptive mechanisms studied were:

1. fuel pin lattice-pitch reduction
2. fuel pin swelling
3. conversion from pin cell geometry to homogeneous $U_3O_8 + UO_2 + H_2O + B$ mixtures.

These geometry changes can be generally characterized as variations in the water-to-fuel volume ratio. One limit to this variation is the case of an infinite medium of dry $U(2.96)O_2$ which has a multiplication factor of 0.66. Thus, some content of water and its associated neutron moderation must be present for the system to be critical.

Pressurized water reactor fuel is normally considered to be undermoderated; that is, at less than an optimum water-to-fuel volume ratio for maximum reactivity. Undermoderation can be inferred for the unbored water case shown in Fig. 1. Reducing the lattice pitch from the design value lowers the multiplication factor. However, for the "cold boded" situation, the opposite effect is observed. The most reactive lattice pitch is significantly less than the design value. Eventually, the negative reactivity due to the loss of water overtakes the positive reactivity due to the loss of boron and the system multiplication factor comes back down.

The same behavior was observed with fuel swelling and coolant displacement by ZrO_2 . The overall effect for the transition region of the "Three Jump Slump" model was

Fig. 1. TMI infinite lattice pitch variation.

TABLE I
Summary of TMI-2 Multidimensional Core Analyses

Model	Case No.	Case Description	Monte Carlo Code	Multiplication Factor
As-built, normal core at cold shutdown with 3180-wppm boron	1a	Base configuration	KENO-IV	0.737 ± 0.006
	1b	Base configuration	MORSE-SGC/S	0.752 ± 0.007
	2	Case 1 with control rods out	MORSE-SGC/S	0.805 ± 0.006
	3	Case 1 with LBP rods removed	MORSE-SGC/S	0.778 ± 0.008
"Three Jump Slump," Disrupted Core	4	Control and LBP rods out	MORSE-SGC/S	0.819 ± 0.007
	5	Base configuration ^a	MORSE-SGC/S	0.862 ± 0.006
	6	Case 5 with control rods out	MORSE-SGC/S	0.875 ± 0.006
	7	Case 5 with LBP rods out	MORSE-SGC/S	0.868 ± 0.006
	8	Control rods and boron ^b out	MORSE-SGC/S	1.079 ± 0.012
	9	LBP rods and boron ^b out	MORSE-SGC/S	1.043 ± 0.010
	10	Control rods in, boron ^b out	MORSE-SGC/S	0.988 ± 0.011
"Displaced-Fuel Slump," Disrupted Core	11	Base configuration ^c	KENO-IV	0.845 ± 0.006
	12	Case 11 with control rods out	KENO-IV	0.870 ± 0.006
	13	Case 11 with boron ^b out	KENO-IV	1.080 ± 0.006
"In-Place Fuel Slump," Disrupted Core	14	25% swelling, ^d clad o.d. = 1.179 cm	KENO-IV	0.807 ± 0.006
	15	50% swelling, clad o.d. = 1.273 cm	KENO-IV	0.845 ± 0.005
	16	75% swelling, clad o.d. = 1.360 cm	KENO-IV	0.840 ± 0.006
	17	100% swelling, clad o.d. = 1.443 cm	KENO-IV	0.812 ± 0.073

^a13.5% of upper middle core collapsed as U_3O_8 - UO_2 - H_2O mixture, ZrO_2 distributed in coolant channels of lower core, fuel pins swollen by 30%.

^bBoron remaining in homogeneous, fueled portions of reactor.

^cUpper 50% of core collapsed as U_3O_8 - UO_2 - H_2O mixture, upper portions of control and LBP rods missing, lower half of core normal.

^dFuel height reduced to conserve fuel volume, clad volume constant, base configuration is Case 1A.

+5.5% k/k_2 with slightly more than half of the positive reactivity being due to fuel swelling. An even larger effect is seen in converting from the pin cell to the homogeneous fuel geometry. The water-to-fuel volume ratio went from 1.65 to 0.46 and the reactivity increased by 8.2% k/k_2 . As a function of water-to-fuel volume ratio, undermoderation sets in at a value less than 0.6 for systems with 2400-wppm boron and at a value less than 0.4 for systems with 3180 wppm.

The results of the multidimensional core analyses are summarized in Table I. Comparison of the multiplication factor for the cold-shutdown, normal-core base case with those for the three disrupted core models led to the following conclusions for this heavily borated system:

1. Positive reactivity insertions due to the various core disruptive mechanisms increased the system multiplication factor from ~0.74 to 0.86.

2. To a first-order approximation, the increase in reactivity for the three models can be correlated with a decrease in the borated water-to-fuel volume ratio.

3. The reactivity worths of the control rods and lumped burnable poison rods are significantly reduced by the high soluble boron content in the reactor.

4. Although the core disruptive mechanisms have introduced significant positive reactivity insertions, the disrupted core is still shut down by a substantial margin.

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Data and Analysis for Nuclear Criticality Safety

Criticality Safety Criteria, W. Marshall (CEGB-UK), P. D. Clemson (BNFL-UK), G. Walker (UKAEA-Warrington)

INTRODUCTION

This paper describes three criticality safety criteria for use with computed values of k_{eff} . Two of the criteria are currently used in safety submissions within the UK; the third could be employed if additional statistical refinement were required. It is legitimate to use any one of the criteria, provided the associated data are chosen prudently. It is stressed that other formulations could be equally acceptable.

CRITERION 1

$$k_N + EPD + ESM + ER + 3\sigma_N \leq L \quad (1)$$

This criterion¹ is used with the UK Monte Carlo code MONK.² k_N , σ_N are computed values of k_{eff} and its standard deviation for the new case. EPD and ESM are allowances for errors due to program, data, specification, and modeling of the system. ER allows for the possible existence of situations more reactive than that modeled, and L is a limit $\leq$ unity which can give additional pessimism, or impose a particular shutdown margin.

EPD values have been obtained³ for many systems from validation comparisons between calculated and experimental results. Values for the other allowances are selected by the assessor. The overall confidence level for the criterion cannot be stated, because some of the allowances confound systematic and random effects. Prudent choice of values can offset this problem, and will ensure adequate safety.

This criterion has been used by BNFL in submissions for process plant which have been accepted by the Nuclear Installations Inspectorate.

CRITERION 2

$$k_N + ST + 3\sigma_T \leq L \quad (2)$$

This criterion⁴ is currently used by CEGB for AGR fuel. Systematic and random effects are separated to give totals ST and σ_T . SD and σ_D , the effects due to errors in data and code, are obtained⁵ from the (k_i, σ_i, M_i) , $i = 1 \dots n$, of the n relevant validation comparisons, where M is number of stages:

$$\bar{k} = \sum_{i=1}^n k_i(\sigma_i)^{-2} / \sum_{i=1}^n (\sigma_i)^{-2} \quad (3)$$

$$(\bar{\sigma})^2 = \sum_{i=1}^n (\sigma_i)^2 M_i / \sum_{i=1}^n M_i \quad (4)$$

$$SD = 1 - \bar{k} \quad (5)$$

$$(\sigma_D)^2 = \left[\frac{n \sum_{i=1}^n (k_i - \bar{k})^2 (\sigma_i)^{-2}}{(n-1) \sum_{i=1}^n (\sigma_i)^{-2}} \right] - (\bar{\sigma})^2 \quad (6)$$

Data³ for low-enriched U/VO₂ rods in water lead⁶ to SD = 0.0, σ_D = 0.011.

$$ST = SD + S_a + S_b + \dots \quad (7)$$

$$\sigma_T = [\sigma_N^2 + \sigma_D^2 + \sigma_a^2 + \sigma_b^2 + \dots]^{1/2} \quad (8)$$

where a, b, ... are separate effects external to the code.

This criterion has been used in transport submissions approved by the UK Competent Authority. It has also been used in assessments for power stations accepted by the Nuclear Installations Inspectorate.

CRITERION 3

$$k_N + ST + t_{p\nu}\sigma_T \leq L \quad (9)$$

This gives a more precise treatment of confidence level, allowing for the limited availability of validation comparisons. $t_{p\nu}$ is Student's t at a confidence level of p for ν degrees-of-freedom. ν is obtained from the individual degrees-of-freedom f_N , f_D , f_a , ... of the components of σ_T by application⁶ of Welch's⁷ analysis.

$$\nu = \frac{\sigma_T^4}{\left[\sigma_N^4/(f_N + 2) + \sigma_D^4/(f_D + 2) + \sigma_a^4/(f_a + 2) + \dots \right]} - 2 \quad (10)$$

f_N and f_D are equal to $(M_N - 1)$ and $(n - 1)$, respectively.

Although t is greater than the corresponding factor from the normal distribution, it can be shown that an excessive penalty is only incurred when (a) only two or three validation comparisons exist, and simultaneously (b) σ_D is the dominant contributor to σ_T .

The confidence level treatment offered by this criterion makes it suitable for analyses of total risk.

COMPARISON OF THE CRITERIA

Assume that (a) 100 stages were used for a hypothetical MONK 4 calculation for AGR elements in water, giving $\sigma_N = 0.015$, (b) the worst situation was represented, and was modeled accurately, (c) there were no sources of error outside the program.

Recommended³ values of EPD and L are 0.03 and 0.95. From the assumptions above, ESM = ER = 0.0. Criterion 1 becomes

$$k_N \leq 0.95 - 0.03 - 3(0.015) = 0.875 \quad (11)$$

For Criterion 2, SD is 0.0, σ_D is 0.011. No "external" allowances are required, so ST is 0.0, σ_T is 0.0186. Then,

$$k_N \leq 0.95 - 3(0.0186) = 0.894 \quad (12)$$

Equation (10) gives $\nu = 44.2$. At the 99.9% level, t is 3.285, and Criterion 3 becomes

$$k_N \leq 0.95 - (3.285)(0.0186) = 0.889 \quad (13)$$

The criteria thus give similar results for this example.

STATISTICAL LIMITATIONS

The true statistical distributions of the variables used in criticality calculations are not known exactly. Thus, calculated confidence levels are not precise. For the same reason, statistical treatments more complex than those above may not give real improvements.

CONCLUSIONS

If it is not possible to separate the systematic and random effects which must be allowed for, prudent use of Criterion 1 is acceptable.

If separation of systematic and random contributions is possible, use of Criterion 2 is preferable.

Criterion 3 should be used if a more precise statistical treatment is required, because Student's *t* distribution is used to calculate confidence limits for k_{eff} . The formula needed for number of degrees-of-freedom has been derived.

Criterion 3 could be used in analyses of total risk.

Data for use with the criterion must be appropriate to the particular situation being studied.

Additional statistical refinements may not lead to real improvements in the quality of the final information.

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Sheba: A Solution Critical Assembly, R. E. Malenfant, H. M. Forehand, J. J. Koelling (LASL)

Sheba, a clean geometry critical assembly employing fuel of 4.8% enriched uranium as the fluoride, was constructed during 1980. The primary applications of the machine are to evaluate accidental criticality alarm detectors for enrichment plants, to provide radiation spectra and intensity measurements to benchmark calculations on a low-enrichment solution system, and to provide radiation fields to calibrate personnel dosimetry. Although not in the immediate experimental plan, it is intended to work toward a solution burst machine. Even the name, Sheba, is the acronym for Solution High Energy Burst Assembly. To that end, experiments following the radiation measurements are aimed at evaluating prompt critical quench mechanisms in a bare, cylindrical, low-enriched solution. Design, construction, and applications of Sheba are discussed.

BACKGROUND

Goodyear Atomic Corporation had requested the assistance of the Oak Ridge National Laboratory in the evaluation of the spacing for nuclear criticality alarm detectors for enrichment plants. Due to uncertainties in the calculation and the high cost of safety through conservatism, Goodyear and ORNL requested that LASL run a solution critical to evaluate neutron and gamma leakage from a solution of relatively low-enrichment uranyl fluoride. An extension of the experiments to evaluate personnel dosimetry was obvious. It is interesting to note that all accidents since 1958

Fig. 1. Sheba: A solution critical assembly.

have involved solutions, although dosimetry is evaluated with metal assemblies such as the HPPR.

DESIGN

Sheba is of the simplest possible design in keeping with its application to benchmark calculations. The reactor vessel is a simple stainless-steel cylindrical tank with a 6.35-mm ($\frac{1}{4}$ -in.)-thick wall. A single safety rod along the axis provides shutdown without sacrificing cylindrical symmetry. Control is affected by varying solution level through calibrated metering valves. Rapid shutdown is accomplished by solution dump through a three-inch pinch valve. The entire reactor assembly, including storage and dump tanks, is mounted on a pallet so the distance above the ground may be varied. To facilitate measurements out to 300 m, Sheba is housed in a thin metal shed outside of Kiva I at the Los Alamos Critical Assemblies Laboratory. Figure 1 illustrates Sheba's simplicity.

The concentrated uranyl fluoride [$H/U \approx 550$, $\rho(U) \approx 1$ kg/litre] is highly corrosive ($pH \approx 1$). However, the nature of the facility is such that a leak could be tolerated. When operating at high power (~ 2 kW), radiolytic gas should form at the rate of 1 litre/min (indeed, a part of the experimental effort during late 1980 will be to evaluate the rate of formation of radiolytic gas). For purposes of the radiation experiments, the gas will be bled into a large holding tank to maintain a noncombustible atmosphere and to allow for the decay of fission product gases under confinement.

MEASUREMENTS

In addition to the evaluation of criticality alarm response to steady-state, ramp, and simulated burst operation, neutron and gamma spectra were evaluated and the responses of personnel dosimeters were determined. Measurements were made from near contact out to 300 m for several core elevations in free air, using shields of steel and concrete. Burst operation was simulated by establishing a point slightly below delayed critical and pulsing the system with a neutron generator.

EXTENSION

Two additional applications of Sheba are being pursued. We hope to continue to provide the facility for extensions of the measurements described above. In addition, we would like to pursue the development of a true solution burst capability with information generated from experiments with the present assembly. We propose to separately evaluate radiolytic gas formation and thermal expansion as quench mechanisms in solution burst machines. A high pressure (6 atm) head has been fabricated for Sheba which will allow the demonstration of depressurization of an operating solution assembly as a scram mechanism.

CONCLUSION

Sheba has demonstrated its utility to evaluate criticality alarm systems, personnel dosimetry, and benchmark calculations. In addition, the critical assembly machine is not only eminently useful for continued applications as designed but lends itself well to the analysis of prompt critical excursions in solution systems.

Application of Exponential Experiment to High Subcriticality Determination, T. Suzuki, Y. Komuro, H. Tsuruta, I. Kobayashi (JAERI-Japan)

From the viewpoint of criticality safety, there arises a need to develop better methods to evaluate high sub-

criticality of nuclear fuel systems with light water as neutron moderator. For this purpose, an exponential experiment technique was examined on water-moderated 2.6 wt% enriched UO_2 lattices at a critical assembly, TCA, of the Japan Atomic Energy Research Institute.

The lattices were assembled in a tank of 1.83-m diam with fuel rods which were made of 94.9% TD UO_2 pellets of 12.5-mm diam and a 0.76-mm-thick Al cladding of 14.2-mm o.d. The fuel rods were arrayed in square lattices $n \times n$, where $n = 17, 15, \dots, 3$, with lattice pitch of 19.56 mm. The central fuel rod was replaced by an Al tube of 18-mm o.d., in which a BF_3 counter was traversed to measure the vertical neutron flux distribution and a ^{252}Cf neutron source of 5.5 mCi was positioned at an elevation of 16.8 cm lower than the bottom of fuel active zone. A polyethylene rod, 18 cm long, was set between the source and the fuel active zone to shield the direct neutron beam. The active height of the lattices was fixed at 122.5 cm, which was the critical water level of the $n = 17$ lattice.

The measured distributions of thermal-neutron flux are shown in Fig. 1 for the lattices of $n = 15$ to 3. The decay constant ν obtained from the exponential distribution is related to an effective multiplication factor k_{eff} under the one-group diffusion model with continuously slowing down sources as

$$k_{eff} = \exp[-M^2(\nu^2 + B_g^2)] \quad (1)$$

Fig. 1. Vertical flux distribution obtained by exponential experiment.

where M^2 is a migration area and B_g^2 is a geometrical buckling in the vertical direction. For these lattices, M^2 was obtained as $31.5 \pm 0.5 \text{ cm}^2$ by using the measured reactivity coefficient of water level at a nearly critical state and the calculated effective delayed-neutron fraction β_{eff} of 0.7478%.¹ The value of B_g^2 was determined from the water level and the vertical reflector saving, $12.2 \pm 0.3 \text{ cm}$, which was obtained from a power distribution measurement. The resultant values of k_{eff} are shown in Fig. 2.

For comparison, k_{eff} 's were measured by a pulsed-neutron source technique on the same lattices as those of the exponential experiments. The prompt-neutron decay constant α is related to k_{eff} as

$$\alpha = [(1 - \beta_{\text{eff}})k_{\text{eff}} - 1]/\ell \quad (\text{Ref. 2}) \quad (2)$$

The prompt-neutron lifetime, ℓ , was obtained from the extrapolated value of α at $k_{\text{eff}} = 1$ and was found to be $(4.06 \pm 0.12) \times 10^{-5} \text{ s}$. Assuming ℓ as a constant, k_{eff} was evaluated by using Eq. (2). The results are shown in Fig. 2.

Criticality calculations were performed with the multi-group Monte Carlo code KENO IV.³ Neutron cross sections of 137 energy groups were prepared by using the ENDF/B-IV library data. The calculated results of k_{eff} after the trace of 60,000 neutron histories are also shown in Fig. 2. Although the simple one-group diffusion model was used for the exponential experiment, the results agree well with those of KENO IV in the wide range of k_{eff} from 0.94 to

Fig. 2. Comparison among experimental and calculational effective multiplication factors k_{eff} . Closed circles, open circles, and triangles represent KENO IV, exponential, and pulsed-neutron source results, respectively. N is of $n \times n$ square lattices.

0.45. On the other hand, the results of the pulsed-neutron source experiments show a marked tendency to saturate in the subcritical state of $k_{\text{eff}} < 0.95$. Since the saturation is due mainly to the variation of ℓ , a proper estimation of ℓ is required for this technique.

Another application of the exponential technique has been examined on the lattices loaded with blade-type absorbers. The fuel rods were arrayed in a 15×20 lattice, which was separated by an absorber into two 15×10 regions. The subcriticality was varied with the absorbers which had seven B_4C concentrations. In this case, the k_{eff} from the exponential experiments showed good agreement with those from the pulsed-neutron source experiments in the range of $0.97 > k_{\text{eff}} > 0.90$.

In conclusion, the exponential experiment is an effective technique to determine k_{eff} of a highly subcritical system, when M^2 of the system is known by some method and there is a sufficient region of uniform composition in at least one direction.

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SCALE System Cross-Section Validation for Criticality Safety Analysis, A. M. Hathout (Al-Azhar Univ-Egypt), R. M. Westfall (UCC-ND), H. L. Dodds, Jr. (Univ of Tenn)

The purpose of this study is to test selected data from three cross-section libraries for use in the criticality safety analysis of UO_2 fuel rod lattices. The libraries, which are distributed with the SCALE system,¹ are used to analyze potential criticality problems which could arise in the industrial fuel cycle for PWR and BWR reactors. Fuel lattice criticality problems could occur in pool storage, dry storage with accidental moderation, shearing and dissolution of irradiated elements, and in fuel transport and storage due to inadequate packing and shipping cask design. The data were tested by using the SCALE system to analyze 25 recently performed critical experiments.

The SCALE system consists of a driver module, control modules, functional modules, and a data base. One of the control modules is CSAS2, which stands for Criticality Safety Analytical Sequence No. 2. The CSAS2 control module reads a single unified set of input. After "messaging" the input and performing several auxiliary calculations (e.g., Dancoff factor determination), problem-dependent cross-section processing and a subsequent calculation of the system multiplication factor are performed. The execution path includes the functional modules BONAMI and NITAWL for problem-dependent cross-section processing, and the KENO-IV module for a Monte Carlo determination of the system multiplication factor.

The three cross-section libraries utilized in the analysis presented here are the 27-group CSRL library,² the 123-group GAM-THERMOS library,³ and the 16-group Hansen-Roach library.⁴ The 27-group library was collapsed from the 218-group CSRL library,⁵ which was derived from ENDF/B-IV data.⁶ Each of these libraries is available in the SCALE system data base.

Two series of experiments⁷ involving 4.75 wt% enriched UO_2 rods were analyzed. The rods had a fuel diameter of

0.79 cm, an active fuel length of 90 cm, and an AG5 aluminum cladding of 0.06 cm.

The first series of experiments utilized a square pitch lattice immersed in a sodium nitrate solution ($\text{NaNO}_3 + \text{H}_2\text{O}$). The parametric variations in this series were lattice pitch, NaNO_3 concentration, and the total number of rods in

nearly square arrays. Criticality was determined by varying the height of the sodium nitrate solution.

The second series utilized four assemblies of rods in an 18 X 18 square pitch lattice. Cross-shaped, watertight containers with various gap thicknesses (2.5, 5, and 10 cm) were positioned between the four assemblies. The containers

TABLE I
CSAS2 Analysis of 4.75% UO_2 Rods in Sodium Nitrate Solution

Lattice		Solution			Calculated $k_{\text{eff}} \pm 1\sigma$		
Lattice Pitch cm	Array Size	Density g/cm ³	NaNO_3 Conc. g/cm ³	Critical Height cm	27-Group ENDF/B-IV	123-Group GAM-THERMOS	16-Group Hansen-Roach
1.26	24x25	1.1521	0.247	80.9	0.99598 ± 0.00464	1.00401 ± 0.00477	0.98823 ± 0.00478
1.6	18x18	1.1503	0.243	76.1	0.98728 ± 0.00478	1.00216 ± 0.00455	0.99259 ± 0.00484
2.1	17x17	1.1523	0.246	76.2	0.99817 ± 0.00402	1.00846 ± 0.00379	1.01476 ± 0.00441
2.1	16x16	1.1094	0.178	89.5	1.00157 ± 0.00442	0.99509 ± 0.00430	1.01586 ± 0.00421
2.1	15x16	1.0565	0.090	84.8	1.00586 ± 0.00416	1.00658 ± 0.00389	1.0128 ± 0.00477
2.52	20x21	1.1523	0.246	85.9	0.99812 ± 0.00385	1.00239 ± 0.00418	1.01895 ± 0.00363

TABLE II
CSAS2 Analysis of Four Assemblies of 18 X 18 $\text{U}(4.75\%)$ Rods at 1.35-cm
Lattice Pitch, Separated by Various Hydrogenous Materials

Assembly Configuration					Calculated $k_{\text{eff}} \pm 1\sigma$		
Δ cm ³	Interstitial Material	Compounds		Water Critical Height cm	27-Group ENDF/B-IV	123-Group GAM-THERMOS	16-Group Hansen-Roach
		Density g/cm ³	Conc. H g/cm ³				
0	Water	1.0	0.1119	23.8	1.00806 ± 0.00507	1.02868 ± 0.00545	1.01481 ± 0.00514
2.5	Box + Air	0	0	29.03	1.01663 ± 0.00488	1.01591 ± 0.00476	1.00460 ± 0.00504
	Box + $(\text{C}_8\text{H}_8)_n$	0.0323	0.0025	28.61	1.00506 ± 0.00499	1.01608 ± 0.00491	0.99856 ± 0.00517
	Box + Powder $(\text{CH}_2)_n$	0.2879	0.0414	26.98	1.02348 ± 0.00472	1.02786 ± 0.00455	1.00797 ± 0.00471
	Box + Ball $(\text{CH}_2)_n$	0.5540	0.0800	25.54	1.03263 ± 0.00504	1.03542 ± 0.00457	1.02599 ± 0.00461
	Box + Water	1.0	0.1119	25.66	1.02827 ± 0.00454	1.03379 ± 0.00474	1.02991 ± 0.00485
	Water	1.0	0.1119	24.48	1.00391 ± 0.00399	1.02718 ± 0.00481	1.00673 ± 0.00376
5	Box + Air	0	0	34.48	0.99886 ± 0.00462	1.01276 ± 0.00495	1.00366 ± 0.00451
	Box + $(\text{C}_8\text{H}_8)_n$	0.0262	0.002	34.39	1.00614 ± 0.00537	1.01100 ± 0.00477	1.00696 ± 0.00432
	Box + Powder $(\text{CH}_2)_n$	0.3335	0.0480	30.16	1.02215 ± 0.00527	1.03381 ± 0.00412	1.0290 ± 0.00464
	Box + Balls $(\text{CH}_2)_n$	0.5796	0.0833	30.73	1.02913 ± 0.00446	1.02881 ± 0.00494	1.03299 ± 0.00519
	Box + Water	1.0	0.1119	32.78	1.02257 ± 0.00415	1.03349 ± 0.00463	1.02909 ± 0.00478
	Water	1.0	0.1119	31.47	0.99973 ± 0.00451	1.01139 ± 0.00473	1.01094 ± 0.00424
10	Box + Air	0	0	46.08	0.99138 ± 0.00441	1.01345 ± 0.00498	0.99519 ± 0.00537
	Box + $(\text{C}_8\text{H}_8)_n$	0.0288	0.0022	45.62	1.01184 ± 0.00488	1.01393 ± 0.00526	0.98867 ± 0.00579
	Box + Powder $(\text{CH}_2)_n$	0.3216	0.0464	42.05	1.01247 ± 0.00424	1.03046 ± 0.00454	1.02375 ± 0.00444
	Box + Balls $(\text{CH}_2)_n$	0.5680	0.0816	49.94	1.01674 ± 0.00450	1.00612 ± 0.00514	1.00898 ± 0.00499
	Box + Water	1.0	0.1119	64.12	0.99758 ± 0.00453	0.99949 ± 0.005	1.00396 ± 0.00512
	Water	1.0	0.1119	64.34	0.99352 ± 0.00446	0.98991 ± 0.00499	0.99659 ± 0.00521

^a Δ is the value of the gap width between the assemblies, including 0.6 cm which is twice the wall thickness of the aluminum container box.

were filled with either water, air, expanded polystyrene (C_6H_8)_n, polyethylene powder, or polyethylene balls, thus providing different concentrations of hydrogenous material. The parameters of interest in this series were the composition and density of the hydrogenous materials and the separation distance (i.e., gap thickness) between the assemblies. Criticality was determined by varying the lattice water height.

The results obtained with the SCALE system for the first series of experiments are presented in Table I. Using the 27-group ENDF/B-IV library, the average k_{eff} is 0.998 ± 0.004 ; using the 123-group GAM-THERMOS library, k_{eff} is 1.003 ± 0.004 ; and using the 16-group Hansen-Roach library, k_{eff} is 1.007 ± 0.004 . Of the three libraries, the 27- and the 123-group results are more accurate than the 16-group results. When both data accuracy and computing efficiency are considered, the 27-group ENDF/B-IV library is preferred because it has substantially smaller computer storage requirements than the 123-group library.

In Table II, results are presented for the second series of experiments. For assemblies with water only as the interstitial material, the 27-group library gives better agreement with experiment (i.e., measured $k_{eff} = \text{unity}$) than the other two libraries. For assemblies separated by the aluminum container filled with different materials, the calculated k_{eff} 's using all three libraries are, for the most part, equal to or slightly greater than unity. Also, considering statistical uncertainties, k_{eff} increases slightly with increasing hydrogen concentration for gap thicknesses of 2.5 and 5.0 cm, but not for the 10.0-cm gap. Finally, for this series of experiments, the three libraries give essentially the same values for k_{eff} . Therefore, for assemblies separated by the hydrogenous materials used in this work, either the 16-group Hansen-Roach or the 27-group ENDF/B-IV library would be preferred over the 123-group library because of their much smaller (approximate factor of 5) computer storage requirements.

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Effect of Heavy Metal Reflectors on the Criticality of Water-Flooded Fuel Rod Arrays, B. M. Durst (BNWL)

A research program for the U.S. Nuclear Regulatory Commission was started in 1976 to obtain experimental criticality data on systems simulating shipping casks for

spent-fuel assemblies. The objective of this program was to procure data that could be used to validate calculational methods used in criticality assessments of shipping containers for LWR-type fuels.

One phase of this research involved critical experiments with low-enriched uranium fuel rods with slabs of uranium or lead on two sides immersed in water. Rather interesting results were noted in this research.¹ Bierman observed that a lead or a depleted uranium wall (0.2 wt% ^{235}U) backed by water was a better reflector than water alone. Moreover, for the case of uranium shielding walls there was an optimum water gap spacing between fuel and wall that resulted in a more efficient reflector combination than if a closely fitting uranium wall were used. The lead wall did not show this effect but was most reactive when positioned at the fuel cell boundary. As a result of these findings, a calculational study was made to examine the effects on criticality of changing various parameters. These included distance from wall to fuel, lattice pitch, wall thickness, and fuel composition. The results of this study substantiate the experimental results found by Bierman, and further indicate that the effect of the metal reflector on criticality varies with all of the above-mentioned parameters.

The calculations were performed using the EGGNIT² code to generate 18 multigroup cross sections (17 epithermal, 1 thermal), from ENDF/B-IV data. The diffusion theory code, HFN,³ was then used to calculate critical infinite fuel slab thickness as the reflecting wall thickness was varied, and then as the water gap between fuel and wall was varied. This calculational model is shown in the upper corner of Fig. 1. A reflecting boundary condition was used on the right boundary. The results of calculations for UO_2 rods (4.3 wt% ^{235}U) in water reflected by depleted uranium are shown in Fig. 1. In these calculations, the fuel cell parameters were the same as used in Bierman's experiments, but now the wall thickness was varied from 0 to 300 mm of uranium. As the wall thickness was increased to 300 mm, the critical slab thickness continued to decrease. Also, the optimum water gap spacing for minimum critical slab thickness decreased from a thickness of 40-mm water (edge of fuel cell to wall) for a 10-mm-thick wall to a thickness of about 20 mm for a 300-mm-thick wall. At an 80-mm wall thickness, the wall thickness of uranium used in the experiments, the optimum gap is near 25 mm, consistent with Bierman's results.

The same effect occurred as the lattice pitch was reduced, but now the optimum water gap increased in value. With reference to Fig. 2, for a 10-mm-thick wall, the optimum gap was >40 mm, while a 300-mm-thick wall yielded a minimum critical slab thickness for a water gap of about 30 mm. Furthermore, the uranium wall had to be at least 100 mm thick to be a better reflector than a 150-mm-thick wall reflector alone. At near-optimum lattice spacing, 40-mm wall was more reactive than water. Another feature of interest present in Fig. 2 is an apparent crossover in critical slab thickness in the 0- to 5-mm water gap range. This implies that in this tight-pitched undermoderated lattice a 10-mm wall held tight to the fuel cell boundary is a better reflector than a 20-mm-thick wall. On the other hand, near-optimum moderation at a 25.4-mm pitch, a wall is always a better reflector as its thickness is increased (see Fig. 1).

These calculations were extended to include lattices of other type fuels and also to lead-reflected systems.

The results of this study show that in evaluating criticality of fuel elements in water it is important to consider not only the thickness of the metal reflector but also its distance from the fuel.

Fig. 1. Critical slab thickness versus uranium wall thickness and water gap to wall.

Fig. 2. Critical slab thickness versus uranium wall thickness and water gap to wall.

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Critical Dimensions for Arrays of $(\text{U,Pu})\text{O}_2$ Rods in Water, R. A. Libby, R. I. Scherpelz, E. D. Clayton (BNWL)

A series of calculations has been performed to determine the critical sizes of lattices of $(\text{U,Pu})\text{O}_2$ rods in water. The results obtained will complement the homogeneous data already available.¹ The calculations utilized the codes BRT-I and EGGNIT-II to generate macroscopic cross sections, and HFN to determine k_{eff} . This method of calculations has been verified by calculating k_{eff} for three different experimentally determined critical arrays of rods in water. Calculations have been completed for fuel rod material of three different compositions: 3 wt% Pu, 15 wt% Pu, and 30 wt% Pu in $(\text{U,Pu})\text{O}_2$. The calculations were performed over a wide variation of rod sizes and lattice pitches for each composition to determine the smallest critical arrays in three different geometries: an infinite slab, infinite cylinder, and a finite sphere. Calculations were then repeated to estimate sizes for arrays corresponding to $k_{\text{eff}} = 0.95$.

The analyses used the code EGGNIT-II² to generate 17-group macroscopic cross sections in the epithermal-neutron energies for a cell of fuel rod material and water, calculated with the cross sections for the single thermal group calculated with the code BRT-I (THERMOS).³ These two codes used cross-section data from the ENDF/B-III and ENDF/B-IV libraries, which were processed by FLANGE⁴, ETORG.⁵ The macroscopic cross sections were used in HFN,⁶

a one-dimensional diffusion theory code. Each HFN calculation was run as a critical search on the major dimension: the slab width, cylinder diameter, or spherical radius. HFN calculations were made for the rod size and pitch that yielded the maximum material buckling (B_m^2) in EGGNIT.

The calculational technique was verified for the 3 wt% Pu material by calculating k_{eff} for two arrays of $(\text{U,Pu})\text{O}_2$ rods in water which were determined to be critical in an experiment by Uotinen et al.⁷ In one array there were 152.1 rods of 12.83-mm diam positioned on triangular lattice pitch of 26.67 mm. The PuO_2 content of the mixed oxide was 2 wt%. The calculated k_{eff} was 1.018. The second array was composed of 180.9 rods of 12.64-mm diam in a triangular lattice of 40.64-mm lattice pitch, wherein the PuO_2 content of the mixed oxide was 4 wt%. For this case, k_{eff} was calculated as 1.013. Thus these calculations slightly underestimate the critical dimensions; i.e., the calculated dimensions are smaller than those measured.

Verification of the calculations involving 15 wt% Pu and 30 wt% Pu was obtained by calculating k_{eff} for a critical array experimentally determined by Bierman et al.⁸ The fuel pins in this experiment were composed of 19.84 wt% Pu in $(\text{U,Pu})\text{O}_2$, with the plutonium containing 11.5 wt% ^{240}Pu . The fuel material had a diameter of 4.94 mm, was clad in Type 316 stainless steel, and the rods were arranged in a 9.677-mm square lattice. The calculated value of k_{eff} was 1.023, which also indicates that these calculations give conservative dimensions.

The fuel rods modeled in the calculations presented herein were somewhat idealized, with the fuel material consisting of only the isotopes ^{239}Pu , ^{235}U , ^{238}U , and ^{16}O . The uranium contained 0.711 wt% ^{235}U . Densities of the fuel material were assumed to be 10.98 kg/litre for 3 wt% Pu, 11.04 for 15 wt% Pu, and 11.11 for 30 wt% Pu. The rod material was assumed to have no cladding around it. A 300-mm water reflector surrounded the arrays of rods.

Figures 1 through 3 show the calculated minimum critical parameters. Note that homogeneous values calculated by Clark and by Chalmers are also included.¹ Comments will be provided on the effect that higher isotopes of plutonium have on the critical parameters presented.

Fig. 1. Minimum critical slab thickness.

Fig. 2. Minimum critical cylinder diameter.

Fig. 3. Minimum critical sphere volume.

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Criticality of Moderated and Undermoderated Low-Enriched Uranium Oxide Systems, George R. Goebel (Rockwell Int)

Critical experiments, sponsored by the U.S. Nuclear Regulatory Commission, were performed at the Critical Mass Laboratory at Rockwell International's Rocky Flats Plant in Golden, Colorado, to provide accurate criticality data for low-enriched damp uranium oxide (U_3O_8) systems.

The experimental critical parameters provide a means of validating calculational methods to assure criticality safety in the nuclear industry.

Thirty-two critical experiments were performed. Arrays of cans, each containing ~15.2 kg of uranium oxide, were assembled with and without methyl methacrylate plastic moderator material placed between the cans. A 5 X 5 X 5 array of uranium oxide cans was used when no plastic moderator material was placed between cans. Since this 5 X 5 X 5 array alone is not critical, even fully reflected, high-enriched (~93% ^{235}U) uranium drivers were used to achieve criticality.

Criticality was achieved for smaller arrays without a driver when the plastic moderator material was placed between the cans.¹ In some moderated experiments, thin sheets of either polyvinyl chloride (PVC) or mild steel were placed between cans for neutron absorption.

The uranium oxide used in these experiments was enriched to 4.46 wt% ^{235}U and compacted to a density of 4.68 g/cm³. Approximately 15.2 kg of uranium oxide was packed into 1.6-mm-thick cubical aluminum cans, 152 mm on a side. Water was added to the oxide to obtain an H/U atomic ratio of 0.77 (Ref. 2).

Criticality data for these experiments were obtained for three different reflector shells. The minimally reflected case used 6.35-mm-thick steel plate to contain the oxide cans. The other two reflectors were thick-walled (~254-mm) cubical shells composed of concrete or methyl methacrylate plastic.

All experiments were performed remotely on a horizontal split-table machine. Each half of the table supported a portion of the oxide cans and one part of a reflector shell. The separation between the two portions of the table, supporting the oxide cans and reflector, was decreased until criticality was achieved and a long positive period was obtained. The table was then opened slightly until the array was slightly subcritical and a long negative period was obtained. Interpolation of this period data gives the table separation at critical.

Twelve critical experiments were performed using a 5 X 5 X 5 array of oxide cans with no plastic moderator material between cans. This array required a high-enriched (~93% ^{235}U) uranium driver to achieve criticality. The drivers used were an enriched uranium metal sphere (density 18.76 g/cm³), high concentration (351.2 g U/litre, density 1.489 g/cm³) uranyl nitrate solution, and a low concentration (86.4 g U/litre, density 1.120 g/cm³) uranyl nitrate solution. Critical parameters are given in Table I. Under the three reflection conditions, the critical core separation ranged from 6.3 to 18.2 mm for the following oxide and driver masses: the mass of the uranium metal driver ranged from 29.87 to 33.54 kg for an oxide mass of 1864.6 kg; the mass of the high concentration solution driver ranged from 14.07 to 16.14 kg for an oxide mass of 1824.9 kg; and the mass of the low concentration solution driver ranged from 12.4 to 14 kg for an oxide mass of 1824.9 kg.

Twenty critical experiments using the plastic and concrete reflectors had 9.2-, 12.4-, or 24.3-mm-thick plastic moderator material between cans. These arrays achieved criticality without the necessity of a driver. Some experiments also had polyvinyl chloride or mild-steel neutron absorber material between cans. Critical parameters are given in Table I. The critical core separation varied between 2.3 and 18.4 mm. Critical oxide weights varied from 603.5 to 1508.8 kg.

These critical experiments provide data on moderated and undermoderated low-enriched uranium oxide systems

TABLE I
Critical Parameters of Plastic- and Concrete-Reflected
Low-Enriched Uranium Oxide Systems at H/U = 0.77

Oxide Weight (kg)	Driver Type	Driver Mass (kg)	Reflector Type	Moderator Plastic Thickness (mm)	Neutron Absorber	Critical Separation (mm)
1824.9	Solution 86.4 g U/t	12.875	Plastic	---	---	9.12
1824.9	Solution 86.4 g U/t	13.001	Plastic	---	---	11.17
1824.9	Solution 351.2 g U/t	14.844	Plastic	---	---	9.42
1864.6	Metal 93.12% Enriched	29.870	Plastic	---	---	8.01
1864.6	Metal 93.12% Enriched	29.870	Plastic	---	---	8.26
1824.9	Solution 86.4 g U/t	12.446	Concrete	---	---	8.64
1824.9	Solution 351.2 g U/t	14.066	Concrete	---	---	6.68
1864.6	Metal 93.12% Enriched	29.870	Concrete	---	---	12.60
1864.6	Metal 93.12% Enriched	29.870	Concrete	---	---	18.16
1824.9	Solution 86.4 g U/t	13.999	Steel	---	---	6.30
1824.9	Solution 351.2 g U/t	16.143	Steel	---	---	9.04
1824.9	Metal 93.12% Enriched	33.543	Steel	---	---	14.64
633.7	---	---	Plastic	24.3	---	3.06
648.8	---	---	Plastic	24.3	---	15.17
769.5	---	---	Plastic	24.3	0.60 Steel	2.36
844.9	---	---	Plastic	24.3	0.54 PVC	4.24
905.3	---	---	Plastic	24.3	1.17 Steel	2.34
935.5	---	---	Plastic	24.3	1.17 Steel	12.83
935.5 ^a	---	---	Plastic	24.3	---	5.33
965.6 ^b	---	---	Plastic	24.3	---	11.27
1116.5 ^c	---	---	Plastic	24.3	---	6.79
1508.8	---	---	Plastic	9.2	---	18.37
1508.8	---	---	Plastic	9.2	---	10.47
1508.8	---	---	Plastic	12.4	1.17 Steel	7.04
603.5	---	---	Concrete	24.3	---	5.67
603.5	---	---	Concrete	24.3	---	6.17
799.7	---	---	Concrete	24.3	0.54 PVC	10.47
875.1	---	---	Concrete	24.3	1.17 Steel	2.51
890.2	---	---	Concrete	24.3	1.17 Steel	11.84
1448.5 ^d	---	---	Concrete	24.3	---	5.83
1478.6	---	---	Concrete	9.2	---	4.84
1508.8	---	---	Concrete	9.2	---	11.87

^aA 12.7-mm air gap existed between adjacent plastic moderator pieces.

^bA 12.7-mm air gap existed between adjacent plastic moderator pieces in the horizontal direction at right angles to the direction of table motion. A 19.0-mm air gap existed between adjacent plastic moderator pieces in the vertical direction and in the horizontal direction parallel to table motion.

^cPlastic moderator was placed between 2 x 1 x 2 arrays of cans, not between individual cans.

^dPlastic moderator was placed between 2 x 2 x 2 arrays of cans on one half of the table and 2 x 3 x 2 arrays of cans on the other half of the table, not between individual cans.

which can be used to validate calculational methods to assure criticality safety. The details of this work are to be published in NRC topical reports.

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Calculated K_{eff} for Partially Concrete-Reflected FTR Pins in Water, R. T. Primm, III, J. F. Mincey (ORNL)

This paper presents the results of a calculational study of four critical experiments. The objective of the study was to calculate K_{eff} for fast test reactor (FTR) pin arrays moderated by water and partially reflected by concrete. Calculations using the same methods for critical, water-reflected and -moderated FTR pin arrays show good agreement with experiment. However, some discrepancy is shown to exist between calculated and experimental K_{eff} values for these four partially concrete-reflected pin arrays.

During September 1978, four critical experiments involving fast test reactor fuel pins immersed in water and partially reflected with concrete were performed at Battelle Pacific Northwest Laboratory (PNL). Documentation of the experiments has been published elsewhere.¹ Figure 1 shows the axial configuration of the pins used in the experiments. In all of the experiments, the pin arrays were surrounded with 0.40 m of concrete. Approximately 0.02 m of water was between the outermost pins and the concrete reflector.

Representations of these experiments were prepared for the KENO-IV² Monte Carlo computer code. Cross sections were supplied from the standard 27-energy-group subset of the Criticality Safety Reference Library³ (CSRL) and processed by the NITAWL⁴ module of the AMPX-II cross-section processing system. The 27-group subset of the CSRL is a P_3 library. However, the version of KENO-IV used in these calculations computed K using only the P_0 and P_1 portions of the scattering matrix. All axial and radial zones shown in Fig. 1 were included in the KENO-IV representation.

Table I shows the calculated K_{eff} values for the four experiments. Using the mean values for pin pitch and distance between outer row of pins and concrete reflector, the calculated K_{eff} 's are 1.5 to 2.0% high. This discrepancy was not seen in any of the calculations of water-reflected pin arrays, which are also reported in Table I for ease of comparison.

Since calculations using mean values of the pitch and the water gap between array and reflector yielded K_{eff} 's significantly greater than 1, the dependence of K_{eff} on variations in these experimental parameters was explored. Changes in the parameters which would lead to a reduction in K_{eff} were examined. Durst et al.^{5,6} have reported that none of the four lattice pitches was overmoderated. Calculations performed using KENO-IV, 27-group CSRL cross sections and confirmed by experimental evidence at PNL have shown concrete to be a better reflector than water for these water-moderated pin arrays. Considering these two observations, a series of calculations was conducted in which the water gap surrounding the array was increased by the reported uncertainty and the pitch of the array decreased by the reported uncertainty.⁵ The results of this sensitivity study are also shown in Table I. Taking into account statistical uncertainties in the K_{eff} calculation, no improvement in the calculated K_{eff} 's is evident.

TABLE I
Calculated K_{eff} for FTR Pins in Water

Reflector	Experimental Parameters ^{1,5}			Calculated K_{eff} ^a	
	Pitch (10^{-3} m)	Critical number of pins	Lattice width (pins)	Mean values of pitch + gap	Pitch - $1\sigma^b$ gap + $1\sigma^b$
Concrete	9.530	554	28	1.019	1.013
Concrete	12.640	260	18	1.018	1.013
Concrete	15.410	191	18	1.017	1.018
Concrete	19.060	152	14	1.028	1.022
Water	7.671	1,037	36	0.994	-
Water	9.525	605	28	0.996	-
Water	12.588	279	18	0.999	-
Water	15.342	205	18	1.000	-
Water	19.050	162	14	1.006	-

^a1 Standard deviation ≤ 0.003 .

^b σ = Quoted statistical uncertainty in the experimental parameter.

^aREFERENCED DIVISION OF REACTOR DEVELOPMENT AND TECHNOLOGY STANDARD

^b²⁴¹Am CONTENT OF TOTAL PLUTONIUM WAS MEASURED TO BE 0.13 wt% DURING APRIL AND MAY OF 1972 EXPERIMENTS PERFORMED SEPTEMBER 1978

^cALL GRID PLATES ARE 13.7 ± 0.1 mm THICK

Fig. 1. Axial configuration of concrete-reflected experiments.

Calculations presented in this paper indicate that a K_{eff} bias exists in the KENO-IV/CSRL representation of these experiments. Considering the calculations of the water-reflected FTR pin array experiments, the demonstrated bias in these four experiments appears to be related to the partial

concrete reflection. Possible explanations include the inadequate calculation of the concrete-reflected neutron spectra due to truncating the P_2 and P_3 components of the 27-group library and/or the change in the absorption rate of neutrons when the water reflector is replaced with concrete.

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Criticality Safety, Safeguards, and Security

Potential Impacts of Material Safeguards on Nuclear Criticality Safety, R. A. Knief* (GPU Serv Corp)

Increased concern over the potential diversion of nuclear materials has led to an escalation in materials safeguards requirements nationally and internationally. As with any externally imposed regulations, these may be expected to impact operations and nuclear safety. This paper identifies major concerns as they relate to the conduct of nuclear criticality safety.

BASES FOR SAFEGUARDS

As currently employed, "material safeguards" is a somewhat vague term representing a wide range of measures used to assure that nuclear materials are not diverted for unauthorized uses. Domestic and international concerns are fundamentally different and, thus, require separate methods.

Domestic safeguards directed toward subnational terrorist groups have the following constituents: material accountancy, material control, and physical protection. It has been recognized that none of the three is capable by itself of preventing theft by insiders and/or outsiders. However, by complementing each other the risks may be reduced greatly.

International safeguards are directed toward limiting the spread of nuclear weapons, or proliferation, among the nations of the world. Under the terms of the Non-Proliferation Treaty (NPT) and various other agreements, the International Atomic Energy Agency (IAEA) is charged with assuring that commercial nuclear technology does not result in proliferation.

IAEA safeguards are to be employed "in a manner designed to avoid hampering a state's economic and technological development" and "to be consistent with prudent management practices required for the economic and safe conduct of nuclear activities."¹ Under these guidelines, verification of the state's system of nuclear material accountancy has become the safeguards measure of fundamental importance. Use of devices and equipment of surveillance and verification of containment serve to augment the accountancy.

POTENTIAL IMPACTS ON NUCLEAR CRITICALITY SAFETY

Each of the facets of safeguards has the potential for impacting operations and nuclear safety in a different manner. The following observations are based on tours and discussions at a number of different types of cycle facilities.

Material accountancy appears to have the potentially most positive effect on criticality safety since knowledge of quantities and locations of fissile materials is very important to both. When accountancy requires changes in process locations and flows, however, some disruption may occur. A specific example is a mixed-oxide fuel fabrication line with a geometrically favorable design. Proposed accountancy measures restricted throughput to the extent that dual lines or substantial design changes (i.e., nonfavorable geometry) might be required.

Material control can also be a positive impact on criticality safety in providing specific, secured locations for materials at times they are not actively in process. Detailed information on quantities and locations augments the accountancy function. Required changes in long-established "natural and workable" procedures, however, could result in short-term criticality safety problems.

Physical protection is generally applied around the periphery of a facility and at interfaces between major operations. As the safeguards measure which is least integral with operations, it also appears to have the greatest potential for adverse impact. Hands-on search has led to a decline in morale and occasional loss of highly experienced personnel. Safety's requirement for ready evacuation of personnel and security's requirement for restricted egress produce another major conflict. Additional concern exists around security's desire to keep fissile material in confined locations, while criticality safety has a preference for geometric separation.

A weapons country like the United States will implement international safeguards only as a diplomatic expedient to encourage other nations to accept IAEA inspection. Although not yet in effect here, the safeguards may be expected to affect criticality safety mainly through process changes which may be required to assure that inventories can be verified independently by IAEA personnel. The presence of an on-site inspector and the use of containment and surveillance methods will also perturb the overall operations.

The most significant impact of safeguards on criticality safety appears to be related to the rate at which Federal safeguards regulations are escalating. The often short lead time on compliance rarely allows for adequate assessment of net effects on operations and safety.

1. "The Agency's Safeguards System 1965, as Provisionally Extended in 1966 and 1968," INFCIRC/66/rev. 2, International Atomic Energy Agency (Sep. 16, 1968).

International Safeguards for Fast Critical Facilities, P. I. Amundson, S. B. Brumbach (ANL-Idaho), D. D. Cobb, J. L. Sapir* (LASL)

Fast critical facilities pose a difficult problem for the application of international safeguards. The inventory of nuclear material typically consists of thousands of small fuel pieces which together contain large quantities of plutonium (Pu) and highly enriched uranium (HEU). The International Atomic Energy Agency (IAEA) must verify that the nuclear material inventory is as declared by the facility operator. The IAEA safeguards goal is to detect a diversion of 8-kg Pu or 25-kg ²³⁵U in HEU with detection times of 1 to 3 weeks for "abrupt" diversion and of one year for protracted diversion. The IAEA inventory verification must provide independent assurance of the amount of fissile material present without unnecessarily duplicating the facility operator's material accounting and control measures, without unduly disrupting facility operations, and without compromising health and safety, including criticality safety.

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The IAEA safeguards system relies on materials accounting reports provided by the state, on accounting and operating records maintained at the facility, and on inspections for independent verification. The inspections include nondestructive assay (NDA) measurements on the facility's fissile materials. The problem is to make measurements on a statistically significant fraction of the inventory within reasonable limits of manpower, time, and facility disruption. If fuel assemblies and storage canisters, rather than fuel pieces, can be verified, or if fuel in the assembly core can be verified collectively, then considerable savings in measurement time, effort, and impact can be realized. Because safeguards-oriented NDA measurements normally require fuel handling and fuel movement and may involve the presence of moderator material, it is essential that criticality safety factors be analyzed for any measurements.

Inventory verification systems for fast critical facilities, including a variety of NDA methods, have been considered.¹ Table I lists measurement techniques that could be used for inventory verification and gives comments on the capabilities and limitations of the techniques. The choice of techniques depends on the design and operating features of the facility. The gamma-ray and neutron measurements are currently used by international inspectors at critical facilities to verify fuel pieces, fuel assemblies, and storage canisters on a random sampling basis. Autoradiographic methods are currently in use at ANL fast critical facilities.² In most fast critical assemblies, the moderator in film and film packaging materials will have a positive reactivity effect when placed inside the assembly core. For example, x-ray film had a measured reactivity worth of 103 lh/kg when randomly distributed through one quadrant of one-half of the ZPR-9 assembly.

TABLE I
Material Measurement Techniques

Measurement Type	Capabilities	Limitations
Gamma spectroscopy	Isotopic ratios; fissile Pu enrichment.	Significant self-attenuation; affected by gamma background; marginal for HEU fuels; fuel handling required.
Passive neutron Total counts	Content of even Pu isotopes; relatively simple instrumentation.	Fissile Pu isotopes not measured directly; affected by (α ,n) reactions; affected by neutron background; fuel handling required; no HEU fuels.
Passive neutron Total and coincidence counts	Content of even Pu isotopes; sensitive to changes in geometry; corrections for background and (α ,n) reactions.	Fissile Pu isotopes not measured directly; fuel handling required; no HEU fuels.
Active neutron	²³⁵ U content of HEU fuels; total fissile content of mixed (U,Pu) fuels; can be operated in passive mode for Pu fuels.	External radiation source required; relatively bulky and complex instrumentation; fuel handling required.
Passive Autoradiography	Image of edge of each fuel element; simpler for Pu than HEU; very fast; minimum fuel handling.	Surface effect; poor gamma-ray energy resolution; limited ability to discriminate between isotopes.
Active Autoradiography	Image production depends on presence of fissile isotopes and on critical mass; simpler for HEU than Pu.	Small sample size in some applications due to positive reactivity effect of film.
Reactivity	Total in-reactor inventory; very sensitive.	Difficult for inspector to verify independently; supplementary measurements required; some fuel handling required.
Material worth	Reactivity-compensating changes in fuel density, neutron spectrum, and power profile.	Supplementary to reactivity measurements.
Foil activation	Reactivity-compensating changes in fuel density, neutron spectrum, and power profile. Needed to supplement reactivity measurement.	Limited sample size because of counting requirements and short half-lives.

Consequently, it is important to analyze the criticality safety implications of having fuel in an assembly core, especially if the assembly is to be brought to a critical or near-critical configuration.

A series of experiments was recently conducted to evaluate the feasibility of reactivity measurements and foil activation methods for safeguards applications. Reactivity measurements are attractive because small changes in the fissile material content of an assembly core can result in large changes in reactivity. The sensitivity of reactivity measurements to fissile removal is determined by the reactivity worth of the fissile material and by the uncertainty in the reactivity measurement. Reactivity worth of fissile isotopes was measured as a function of location in the assembly. For a given amount of fissile material, the reactivity of a critical assembly also depends on the core geometry and on the neutron energy spectrum. Therefore, core geometry and neutron energy spectrum must also be verified for a reactivity verification measurement to be meaningful. For safeguards purposes it may prove most convenient to make reactivity measurements with the assembly in some reference configuration whose reactivity and fissile material content have been previously ascertained.

Changes in reactivity were measured after removing ~2 kg of fissile material from various regions of a subcritical reference core of assembly 10 of the zero-power plutonium reactor (ZPPR). Reactivity was determined by a source multiplication method utilizing 64 in-core fission counters to monitor the neutron flux. In one case, a 2-kg fissile removal localized near the core center (high fissile worth region) resulted in a 0.16\$ reactivity decrease. The uncertainty in the measurement was estimated to be <0.01\$. Another 2-kg fissile removal distributed along the outer core-blanket interface (low fissile worth region) resulted in a reactivity decrease of 0.02 ± 0.01 \$.

A foil irradiation method was evaluated for its ability to verify the neutron energy spectrum in the assembly. In this method a spectral index was measured using indium foil. The spectral index was the ratio of the rate of the $^{115}\text{In}(n, \gamma)$ reaction (sensitive to low-energy neutrons) to the rate of the $^{115}\text{In}(n, n')$ reaction (sensitive to high-energy neutrons). The ratio of the gamma-ray intensities from the products of these reactions comprise the experimental data. This spectral index was able to detect the presence of a 0.64-cm-thick, 5.08-cm-high polyethylene column 10 to 15 cm away from the indium foil.

1. D. D. COBB et al., "Concepts for Inventory Verification in Critical Facilities," LA-7315, Los Alamos Scientific Lab. (Dec. 1978).
2. S. B. BRUMBACH and P. I. AMUNDSON, "Fast Critical Assembly Safeguards," ANL-80-13, Vol. I, Argonne National Lab. (Feb. 1980).

Criticality and Safeguards at the Idaho Chemical Processing Plant, G. P. Kodman (IAEA), R. E. Wilson (Exxon Nucl Idaho)

Reprocessing of highly enriched irradiated reactor fuel at the Idaho Chemical Processing Plant (ICPP) presents significant problems to the Criticality Safety (CS) and Safeguards & Security (S&S) Sections. Two major interactions between these sections occur when irradiated fuel is stored and fuel is dissolved.

S&S is assigned the responsibility of maintaining a centralized records and reporting system which provides detailed, timely knowledge of the location, quantity, and measurement

uncertainties associated with accountable nuclear material, including uranium and plutonium. The CS Section uses this information in providing criticality safety evaluations with support analyses, inspection, field surveillance, and audits to ensure criticality safety implementation. The interaction of these sections has minimized operational constraints and maximized criticality and safeguards controls.

FUEL STORAGE

The irradiated fuel storage basin consists of three rectangular 6.1-m-deep swimming pools, designated the North, Middle, and South Basins. The North and Middle Basins use a monorail system of 29 rows X 19 columns with fuel being placed in numbered buckets prior to storage. Buckets between rows can be freely accessed but column access is on a last-in, first-out (LIFO) basis. The South Basin consists of several discrete racks. The size and location of the rack depends on the type of fuel being stored.

The spent reactor fuel is received into the Basin only after assuring compatibility with the proposed storage container material (a stainless steel or aluminum bucket or rack), and a criticality safety evaluation has shown it can be safely stored.

Once the fuel is received, S&S is responsible for receipt verification and periodic physical inventory of the material. During a recent S&S inventory audit, pieces of fuel were observed in unapproved storage configurations. Operations in the storage facility were immediately suspended until an evaluation demonstrated criticality safety and an additional CS compliance audit of all stored fuel was completed.

A dedicated computer system operated by S&S maintains all plant accountability records. In addition to Department of Energy (DOE) required information, such as location, description, and postirradiated uranium/plutonium values, S&S maintains the preirradiated values and gross weights of the fuel. This information is readily available on the S&S computer system interactive mode. Typical searches performed to assist in CS audits include row-by-row descriptions, contents of a single bucket, and the location of all empty buckets.

FUEL DISSOLUTION

Processing campaigns are scheduled by operations and range from 400 to 1000 kg ^{235}U . The Safeguards and Security Section is responsible for preparing the batching schedule for each campaign. A detailed review of the entire Basin inventory is made and process compatible fuels with longer Basin residence time are given first priority on the schedule.

The fuel is transferred from the storage Basin on a batch basis to one of the process area dissolvers. Acid is added and the solution containing the dissolved fuel is pumped to the downstream accountability tanks prior to the next transfer from the Basin.

Use of dissolvers not critically safe by geometry requires conservative estimates of the dissolver heel. As fuel is charged to the dissolver, the as-built (beginning-of-life or manufacturer's) values as well as the after-burnup ^{235}U content are considered in criticality evaluations. Use of the as-built values accounts for the possibility that an unirradiated element with no poison enters the process. The after-burnup values and the input accountability measurements produce an estimate of the amount of material remaining in the dissolver. The uncertainty associated with the difference between the as-built values and the after-burnup values guides the application of the dissolver limits. An example is the zirconium dissolver, which has the following restrictions:

1. The total amount of ^{235}U charged to the dissolver in a single batch must not exceed 13 kg.

2. The total amount of uranium cannot exceed 1.6% by weight.

The final batching schedule is several similar schedules with output tailored to each section's needs. A generalized schedule used by Operations and Storage Basin personnel includes descriptions, locations, bucket numbers, and uranium content. A more detailed schedule confirms batch compliance with dissolver limits and includes the statistical uncertainties for each batch. An additional list of items not processed allows "do not process" tagouts, minimizing potential operator error.

UNIRRADIATED FUEL

A major criticality safety concern is the fissile material without fission fragments which is stored in essentially unshielded areas.

These areas are the most secure areas in the plant since the material is attractive to a potential diverter (thief).

During the semi-annual physical inventory conducted by S&S, a detailed hands-on inventory and NDA verification program necessitates extensive fuel handling. This additional handling in unshielded areas increases the criticality risk for the plant but also provides assurance that fuel is being properly controlled. The S&S and CS interface maintains this delicate balance.

CONCLUSION

The safeguards and security-dedicated computer system at the ICP provides storage basin and operations personnel with a detailed processing list, criticality safety with additional administrative controls, and S&S with a means of monitoring process activities. In addition to providing criticality controls, various operational plans can be readily evaluated. The computerized system has been used for the 1978 and 1980 dissolution campaigns and has proved an invaluable aid to plant operation.

The information generated during these dissolution campaigns provides additional input for future dissolution head-ends. These dissolvers will have criticality safety-derived limits on both charge and heel determined by S&S accountability measurements and the shippers end-of-life values.

Fissile Material Control: Pitfalls and Promises, H. A. Worle (EG&G Idaho)

At EG&G Idaho, three separate organizations have responsibility for overview of handling and storing fissile materials: the Criticality Safety Section within the Safety Division, the Safeguards Section and the Physical Security Section within the Safeguards and Security Division. These groups have somewhat differing goals and use different means to achieve them. The operating organizations are charged with compliance with the regulations promulgated by all three overview groups. This paper addresses some of the problems and successes experienced at EG&G with the interfaces between these groups.

All three overview groups impose restrictions on operating organizations for handling and storing fissile materials. Grossly oversimplifying, one could say that Criticality Safety does not want too much fissile material in any one location, Safeguards does not want too little (unexplained shortages), and Security would like to keep all the material out of unauthorized hands. Three types of area designations are used to control use of and access to fissile materials. Criticality Safety uses the concept of a Fissile Material Control Area (FMCA), defined as any area (a building or part thereof) where more than 15 g of fissile material is stored or handled.

Each FMCA has its own mass and other types of limits. Inventory records are required to maintain mass limits. Safeguards has established Material Balance Areas (MBAs) which may involve several FMCAs as long as the materials do not change custody when transferred between them. Security establishes Material Access Areas (MAAs) which are secured by a fence or similar barrier. The level of access control for an MAA is related to the type and amount of fissile material allowed inside the area. Again, one MAA may, and often does, contain several MBAs.

Conflicts can arise when all the regulations are not considered. Each of the overview groups usually exerts some pressure on the operating organization for compliance with their particular rules. Sometimes this can lead to confusion, causing mistakes, as the following example shows. Fissile material was transferred from a less secure to a more secure location, because the first area was not authorized, from a Security point of view, for uranium of greater than 20% enrichment. The material was moved to its new location and the lid to the storage position tack welded, although an alert Health Physics technician informed the operators involved that the location was not an approved FMCA. The *ex post facto* analysis found the new location quite safe; the violation resulted in severe repercussions.

In another case, the custodian for a hot cell is required to maintain two sets of inventory records. Safeguards regulations require the fissile inventory of some irradiated power reactor fuel elements to be as close to reality as possible by calculating burnup and estimating Pu generated through conversion. The Criticality Safety inventory is intended to be conservatively large, lest incorrect burnup information should lead to low inventory estimates. Therefore, the Criticality Safety log lists "as-fabricated" fissile masses.

At other times, annoying and sometimes costly delays occur in the startup of new operations because, after all Criticality Safety considerations have been dealt with, it is discovered that the proposed facility would be in an MAA that is not authorized for the proposed total amounts of material or for the range of enrichments contemplated.

For all these pitfalls, there are some areas of present or potential cooperation between the three overview groups. A proposed computerized inventory system for the Safeguards Section can be used to reduce the probability of exceeding Criticality Safety limits. Another area of cooperation is training. Because nuclear materials custodians (in charge of MBAs) and fissile material custodians (in charge of FMCAs) are often the same individuals, training sessions can be coordinated and more efficiently conducted. Also, Safeguards assays help to reconfirm Criticality Safety fissile material inventory records.

Much good could come from coordination between the organizations which promulgate policy and regulations for the three concerns at the national level.

Criticality Parameters of 55-gal Waste Drum Arrays, W. A. Blyckert, R. D. Carter (Rockwell Hanford)

During the course of handling fissile material, various wastes contaminated by the fissile material may be generated in significant quantities. This material may be stored prior to reclamation or buried in retrievable storage. To safely store this waste, a study has been made to determine the various parameters that maximize the k-effective of two- and three-dimensionally infinite arrays of 55-gal drums containing plutonium wastes. The results show some significant increases in k-effective compared to k-infinity calculations, which

assume the fissile material is spread uniformly over the entire drum volume.

The calculations were made using the KENO IV Monte Carlo Code¹ with cross sections generated by the GAMTEC II code.² These calculations include the effect of density changes in the contained material at constant grams per drum and hydrogen-to-plutonium (H/Pu) atomic ratios. Initial parameter studies were for wastes simulated as ²³⁹Pu-H₂O mixtures. Two different shapes of fissile material in the drum were selected. For one shape the fissile material was assumed to have the same height-to-diameter (H/D) ratio as the drum (about 1.5), while for the other the fissile material was assumed to extend to the diameter of drum with the height varying inversely as the fissile density (the "flattened" core). In the majority of the calculations, the drum was assumed to have an iron mass equivalent to a 17H drum and the drums were assumed to be in a square array. The height was assumed to be the nominal drum height, but the radius was reduced so the array spacing would be equivalent to a triangular array.

A series of calculations was made for 200 g of plutonium per drum for an infinite array, and for 2-tier and 6-tier arrays which were infinite in the horizontal dimensions. The finite dimension was assumed to be reflected on both top and bottom by 2 ft of soil. The space between the drums was assumed to be a void since this was found to result in a higher k-effective than when soil was interspersed. Each array was at the H/Pu ratio that gave the highest k-effective and the H/Pu ratio was held constant for all volume fractions.

The results of the calculations are shown in Fig. 1 for both flattened and H/D of 1.5 cores. The k-effective for the lowest density material (100% of the drum volume occupied) is at far right. As the plutonium density is increased (the remaining volume is void), the k-effective increases until full Pu-H₂O density is obtained (minimum volume fraction). The increase for the H/D of 1.5 system is quite significant with Δk 's of up to 0.4 greater at the smallest volume fraction than that of the "full drum" arrays. The flattened cores show a much smaller change with a maximum Δk of about 0.17. There appears to be little difference in reactivity between core shapes at greater than 30% of full volume as the flattened core comes closer to the shape of the H/D = 1.5 core. The 6-tier array at full density has a higher k-effective than the infinite full drum array.

If only the core material is considered, then the k-effective (k-infinity) of the infinite array would be the same for the same H/Pu ratio no matter what the core density or shape. However, with the addition of the drum walls another effect occurs. As the material densities increase, or as the H/D of the fissile material comes closer to 1.0, less neutron leakage occurs and the k-effective of the unit increases. In addition, because of the reduced neutron leakage, a smaller fraction of the neutrons is available for absorption in the iron drum walls. This also increases k-effective. It is this latter effect that makes simplified k-infinity calculations nonconservative if significant densification of the fissile material in the waste can occur and is not taken into account.

Fig. 1. 55-gal waste drum in infinite horizontal arrays.

A corollary to this statement is that the iron mass of the drum has a strong effect on k -effective. The k -effective for arrays of the 17C drums may be as much as 0.08 less than that for arrays of lighter 17H drums.

Other calculations show that for full-density, flattened cores, infinite horizontal arrays will be subcritical for vertical dimensions of 1 tier with 400 g/drum, 2 tiers with 300 g/drum, 3 tiers with 250 g/drum, and 6 tiers with 200 g/drum. These numbers, however, include no allowances for errors such as overbatching of individual drums or overstacking of arrays.

The study shows that it can be unsafe to make simplistic assumptions for the density and shape of contained fissile

material and the mass and composition of the enclosing container. The data may be used to provide insight in determining "worst-case" bases for criticality safety analysis. Actual conditions, such as other moderators or different containers, might require more stringent limitations than indicated by this study.

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Physics of Thermal Reactors

Homogeneous Thermal Benchmark Calculations with ENDF/B-V Data Using MCNP,
Richard E. Prael (LASL)

A number of thermal reactor benchmark calculations¹ have been performed for $^{235}\text{U}\text{-H}_2\text{O}$ and $^{239}\text{Pu}\text{-H}_2\text{O}$ homogeneous assemblies using the continuous-energy Monte Carlo code MCNP² with ENDF/B-V data. The purposes of the calculations were to contribute to the CSWEG thermal data testing for ENDF/B-V and to test the cross-section libraries prepared for use by MCNP from ENDF/B-V data with the NJOY nuclear data processing system.³ The results of MCNP calculations using ENDF/B-IV data have been reported elsewhere,^{4,5} as have some infinite lattice calculations using ENDF/B-V data.⁵

The $S(\alpha, \beta)$ scattering data were used for hydrogen scattering below 4 eV, with a short collision time approximation at an effective temperature of 1396.8 K to extend the data over the full range. Scattering by hydrogen from 10 to 4 eV, and by other light isotopes below 10 eV, utilized a free gas scattering model at 300 K. Cross sections for resonant materials were Doppler broadened to 300 K and represented on a very detailed energy grid. Details of MCNP calculational methods have been reported elsewhere.^{2,4}

The calculational results for the $^{235}\text{U}\text{-H}_2\text{O}$ systems ORNL-1 through -4 and ORNL-10 are shown in Table I. δ_{25} is the ratio of epithermal-to-thermal ^{235}U fission. Results for the $^{239}\text{Pu}\text{-H}_2\text{O}$ systems are shown in Table II. Included are results for the infinite slab and infinite medium calculations from Ref. 6. In Table II, δ_{49} is the ratio of epithermal-to-thermal ^{239}Pu fission and ρ_{40} is the ratio of epithermal-to-thermal ^{240}Pu capture. The net leakage shown in Table II is defined as leakage from the *fuel* region. The uncertainties shown represent an estimated one standard deviation. Uncertainties in ratios are approximated by assuming no correlation between numerator and denominator. Typically, the calculations represent the result of 200,000 neutron histories each. In addition, for PNL-8, results for both the simplified (PNL8A) and complex (PNL8B) models are reported.

An examination of Table I shows that, for the spheres of uranyl nitrate solution, neither δ_{25} nor the eigenvalues are greatly changed in going from ENDF/B-IV to -V data. However, the trend toward increased leakage is quite noticeable, from near zero for ORNL-2 to 6% for the large sphere ORNL-10. Except for ORNL-2, increased leakage approximately compensates for the 0.74% increase in ^{235}U thermal ν . An indication of the shift in spectrum in going from Version IV to Version V data is found in the ~2.8% increase in mean source energy for ORNL-1 through -4 and 2.5% for ORNL-10.

For the plutonium nitrate solution critical experiments shown in Table II, we again see a significant trend toward increased leakage, with the largest (~+2%) increases for PNL-8 and PNL-10, and no statistically significant increase for PNL-11. It should be noted that the mean source energy for these assemblies increases about 1.7% in going from Version IV to Version V. The epithermal-to-thermal fission ratio for ^{239}Pu is not changed significantly. However, the

TABLE I
Homogeneous ^{235}U Critical Experiments Calculated with ENDF/B-V Data

Assembly	Leakage	δ_{25}	k
ORNL-1	.1764 ± .0008 (+.0029 ± .0012)	.02141 ± .00007 (-.00002 ± .00010)	1.0025 ± .0019 (+.0023 ± .0027)
ORNL-2	.1719 ± .0008 (+.0002 ± .0012)	.02501 ± .00009 (-.00016 ± .00013)	1.0020 ± .0023 (+.0060 ± .0029)
ORNL-3	.1706 ± .0008 (+.0036 ± .0012)	.02862 ± .00010 (+.00017 ± .00014)	.9972 ± .0022 (-.0025 ± .0030)
ORNL-4	.1688 ± .0008 (+.0013 ± .0012)	.03020 ± .00010 (-.00030 ± .00014)	1.0016 ± .0022 (+.0023 ± .0028)
ORNL-10	.0714 ± .0006 (+.0041 ± .0008)	.01739 ± .00005 (-.00005 ± .00017)	.9977 ± .0020 (+.0006 ± .0026)

(changes from ENDF/B-IV results shown in parentheses)

TABLE II
Homogeneous ^{239}Pu Critical Experiments Calculated with ENDF/B-V Data

	Net Leakage	δ_{49}	ρ_{40}	k
PNL6B	.3869 ± .0011 (+.0023 ± .0016)	.1959 ± .0010 (+.0005 ± .0014)	9.97 ± .10 (-.20 ± .14)	1.0118 ± .0024 (+.0009 ± .0031)
PNL10	.3871 ± .0037 (+.0074 ± .0045)	.1045 ± .0005 (-.0007 ± .0007)	5.11 ± .05 (-.04 ± .07)	1.0075 ± .0023 (-.0044 ± .0033)
PNL11	.1387 ± .0026 (-.0029 ± .0036)	.0306 ± .0001 (-.0001 ± .0002)	1.09 ± .01 (-.01 ± .01)	1.0098 ± .0022 (+.0084 ± .0029)
PNL8A	.3607 ± .0011 (+.0080 ± .0016)	.0360 ± .0002 (+.0003 ± .0002)	2.60 ± .03 (-.07 ± .04)	1.0153 ± .0023 (-.0071 ± .0031)
PNL9	.3049 ± .0011 (+.0044 ± .0016)	.0289 ± .0001 (+.0000 ± .0002)	1.72 ± .02 (-.09 ± .02)	1.0115 ± .0023 (+.0014 ± .0033)
PNL7B	.3540 ± .0032 (+.0034 ± .0040)	.0272 ± .0001 (+.0001 ± .0002)	2.20 ± .02 (-.13 ± .04)	1.0129 ± .0022 (-.0004 ± .0032)
PNL12B	.3255 ± .0035 (+.0036 ± .0050)	.0251 ± .0001 (+.0000 ± .0002)	1.87 ± .02 (-.09 ± .03)	1.0073 ± .0020 (-.0019 ± .0030)
1D SLAB	.3833 ± .0011 (+.0037 ± .0015)	.0606 ± .0003 (-.0014 ± .0004)	4.05 ± .04 (-.16 ± .06)	1.0225 ± .0018 (+.0009 ± .0028)
INF MED	—	.0113 ± .0001 (-.0001 ± .0001)	—	0.9999 ± .0014 (+.0061 ± .0021)
PNL5	.3717 ± .0011 (+.0044 ± .0015)	.0476 ± .0002 (-.0002 ± .0003)	3.40 ± .04 (-.12 ± .05)	1.0110 ± .0024 (-.0010 ± .0033)
PNL1	.3700 ± .0011 (+.0027 ± .0015)	.0398 ± .0002 (-.0002 ± .0003)	2.81 ± .03 (-.15 ± .04)	1.0211 ± .0021 (+.0041 ± .0030)
PNL4	.3252 ± .0010 (+.0015 ± .0014)	.0329 ± .0001 (+.0003 ± .0002)	2.46 ± .03 (-.06 ± .04)	1.0072 ± .0023 (+.0016 ± .0033)
PNL3	.3173 ± .0010 (+.0038 ± .0014)	.0254 ± .0001 (-.0001 ± .0002)	1.91 ± .02 (-.14 ± .03)	1.0003 ± .0023 (-.0004 ± .0033)
PNL8B	.3619 ± .0010 (+.0062 ± .0015)	.0358 ± .0002 (-.0002 ± .0002)	2.61 ± .03 (-.06 ± .04)	1.0161 ± .0021 (-.0003 ± .0030)

(changes from ENDF/B-IV results shown in parentheses)

decrease in ^{240}Pu epithermal capture is quite significant in most of the calculations. The increase in ^{239}Pu thermal ν is reflected in the infinite-medium calculations. Apart from PNL-10, PNL-8, and PNL-11, the observed changes in eigenvalue are not statistically significant; overall, there is no change in the tendency for the calculated eigenvalues for these benchmarks to be 1 to 2% high.

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TRIGA Subcritical Experiments in Support of the k_{∞} Meter, H. Ocampo, S. H. Levine, M. A. Schultz (Penn State Univ)

The objective of this program is to develop a device to measure the k_{∞} of a light-water reactor spent-fuel assembly prior to actually placing the element into a fuel storage rack. Levine et al.¹ described a k_{∞} meter designed to fulfill this objective. The design, however, was based on analytical calculations and it becomes necessary to verify the calculational methods by performing subcritical experiments using the Penn State Breazeale TRIGA Reactor (PSBR) and a ^{252}Cf source. The purpose of this paper is to report the results of these subcritical experiments.

The k_{∞} meter is a cross-shaped subcritical assembly into which a spent-fuel assembly is placed and its k_{∞} determined by the change it produces in the multiplication of the subcritical assembly. Experiments have been performed with the PSBR to verify the multiplication characteristics of a cross-shaped subcritical assembly, as shown in Fig. 1, which is similar to the one proposed in the k_{∞} meter. Changes in the source multiplication as observed by fission counters are compared with the nucleonic code calculations used to design the k_{∞} meter. Multiplication experiments are also performed using a rectangular-shaped assembly. The rectangle has the approximate dimensions of two PWR fuel assemblies. Only relative count rates are considered in these measurements because only relative count rates are used by the k_{∞} meter.

Fig. 1. Cross-shaped fuel assembly using TRIGA fuel elements.

TABLE I
Subcritical Multiplication of Configurations with No Poisons

For Detector #W-7671 in Position B-9					
Conf. No.	EXT-II k-eff	Experimental Results		EXT-II Ratio (X/Y)	% Error EXP/EXT-II
		Counts/s	Ratio (X/Y)		
17-B	0.934	5615	1.0000	1.0000	0.0
17-C	0.940	6464	1.1512	1.1404	0.94
17-D	0.947	7183	1.2794	1.2769	0.19
17-E	0.954	8145	1.4507	1.5240	- 5.05
17-F	0.965	10358	1.8448	2.0356	-10.34
17-G	0.968	11270	2.0071	2.2610	-12.65
14	0.976	13097	2.3326	2.6317	-12.83
8	0.995	46012	8.1943	15.6681	-91.21
Configurations with Control Rods					
2-A	0.979	13797	2.4572	2.6638	- 8.41
5A	0.968	9507	1.6931	1.6984	- 0.31
8A	0.953	6228	1.1092	1.0743	3.15
2B	0.942	5960	1.0614	0.9946	6.30
5-B	0.931	4966	0.8844	0.8150	7.85
8-B	0.915	3839	0.6837	0.6189	9.47
2-C	0.993	41830	7.4496	10.6942	-43.55
5-C	0.980	20091	3.5780	3.6390	- 1.71
8-C	0.962	10495	1.8691	1.7471	6.53
Configurations with Special Poison Rods					
15	0.939	5654	1.0071	0.9854	2.15
16	0.932	5136	0.9148	0.8965	2.00
17	0.925	4761	0.8480	0.8243	2.80
17A	0.899	3627	0.6460	0.6297	2.52
Configurations with Both Type of Rods					
15-A	0.897	2760	0.4917	0.4606	6.34
15-B	0.859	2165	0.3857	0.3536	8.32
15-C	0.907	4001	0.7127	0.6704	5.93

Fission chambers D-1 and D-2 are inserted into positions B-9 and E-12 (see Fig. 1) and the neutron source, S, is placed in the central thimble, CT. The CT is at position H-9, and the control rods SA, RR, TR, and SH are at positions F-9, H-12, J-9, and H-6, respectively. All control rods, except TR, have fuel followers; hence, the center line of symmetry intersects the centers of the central thimble, SA, and TR.

The degree of subcriticality is varied by replacing 12 wt% U-Zr fuel elements, each containing ~54.5 g of ^{235}U , with 8.5 wt% U-Zr containing ~38 g of ^{235}U , or vice versa. In this manner, the change of k_{∞} in the central region of the fuel assembly is simulated to compare with the operational characteristics of the k_{∞} meter. Highly subcritical assemblies are attained by inserting control rods in symmetrical patterns. Four special poison rods, each 15 in. long (the height of fuel in a TRIGA fuel element) held in aluminum tubes of the size of the TRIGA fuel elements are inserted into the core to further reduce its reactivity. The special rods are used in core positions F-7, F-11, J-7, and J-11. A total of 24 subcritical configurations have been analyzed.

Two sets of measurements were performed. One set used supercritical configurations for calibration purposes. The second set of measurements consisted of measuring count rates of the fission detectors for the various subcritical configurations.

The analytical methods employed the LEOPARD and EXTERMINATOR-2 codes to compare with the experimental data. The calculations are performed in the same way as used to design the k_{∞} meter. However, half-core symmetry is employed in the TRIGA calculations, whereas quarter-core symmetry is used in the k_{∞} -meter calculations. Measurements from one configuration of each of the different type control rods were used to determine the equivalent poison for these components.

Table I provides the experimental data together with the analytical results. In general, good agreement exists between the calculations and experimental data. The large error obtained for configuration 8 is caused, in part, by an error in the reactivity calculation and by losses in counting rates for the "dead time" of the electronic system. Extremely small errors in the k_{eff} value produce large errors in the calculated count rate when $k_{\text{eff}} > 0.993$; however, the experimental and analytical results generally agree to within 10% after fine tuning the calibration to 0.002 $\Delta k/k$.

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Data and Analysis for Nuclear Criticality Safety

Preliminary Investigation of ^{252}Cf -Driven Neutron Noise Analysis for Subcritical Fuel Solution Systems, J. T. Mihalcz, R. C. Kryter, W. T. King (ORNL)

A method for determining the reactivity of highly subcritical systems of fissile material, using neutron-noise power spectral densities in conjunction with a ^{252}Cf source, had previously been tested in two fast reactor critical assemblies (a mockup of the Fast Flux Test Facility reactor and unreflected enriched uranium metal assemblies¹) and one thermal reactor (a light water moderated and reflected lattice of Oak Ridge Research Reactor fuel elements^{2,3}). The last-mentioned test demonstrated the effectiveness of the method in water-moderated systems and thereby prompted the present study of its application to facilities for fuel preparation, reprocessing, and storage.

An important advantage of the method is that only known or measured properties of the fissile system at the subcritical state of interest are required to interpret the measured quantities. Alternative methods for inferring reactivity usually require calibration at the delayed critical condition, which may not be possible or desirable in many situations. In addition to the neutron multiplication factor k , the ^{252}Cf -driven method inherently yields additional useful properties of the fissile system, such as the prompt neutron decay constant and the inverse neutron count rate, which is related to the source neutron multiplication.

To investigate the applicability of this method to facilities for fuel preparation, reprocessing, and storage, limited experiments were performed with a uranyl fluoride solution. The Los Alamos National Laboratory SHEBA facility,⁴ an unreflected cylindrical tank (56 cm diam), was partially filled with a solution containing 5 wt% ^{235}U -enriched uranium [$\rho(\text{U}) \sim 1 \text{ g/cm}^3$, H/U atomic ratio ~ 550]. The tank had a central 5-cm-diam axial hole in which the ^{252}Cf source was placed at the midpoint of the solution. Measurements were performed at various solution heights from 60% of the critical height to near-critical height ($\sim 36 \text{ cm}$).

The noise analysis method employed requires simultaneous measurement of cross- and auto-power spectral densities (CPSDs and APSDs) between the fluctuating components of the signals from three detectors, so that a ratio of spectral densities, $(G_{12}^* G_{13}) / (G_{11} G_{23})$, that is directly proportional to $\Delta k/k$ can be formed.¹ The measured CPSDs G_{12} and G_{13} between the signals from two ^6Li -loaded glass scintillators (detectors 2 and 3) placed on the tank's outer periphery (120 deg apart and at the solution midheight) and the signal from the centrally located ionization chamber containing the ^{252}Cf source (detector 1), the CPSD G_{23} between the signals from the two external detectors, and the APSD G_{11} of the source were combined to form this spectral density ratio, which is plotted in Fig. 1 as a function of fuel solution height. Also plotted is the inverse count rate⁵ from an additional detector used to monitor the initial critical loading. The obvious nonlinear relationship between the inverse count rate and solution height results from variations in detection efficiency

(produced, in turn, by variations in neutron scattering) that accompany changes in the fuel solution height. Conversely, since the ratio of spectral densities has been shown to be independent of detection efficiency,¹ it is not subject to distortion as the fuel solution height varies, and in the case shown, this ratio plots linearly with the solution height. In principle, detection efficiency corrections to the inverse count rate data, which would improve the linearity of its relationship with solution height, could be obtained experimentally; however, in most practical applications, such correction is not feasible. Hence, Fig. 1 clearly indicates the superior capability of the spectral density ratio to predict criticality in tanks containing fuel solutions by extrapolating from data obtained at far-subcritical states.

A test in which the two exterior ^6Li -loaded detectors were placed adjacent to each other rather than 120 deg apart was performed to investigate spatial effects. Initial evaluation of the results indicates that such placement of the detectors may result in preferential sampling of short fission chains that occur immediately in front of the detectors, thereby yielding a distorted measurement of G_{23} . Placing the detectors 120 deg apart on the tank surface appears preferable, since this permits the behavior of long fission chains to be correlated, with the result that global (rather than localized) system properties are measured.

The authors believe that these preliminary investigations

Fig. 1. Comparison of spectral density ratios and inverse count rates obtained during an approach-to-critical experiment.

show the ^{252}Cf -driven noise measurement to be a promising technique for on-line monitoring of the subcriticality of fuel solution tanks in operating facilities, primarily because a measurement precision of 5% was attained in ~ 30 min when the SHEBA tank was filled to $\sim 70\%$ of critical height. However, the final desired quantity $\Delta k/k$ has not yet been obtained, since the system parameters needed to establish the proportionality constant between the ratio of spectral densities and $\Delta k/k$ were not measured experimentally (due to facility time limitations) and therefore will have to be calculated. Furthermore, the lowest neutron multiplication factor measurable by this technique was not established in the experiments reported. This practical limit, as well as the establishment of optimum detector locations, clearly needs further experimental investigation with a variety of fuel solutions and geometries.

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Critical Experiments Supporting High-Density Fuel Pin Storage, Gary S. Hoovler, M. Neil Baldwin, Ray L. Eng (B&W)

This paper describes a series of critical experiments designed to simulate high-density storage of spent fuel pins removed from the fuel bundle. The Babcock & Wilcox Company (B&W) has been performing critical experiments in support of interim spent fuel storage since 1977. This work has been sponsored by the U.S. Department of Energy (DOE) as part of the Away-From-Reactor (AFR) Fuel Storage Program. These critical experiments are being performed to validate calculational methods used to compute the neutron multiplication of high-density, underwater storage configurations. The results are applicable to both at-reactor and AFR fuel storage.

The critical experiment program can be divided into two major work elements. In the first, a critical array of light water reactor (LWR)-type fuel assemblies was studied.^{1,2} These experiments were designed to simulate high-density storage of LWR fuel assemblies. That work has been completed, and the results have been published.^{3,4}

This paper describes the second work element. In the second work element, which is now under way, the critical assemblies consist of a 5×5 array of specially designed modules containing low-enriched ($2\frac{1}{2}\%$) urania fuel. The fuel pins in each module are more tightly packed than in a conventional LWR fuel assembly. These experiments are designed to simulate fuel pin storage. Pin storage requires the disassembly of spent fuel bundles and the repacking of individual pins into specially designed canisters. A study performed by the Nuclear Assurance Corporation for the DOE concluded that pin storage should be implemented at any

Fig. 1. Plan view of 5×5 module array.

storage basin fixed in size when the need for additional capacity becomes urgent.⁵

Critical experiments simulating pin storage are being performed with modules having three different packing densities. Arranged in order of decreasing density, they are:

1. triangular pitch equal to fuel pin o.d. (pins touching); 15×17 fuel pin matrix each module; intermodule spacing 1.78 to 3.81 cm
2. square pitch equal to fuel pin o.d. (pins touching); 15×15 fuel pin matrix each module; intermodule spacing of 1.78 cm
3. square pitch 16.8% greater than fuel pin o.d.; 13×13 fuel pin matrix each module; intermodule spacing of 1.78 cm.

The framework for each fuel module consists of a top and bottom end fitting (aluminum) fastened to four solid aluminum rods, each of which occupies a corner position. For those modules whose pin pitch equals the pin diameter (items 1 and 2 above), the intermediate pin-to-pin spacing is maintained by two aluminum bands that girdle each fuel cluster. The intermediate pin spacing of the third module type (item 3) is maintained by a single aluminum grid plate located at the axial center of the module.

As already mentioned, each critical assembly consists of a 5×5 array of fuel modules (Fig. 1). The intermodule spacing is uniform for each core, and varies from 1.78 to 3.81 cm for the five cores studied. The fuel modules are $\sim 17.8 \text{ cm}^2$ and 156 cm long. The fuel pins comprising the modules are clad with aluminum and have an outer diameter of 1.207 cm. Boric acid dissolved in the moderator is used for shim reactivity control to make each core critical at full water height.

These experiments and a set of comparative calculations using the KENO IV code are expected to be completed in March 1981. The calculations will be performed by F. G. Welfare and E. L. Maceda of B&W, and will be presented with the final experimental results. These results will provide a new body of benchmark data with which to validate criticality codes used to design high-density pin storage racks.

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Cross-Section Validation for Fuel Storage Pools, P. A. Landeyro, F. Siciliano, Tayyab Abbas (CNEN-Italy)

Nuclear criticality safety in the fuel reprocessing industry demands reliable criticality studies of process cells, fissile material storage, fuel element transportation, and fuel storage pools. The Monte Carlo codes such as KENO II, KENO IV, and MORSE C.G. are commonly used for this purpose. Validation studies¹ of KENO IV and MORSE C.G. codes show that these codes, with 16-broad-group Hansen-Roach cross sections, give acceptable results for criticality calculations for many applications. For other applications more accurate calculations may be needed. Assuming that the required accuracy in the results could be achieved using fine multigroup cross sections, a validation study of an AMPX-II 123-group cross-section library for criticality assessments of fuel storage pools was taken up at COMB-CNEN, Italy.

The objective of this paper is to report on calculations of the benchmark critical experiments regarding the critical separation between subcritical clusters of 2.35 wt% ²³⁵U-enriched UO₂ rods in water with and without fixed neutron absorbers. These calculations will be used for validating a 123-fine-energy-group cross-section library for analysis for fuel storage pools.

A detailed description of fuel rod, cladding, neutron absorber plates, and the experimental system of the selected cases (identified as experiment numbers 014, 017, 020, 028, 032, 050, and 052) is given in Ref. 2. The calculations were performed with the KENO IV code employing a 123-fine-energy-group cross-section library and 16-broad-group Hansen-Roach cross sections. The Master Interface of 123-multigroup library (including 91 epithermal and 32 thermal groups) was generated from ENDF/B-IV data by XLACS2 (XLACS for hydrogen) module of the AMPX-II system³ and was subsequently processed by the NITAWL module for obtaining the cross sections in AMPX working format. All experiments were calculated using both the discrete fuel rod and smeared rod models with 109 batches and 500 neutrons per batch, skipping the initial 3 batches. In the discrete model, the individual rods, including cladding, interstitial water, and aluminum end pieces, are described. In the smeared model, the cluster is taken as a homogeneous mixture of fuel, cladding, and water. The homogenized unit cell average 123-group cross sections produced by the AMPX-II system XSDRNPM module were used in the homogenized cluster region. The homogenized unit cell averaged 16-group cross sections were produced by ANISN.

The results of discrete and smeared model calculations are summarized in Tables I and II, respectively.

The 123-fine-group discrete and smeared model calculations give average multiplication factors of 0.997 and 0.993, respectively. This shows that the discrete and smeared model results are in good agreement within the 0.003 statistical error limits. The 16-broad-group discrete and smeared model

TABLE I
Results of Discrete Model Calculations*

Experiment ID.	123-Group AMPX-II		16-Group Hansen-Roach	
	k _{eff}	SIGMA	k _{eff}	SIGMA
014 ^a	0.99564	+ 0.00303	1.01802	+ 0.00319
017 ^b	0.99666	+ 0.00329	1.02624	+ 0.00300
020	0.99368	+ 0.00310	1.02392	+ 0.00298
028	0.99961	+ 0.00304	1.01913	+ 0.00302
032	1.00180	+ 0.00326	1.01857	+ 0.00291
036	0.99622	+ 0.00291	1.02537	+ 0.00285
050	0.99291	+ 0.00392	1.02253	+ 0.00284
052	0.99960	+ 0.00317	1.02130	+ 0.00282

*Average k_{eff} 123-group 0.99701, average k_{eff} 16-group 1.02188.

^a103 batches input to KENO IV.

^bCentral cluster at 20 × 16 rods. Two outer clusters at 22 × 16 rods each.

TABLE II
Results of Smeared Rod Model Calculations*

Experiment ID.	123-Group AMPX-II		16-Group Hansen-Roach	
	keff	Sigma	keff	Sigma
014	0.98887	±0.00305	1.01945	±0.00300
017 ^a	1.0000	±0.00294	1.02110	±0.00305
020	0.99146	±0.00288	1.01409	±0.00283
028	0.99482	±0.00285	1.02461	±0.00274
032	0.99188	±0.00275	1.02482	±0.00286
036	0.98892	±0.00317	1.02261	±0.00261
050	0.99259	±0.00310	1.02374	±0.00258
052	0.99768	±0.00276	1.02341	±0.00313

*Average keff 123-group = 0.99328, average keff 16-group = 1.02173.

^aCentral cluster at 20 X 16 rods. Two outer clusters at 22 X 16 rods each.

calculations give average multiplication factors of 1.022 and 0.996, respectively.

The discrete model average multiplication factor value obtained using 123-fine-group cross sections is lower than the critical value by ~0.3% (in terms of keff) whereas 16-broad-group discrete model calculations give an average keff higher than the critical value of 2.2%.

The discrete model calculations using 18-broad-group cross sections indicate a conservative bias of ~2% in terms of keff (see Ref. 2). From this comparison, it appears that the criticality calculations with the KENO IV code using 123-fine-group cross sections give more accurate results for the systems simulating fuel storage pools.

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Nuclear Criticality Safety Analyses for the Waste Isolation Pilot Plant Project, M. H. Lipner (Westinghouse-NTD), J. M. Ravets (Westinghouse-AESD)

This paper presents the results of the nuclear criticality analyses performed in support of the Waste Isolation Pilot Plant (WIPP) Project Safety Analysis Report.¹ Criticality safety encompassing initial waste emplacement was considered. The mission of the WIPP facility is twofold: a research and development facility to (a) demonstrate the safe disposal of radioactive wastes resulting from U.S. defense programs, and (b) provide an area where experiments with defense high-level wastes can be conducted. Both these functions will be performed in a mined bedded salt medium located ~2150 ft below grade. To fulfill this mission, the facility is designed for receipt and inspection of wastes con-

tained in protective packages, preparation for underground storage, transportation to the salt bed storage area, and placement into storage.

A small volumetric fraction of the transuranic (TRU) waste generates a surface dose rate exceeding 200 mrem/h. This waste, designated as remote handled (RH) TRU waste, will be received for storage at WIPP contained in metal canisters placed within licensed shipping containers. One form of RH-TRU waste is not processed and may be termed nonfixed. This waste may consist of contaminated concrete and steel, dry process sludges, etc. The fixed waste forms will include borosilicate glass and wastes fixed in ordinary concrete.

Contact handled (CH) wastes are defined as those packages that have a surface dose rate <200 mrem/h. The waste planned for storage consists of material packaged in U.S. Department of Transportation (DOT)-specification² drums and boxes, which will arrive at the site packaged in licensed shipping containers. The materials packaged in the drums and boxes are both combustible and noncombustible.

Reactivity calculations were done using the ANISN-W multigroup, discrete ordinate, transport theory code.^{3,4} The AMPX modular code system,⁵ developed by the Oak Ridge National Laboratory, was used to generate the 27-group cross sections for use by ANISN-W. Cross sections were generated from the Vitamin C 171 fine-group library.⁶

To provide validation of the calculational model, two critical assemblies were analyzed with the 27-energy-group cross-section library and the ANISN-W code. Both a thermal⁷ and a fast⁸ system were studied to ensure the accuracy of the calculations over a wide range of conditions. The calculated eigenvalue (keff) for the benchmark thermal critical system was 1.016, and was 0.987 for the fast system.

The analytical approach taken for the RH wastes was to model conservatively an infinite slab containing waste forms. Only the plutonium isotopic content was averaged over the volume of the canister to obtain the heavy metal atom densities for use in the calculations. This is conservative in that the steel canister and other parasitic neutron absorbers such as strontium and cobalt were omitted. The results indicate that the RH wastes are safely subcritical in both the normal dry configuration and the flooded accident case.

The CH wastes are in a less fixed (more loosely packed) form than the RH wastes, which would therefore allow more of the intermixture with any available moderator; hence, the

Fig. 1. Infinite multiplication factor as a function of H/²³⁹Pu for CH wastes in 17C 55-gal drums.

effect of interspersed low-density moderation was investigated. The 17C 55-gal drum was chosen to be studied because it is a typical CH waste container and the allowable fissile content per unit volume is highest among the nonreinforced containers. The ANISN model included the plutonium isotopes in the waste and a conservative estimate of the hydrogenous combustible material interspersed uniformly in the waste. For additional conservatism, the steel drum thickness was taken as the minimum required for 16-gauge steel.² The 90-mil rigid polyethylene drum liner was also modeled explicitly. The plutonium was modeled as being homogeneously distributed throughout the waste.

The effect on multiplication factor of these fissile loadings (typical, average, and maximum) was examined by varying the hydrogen-to-²³⁹Pu ratio for an infinite array of DOT-specification 17C steel drums. Figure 1 shows the results of the calculations. For any value of H/²³⁹Pu, the typical, average, and maximum fissile content conditions do not lead to critical configurations. It is interesting to note that the array of canisters is actually more reactive in the normal dry operating condition, when the drums contain only the hydrogen from the combustibles. As more moderation is added by, for example, accidental flooding, the multiplication factor decreases, since the configuration is substantially overmoderated. Accidental waste compaction by repository crushing also would not increase the multiplication factor, since only the atom ratios affect the outcome of these infinite media calculations.

A nuclear criticality safety analysis for the WIPP project was performed. The handling and storage of RH and CH TRU wastes were considered. Based on experimentally verified calculations utilizing the ANISN-W computer code employing cross sections generated using the AMPX modular code system, it was determined that neither normal operating conditions nor any credible accident could result in achieving a critical configuration.

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Criticality Safety of High-Level Waste Containing Unseparated Plutonium, K. R. Yates (BCL), R. N. Christensen (Ohio State Univ)

The potential for diversion of separated plutonium during recycle operations continues to be perceived as a major route

to proliferation. One method of reducing this perceived risk is to recycle only uranium and to send plutonium to the high-level waste (HLW) stream. It should be understood that the authors are not advocating such a fuel cycle. Nevertheless, until the final decisions are made regarding reprocessing, waste disposal, and safeguards, one scenario that must be considered is the disposal of HLW containing unseparated plutonium. Therefore, a study was undertaken to examine the criticality safety of geologic disposal of HLW containing unseparated plutonium. Previous studies have shown that compacted calcine containing unseparated plutonium will have a $K_{eff} > 1$ for the waste form configuration postulated for space disposal of HLW.¹

The specific HLW mix considered was the PW-4b mix, developed by the Pacific Northwest Laboratory,² modified by assuming no plutonium separation. This mix corresponds to HLW from an optimized reprocessing plant and contains no sodium or gadolinium and little iron. It also assumes 99.9% uranium separation. Therefore, if no plutonium separation is assumed, the PW-4b mix will result in the highest plutonium concentration of any waste mix. This modified PW-4b mix was then assumed to be incorporated into three waste forms: borosilicate glass, cermet, and calcine. Borosilicate glass is the leading waste form candidate; cermet is representative of metal matrix waste forms; and calcine is the waste form having the highest waste loading. Borosilicate glass was assumed to have 32 wt%, cermet 58.7 wt%, and calcine 100 wt% PW-4b waste loading. All HLW containers were assumed to be 20.32 cm i.d., with a 0.5-cm stainless steel wall.

Waste canisters containing the glass, cermet, or calcine waste mixes postulated above will be individually safe. However, groups of waste containers in close contact may not be nuclearly safe.

Fig. 1. K_{eff} versus number of cermet or glass units.

Fig. 2. K_{eff} versus number of calcine units.

Loss of HLW container spacing may occur in a salt repository by brine intrusion with subsequent salt dissolution. Loss of spacing could result in hundreds of HLW containers in contact. The criticality safety implications of such an event were examined using the KENO code³ with 16-group Hansen-Roach cross sections. Hansen-Roach cross sections were not available for neptunium, americium, or curium. Therefore, ^{242}Pu was substituted for ^{237}Np , since their fission cross sections and fast critical masses are essentially identical. Similarly, ^{235}U was substituted for americium and curium.

Several KENO calculations were run for borosilicate glass and cermet containing the modified PW-4b mix. The effects of square-pitched array size and interstitial water are shown in Fig. 1. Interstitial water can be seen to lessen K_{eff} . The cermet data indicate that its 58.7 wt% waste loading results in an unsafe condition for the events postulated. Reflection has not been specifically addressed in this study because in the repository environment, it is difficult to imagine signifi-

Fig. 3. K_{eff} versus relative plutonium concentration in calcine.

cant reflection (by salt or water) without interstitial water also. Salt itself is a poor reflector.

Similar calculations were also run for compacted calcine-100% modified PW-4b waste mix. The effect of square-pitched array size is shown in Fig. 2.

Finally, the effect of reduced plutonium concentration was investigated for square-pitched arrays of calcine. Figure 3 presents the results. It appears that calcine can safely accommodate ~70% of the unseparated plutonium concentration.

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Administrative Practices and Emergency Systems for Nuclear Criticality Safety

Administrative Practices for Criticality Safety at the Hot Fuel Examination Facility, A. S. Garcia, J. C. Courtney,* F. L. DiLorenzo, J. P. Bacca (ANL-Idaho)

The Hot Fuel Examination Facility (HFEF) complex located at Argonne National Laboratory's site near Idaho Falls, Ida., consists of two major hot cell facilities: HFEF/North and HFEF/South. Both are used for postirradiation handling and examination of U.S. Breeder Reactor Program fuels and materials and fast reactor safety experiments.¹⁻³

Overall responsibility for implementation of the nuclear criticality safety program at HFEF is assigned by the HFEF manager to a single staff member for fuels management. Operations systems engineers and chief technicians (first-line supervisors) have direct supervisory responsibilities for specific work areas and equipment items in the facilities. Actual operations in each criticality control zone are performed only by certified fissile material handlers. Lists of those currently authorized individuals to move nuclear fuel are prominently displayed throughout the facilities. Personnel are certified only after they have demonstrated a thorough understanding of operational and safety procedures. Successful completion of a formal program covering the basics of criticality safety and specific rules governing the handling, transfer, and storage of fissile material as applicable to HFEF is required.⁴ Tests covering the nuclear criticality theory and HFEF specific rules must be passed by the individuals as a prerequisite to entering on-the-job training, and at least six months of practical experience is required before full certification is granted. An additional oral examination is required annually for the chief technicians. Question areas include emergency actions and interpersonal relations, as well as criticality safety theory and specific rules and regulations. All personnel must be recertified annually.

Operations involving nuclear fuels are carried out in specified criticality-control zones; zone boundaries in the hot cells are clearly defined by floors or walls of the hot cells and by floor markers that are distinguished by yellow and black stripes. Signs are posted to identify individual zones, responsible staff persons, and appropriate rules and material limitations. Tag boards indicate the fuel and moderator items in each zone. Details of every transfer of fissile material into or out of a zone are entered in an inventory record. The staff member assigned as the responsible individual for each criticality control zone must approve material transfers into his zone. Audits to confirm the location of fissile materials and inventories are conducted daily or at least every other working day. An independent verification of all records is done weekly to ensure that errors have not propagated. This system of records provides a reconstructible account of all material transfers. A series of cross-checks and manage-

ment approvals of all operations involving fissile materials strengthens the administrative control and reduces the probability of human error.

Precautions against accidental criticality invoke the double contingency philosophy, that is, at least two independent situations, either of which is extremely remote, must occur simultaneously to cause accidental criticality. Moderator-limited zones in the hot cells are defined wherein the quantity of hydrogenous materials is strictly controlled and the possibility of flooding is essentially nil. For moderator-unlimited zones in the hot cells, the allowable masses of fissile materials are given in Table I; these values are based on water-flooded and fully reflected conditions.^{5,6} Although these limits are given for isolated units, they are also safe for small arrays because the minimum dimension of each zone is 91.4 cm, and there is a 20.3-cm-wide exclusion band separating the zones. Fissile material is not allowed in these exclusion zones. Nowhere do more than four zones come together, so the exclusion distance and fuel assembly size provide sufficient separation to assure subcriticality. Both HFEF/North and HFEF/South have fuel storage pits in the hot cell floors. Analysis by the KENO Monte Carlo program indicates that the effective multiplication factor, k , for normal conditions in the pits is 0.42 ± 0.02 ; when calculated for pit flooding, k increases to 0.67 ± 0.03 .

The amount of hydrogen-bearing materials in the hot cells at HFEF is subject to rigorous controls. Only 500 g of any hydrogenous material is permitted in each moderator-limited zone. As examples of these controls, lubricants are used in the minimum quantity necessary to maintain equipment in satisfactory operating condition, paper and cloth wipes are considered contributors to the moderator limit, solid plastic mounts utilized for metallographic samples are treated as equivalent to 25 g of water per mount, and plastic radiological smear holders each count as 8 g of moderator.

Special attention is paid to clean up operations that generate liquids and dry wastes. Collection tanks and sumps for liquids are comprehensively evaluated for physical design, location, etc., as related to criticality control. Vacuum cleaners are used for dry nonaqueous cleaning only; the units are weighed frequently and must be emptied before their contents reach 2000 g. Fissile masses in any single waste container are limited to 500 g of ^{235}U or 260 g of either ^{239}Pu or ^{233}U , and to no more than 500 g of moderator. For cans with unrestricted quantities of dry organic waste, the fuel content is limited to 50 g of ^{235}U . Special waste containers may hold up to the fissile masses specified in Table I if they are not stored in the pits and if no other nuclear fuel is in the same zone.

Our primary objectives at HFEF are to ensure that (a) no credible single contingency can lead to accidental criticality, (b) multiple contingencies will not arise simultaneously from undetected malfunctions, and (c) sufficient audits are made to ensure adherence to the rules. A formal, ongoing program of training and strict administrative controls is the basis for criticality prevention.

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TABLE I
Allowable Masses of Fissile Material in Moderator
Unlimited Zones in HFEF Hot Cells

Form	Masses (kg)		
	^{235}U	^{233}U	Plutonium
Moderated solutions or dispersions of particles	0.35	0.25	0.19
Loose solid capsules or elements	0.35	0.25	0.25
Capsules or elements in a fixed geometry such as a subassembly, if:			
H-to-Fissile Atom Ratio ≤ 2	9.00	2.80	2.40
H-to-Fissile Atom Ratio > 2	9.00	1.30	2.20

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Nuclear Safety Training, Cathy D. Blackwell (LANL)

Intensified training efforts in the area of nuclear safety in the past few years have developed as a result of the growth of the industry as well as incidents such as the accident at Three Mile Island and the Karen Silkwood case. These efforts have resulted in a more innovative era in training programs and techniques. Computer-based education (CBE) is a result of this innovative time, and it offers some particularly fascinating, effective alternatives to traditional teaching methods.

CBE's effectiveness is functional in many aspects of employee training. It may be used to determine particular areas of employee training needs, prescribe or present the appropriate training, administer a comprehensive test following the training, and evaluate the results of the test. CBE capability also enables it to reference the instruction as well as the testing to ensure that specific needs of each user are addressed. This system exhibits great potential in the area of

management of various kinds of safety certifications and documentation.

The Los Alamos National Laboratory (LANL) is planning to expand the development of a criticality safety training program in their plutonium facility, employing CBE techniques. A central program will be developed addressing basic concepts of nuclear safety common to many operations and, consequently, many government contractors. The development of site-specific processes will be dealt with as well that will satisfy actual needs in each facility.

CBE offers many advantages to nuclear safety training. One of the most important advantages is the quality of instruction and consistency in the presentations simply by the virtue of the automated presentation.

This presentation also requires employee interaction by posing periodic questions during the lesson. The employee must successfully answer the question for the training program to continue. Demonstration of mastery of each section is required for the program to progress to the next section. Successful completion of the test is recorded, and certification is documented by the computer.

A nuclear safety training pilot program has been developed and implemented at LANL using CBE techniques. Currently, lessons are not presented by the computer. However, lesson assignment, testing, and evaluation are managed by the computer.

Preliminary evaluation of this program indicates the success and effectiveness of CBE. Individual training and certification of ~200 employees was successfully completed in a period of three months using one computer terminal and one professional responsible for the program.

Since time away from the job averaged 1.5 h, and only one person was trained at a time, no process was significantly interrupted by the certification procedure. The automated record-keeping capability documented that all of the employees understand the basic concepts of criticality safety. The computer also determined the extensiveness of training needed in each case, which enabled the avoidance of under or over training personnel, maximizing the effectiveness of the training sessions.

Perhaps the greatest obstacle of the program dealt with employees with poor reading skills. Due to the necessity of reading skills, the certification process in this case was undoubtedly slowed, as the employees had difficulty understanding the questions. Further attention will be paid to this

problem in future course developments. However, with CBE, it is a problem that can be addressed with minimal effort.

CBE is a very effective, efficient vehicle for managing safety programs and certifications. Plans to continue development of a CBE criticality safety training program focus on developing a course that will be applicable to a variety of processes at various government contractor sites. This project illustrates the outstanding potential for government contractors to develop and utilize CBE training courses that will be mutually effective. CBE is an innovative vehicle that will lead the way for joint training projects in the years to come, providing a safer, more controlled, more productive environment for the nuclear industry.

Nuclear Criticality Safety Training in an NRC-Licensed Facility, R. W. Carson, Jr. (B&W)

Training has been defined as the development of the habits, the thoughts, or the behavior of a person by discipline and instruction. Habits. Thoughts. Behavior. At the Naval Nuclear Fuel Division of the Babcock & Wilcox Company, we try to develop *good* safety habits, *positive* thinking about nuclear criticality safety, and behavior that *strengthens* a good nuclear criticality safety program.

First, good safety habits are encouraged by instructing people in how to do it right. It is not enough to post limits and expect people to follow them. At our facility, instruction begins for a new employee the first day on the job. Approximately 15 min of classroom orientation is presented to the new employee concerning nuclear criticality safety. Topics such as the theory of criticality and the safety record of the nuclear industry are discussed. Also included is a description of the nuclear criticality safety program at our facility—the controls, limits, postings, and audits. Once the new employee reaches his work station, he is given specific nuclear criticality safety training, which includes the location of postings in his area. Annually, a formal training program that lasts ~45 min is given to all our employees (~2200). About six months after the formal training program, a "continuing awareness" session is shown on our in-plant video system. This session usually consists of an interview with a member of the nuclear criticality safety group in which current topics of concern are discussed. The interview lasts ~10 min and is shown to all personnel.

In a typical nuclear criticality safety training program at our facility, we show slides or videotapes of areas, equipment, or components with practical applications of the limits. For example, we might illustrate required distances between units of special nuclear material or show the proper level of Raschig rings in an unfavorable geometry tank. Practical application—training people to develop good safety habits.

Second, we try to develop positive thinking about nuclear criticality safety. Facts and interest promote positive thinking. We tell people what they *need* to know, but go one step further and tell people what they *want* to know. We are often asked: What happens if a criticality accident occurs? What would I see? What would I hear? What would I feel? Would the plant be shut down? Would a criticality accident really affect *me*? These questions and others are addressed in our training programs. We discuss the consequences of a criticality accident. We discuss the history of accidental criticalities. We even let our people experience some of the sights and sounds of an accidental criticality by a simulation in the classroom. Our most recent annual training program was highlighted by a dramatic videotape depicting a scenario of events that led to a simulated accidental criticality. Professional actors played the part of two employees returning from lunch at a nearby grill. The sights

and sounds they experienced as they tried to return to work are typical of what could be expected at the site of a criticality accident.

We try to make our training interesting by involving our people and showing familiar scenes. We have given them the facts and cultivated their interest. We reinforced the facts with the dramatic part of the program. By showing the negative aspects of a criticality accident, positive thinking is channeled toward the support of the criticality safety program. They leave the training area talking nuclear criticality safety and thinking nuclear criticality safety. They return to their work area, not afraid since they know the facts, but concerned because they know the consequences.

The third point, behavior that strengthens a good nuclear criticality safety program, is really what it is all about. If the operators, technicians, supervisors, and even managers do not apply the training they have received, it is all for nothing. Good safety habits coupled with positive thinking will always produce the proper behavior—behavior that reflects a sense of responsibility and even a sense of pride.

Since applying this philosophy of training at our facility, we have seen the number of minor deviations from posted limits decrease at the rate of 59% in 1977, 74% in 1978, 4% in 1979, and 13% in 1980. Gross departures or situations that alone could lead to an accidental criticality have been nonexistent.

Nuclear criticality safety training is a vital part of a total safety program. Good habits and positive thinking yield proper behavior. Proper behavior strengthens a good nuclear safety program.

The Nuclear Criticality Safety Program for the Albuquerque Operations Office, U.S. DOE, Robert R. Walston (DOE-Albuquerque)

Criticality safety for the Albuquerque Operations Office (ALOO) of the U.S. Department of Energy (DOE) encompasses just about every combination of risk that can exist with fissile materials. Figure 1 depicts the various usages of fissile materials by the ALOO contractors. A variety of criticality control methods exist, and a variety of administrative styles are used. In each case, criticality safety responsibility is passed through line management to the line worker. Although each contractor has a nuclear safety program peculiar to his operation, there is a common thread running through the entire group. That thread is

1. appropriate contractor talent
2. DOE requirements met
3. adequate surveillance structures.

Figure 2 shows typical nuclear criticality safety elements found in the various ALOO contractor programs and at the ALOO.

Looking briefly at each one of the common threads, one finds the following:

1. Appropriate contractor talent. This means that the contractor not only has the technical talent to do the physical work with materials and machines, but also has review committees with a strong spirit of inquiry into theoretical, administrative, and physical aspects of criticality safety for each operation.

2. DOE requirements met. This means that responsibilities are adequately handed down, appropriate training is utilized, operationally independent and trained personnel oversee all aspects of operational safety, and finally, that appropriate operational and emergency personnel and systems are always in readiness.

Fig. 1. Fissile materials usage within ALOO.

Fig. 2. Typical elements of safety programs.

3. Adequate surveillance structures. This means that adequate reviews and audits are performed, and at the operational level that . . . surveillance and maintenance schedules exist and are enforced.

Nuclear criticality safety responsibility flows from the DOE Secretary through the appropriate Undersecretary, through the appropriate Operational and Environmental Safety Division (HQ), through the Head of Field Office (manager, Albuquerque Operations Office), through the facility managers (labs, etc.), and finally through facility organizations to operating level supervision.

Often we see organizational structures that have various safety committees that approve plans and procedures for fissile material operations prior to startup. This approval system in our estimation does not preclude the operator or supervisor ultimately being responsible for his safety. In other words, where is the buck going to stop in case of an accident: it stops with line supervision and his employees. In all probability, those safety committees will be treated as consulting or advisory groups in such situations.

We believe that adequate review and audit processes by internal review groups and by DOE staffs can be carried on without at the same time upsetting the flow of operational responsibility. And we believe that hazardous operations can be done safely if given proper flow of responsibility and adequate safety measures.

Other DOE field organizations use a variety of techniques to ensure nuclear criticality safety. One such technique is to provide specific DOE approval of individual operations, i.e., DOE sign-off on individual procedures. Another technique is to have site representatives who play a close support role in the contractor's procedural development. Each of these techniques would probably involve close liaison with the contractor in his safety committee actions and his surveillance activities.

The ALOO performs its criticality safety assurance role in a slightly different manner. Starting with a new facility or new operation, one of the safety division groups would have had significant input back in the early design stages and would have utilized numerous general design criteria as input. During final preparation of the new operation, a pre-operational review would be conducted to ensure that no new risks were evident and that the contractor was indeed staffed, trained, and in general had his systems operational per plan. After routine operations have been established by the contractor, the ALOO performs routine appraisals (annually) of the contractor's overall nuclear criticality safety program. This is done by reviewing the operation with supervision, inspecting the facility, and by having discussions with management and safety committees. Procedures and

documentation on training and internal reviews are also inspected.

Findings and recommendations are written into the ALOO appraisal report when improvements or corrections need to be made. They constitute a formal process of specific communication. These require a formal response from the contractor management and usually include proposed corrective action and due date. These items are followed up on subsequent visits and appraisals.

Administrative Practices in the United Kingdom in Relation to Criticality Safety, *Keith Smith (NII-U.K.)*

An outline of the nuclear industry in the United Kingdom is given, with the main object of drawing a distinction between "licensed" and "nonlicensed" sites. Reference is made to the requirements, under the Nuclear Installations Act, for all except the U.K. Atomic Energy Authority (UKAEA) and government departments to be licensed. It is made clear that although not subject to licensing, the UKAEA and government departments have similar duties imposed on them by the Nuclear Installations Act. The Health and Safety at Work Act imposes requirements on all, as far as health and safety are concerned, and these are mentioned.

Reference is made to current Nuclear Installations Inspectorate (NII) policy, in criticality safety matters as in other safety fields, not to impose rigid requirements, and reasons are given for believing this course to be beneficial. The emphasis in current licences on the requirement for a licensee to make "arrangements" in consultation with his safety committees is brought out, and the regulatory control of the structure and membership of the safety committees is referred to.

The general "criticality clearance" practice of having a set of rules supported by a criticality safety argument is mentioned, and comment is made on the different approaches used in the United Kingdom. Reference is made to the checking of the operational assumptions, and the validity of the technical data used, and some ways in which documentation is processed from drafting to approval for use are outlined.

In a section concerned with training practices, the various groups involved in a criticality safety "system" are listed. The general practice of individual sites in the United Kingdom taking care of their own training needs is referred to, and the approaches used on major sites are outlined.

Comment is made on the influence of limited resources and the need for ensuring the most effective deployment of safety effort.

Benchmarks and Validations for Nuclear Criticality Safety

Remarks on the Need for Experimental Benchmarks in Nuclear Criticality Safety, *Hugh K. Clark (SRL)*

A cardinal principle in nuclear criticality safety evaluations is the validation of any computational method by correlation with pertinent criticality experiments to establish its bias. Commonly, the correlation is expressed in terms of values of k_{eff} calculated for the experiments, in which case the bias is the deviation of k_{eff} from unity. Both the bias and its uncertainty are generally functions of the material composition and geometric arrangement. The uncertainty in the bias is an estimate of both the precision of the computational method and the accuracy of the experimental data. The uncertainty in the bias must allow for extension of the bias beyond the range of experimental data. Such an allowance is always required. Otherwise, there would be no need for calculations, because directly applicable experimental data would exist for the conditions being evaluated.

The most practical approach in an evaluation is to choose k_{eff} as a function of the system variables such that, with due allowance for the bias and its uncertainty, one has confidence that conditions calculated to correspond to this k_{eff} will be subcritical. Where there is little extension outside the range of experimental data, and both the calculational method and the experimental benchmark data are quite precise, a k_{eff} 0.01 or 0.02 below the values of k_{eff} calculated for the critical experiments may not be difficult to justify. However, where the extension is large, the closest data are poor, and the method is imprecise, a margin as large as 0.1 may not be sufficient to provide confidence. The choice of the subcritical value of k_{eff} clearly requires judgment, and the evaluator is generally best qualified to make the judgment. A large margin below the critical k_{eff} is obviously easier to justify than a small one, but an unnecessarily large margin may prove expensive in terms of operating costs.

Careful attention to the establishment of bias is of particular importance. It is important to examine the benchmark data for consistency and to determine the effect on the critical value of k_{eff} of quoted experimental errors. It is desirable to correlate with all data of reasonably good quality in the area of interest. Scatter in the data, expressed as scatter in the critical value of k_{eff} , may be a better indication of uncertainty than quoted experimental errors.

Calculational methods used for criticality safety evaluations are commonly one-dimensional (diffusion or transport theory) or three-dimensional Monte Carlo. Suitable benchmark experiments for the former are also necessarily essentially one-dimensional (i.e., experiments with spheres or with series of cylinders or cuboids that permit extrapolation to infinite height or extent). In principle, any experimental configuration may be represented by the Monte Carlo geometry package, but if the configuration is complex, it will be

tedious and subject to error to do so. Furthermore, it may not be possible to use simplified packages, such as KENO geometry, and the result will be longer computing times. Thus, there is incentive for three-dimensional experiments also to be relatively clean.

It is important that experiments to be used as benchmarks be described accurately, completely, and unambiguously, so that the preparation of computer code input for correlation does not require assumptions or a phone call to the experimenter. Directly measured parameters should be given rather than derived quantities. It is essential that the same recipe for computing aqueous solution densities be followed in criticality safety evaluations as was followed in establishing bias. It is important that estimates be provided of the uncertainty in the experimental results.

Good benchmark data in previously uncharted areas should reduce the margin allowed for uncertainty and permit more economical operations. However, for any particular problem, one must proceed with what data are available. It would be presumptuous to attempt to identify all those areas where data are needed. A recent examination of limits in the ANS-8 standard and the calculation of additional limits to be proposed for inclusion revealed some areas where one-dimensional experiments would be helpful.

Experiments with aqueous solutions of ^{233}U at high concentrations are needed. Experiments with dry or damp ^{235}U and ^{233}U oxides would be useful. A comparison of the water-reflected undermoderated sphere and cube would be of some interest. Experiments with undermoderated lattices would be helpful if one wishes to use moderation control. Many early criticality experiments were not done with the thought that they might be used as benchmarks. Even at this late date, it might be worthwhile to dig back through old log books and perhaps improve the quality of the data for benchmarking purposes, publishing it in more readily accessible reports.

In conclusion, benchmark experiments are of great importance in nuclear criticality safety evaluations. A huge debt is owed to those who have been so diligent in supplying data. However, many uncharted areas still remain.

Experimental Benchmark Needs for Criticality Safety, *Ryohei Kiyose (Univ of Tokyo-Japan), Iwao Kobayashi (JAERI-Japan), Yasushi Nomura (Japan Inst Nuclear Safety)*

Many criticality safety experiments have been performed in the world; however, the experimental criticality data are not yet sufficient for reasonable and effective design and operation of nonreactor nuclear facilities in the fuel cycle of light water reactors and fast breeder reactors. The

data needed are of low-enriched uranium, plutonium, and plutonium-uranium mixtures, of those chemical and physical forms as are found in nuclear fuel reprocessing, fabrication, storage, and transportation facilities.

Experimental data of the following items are needed especially for Pu-U mixture systems, although information on similar items for low-enriched uranium and plutonium systems is also not quite sufficient:

1. criticality of liquid solutions of Pu-U mixtures in cylindrical, annular, and slab vessels
2. criticality of dry and damp powders of mixed oxides of PuO_2 in natural UO_2
3. effect of degree of moderation (low-density water or hydrogen content below that of full-density water) on criticality of undermoderated mixed-oxide units and fuel assemblies
4. criticality of liquid solutions of Pu-U mixtures where the isotopic composition of plutonium is ~ 20 to 30 wt% ^{240}Pu
5. interacting arrays in water of fuel bundles, assemblies, damp oxides, or solution vessels, with a water, lead, or steel reflector
6. effect of neutron absorbers on criticality of homogeneous systems and heterogeneous systems such as fuel pins in Pu + U solutions

7. effect of fixed absorbers on criticality of aqueous solutions

8. criticality experiments to establish equivalence of ^{235}U versus plutonium in mixed-oxide systems of 20 to 30 wt% plutonium in enriched uranium (10 to 15 wt% ^{235}U).

Benchmark calculations have been done thus far in Japan for some 800 cases of criticality experimental data to validate the KENO-IV Monte Carlo code¹ and MGCL Multigroup constants library.² MGCL is generated with the computer code system, MGCL-ACE, from ENDF/B-IV, and the data form is same as the ABBN set or the JAERI-FAST set with the Bondarenko-type shielding table and consists of one- and two-dimensional data. The energy group structure is the same as the Japan Atomic Energy Research Institute (JAERI) standard 137 group structure. The cross sections of MGCL can be collapsed to any number of energy groups with a standard neutron energy spectrum, and a submaster library (usually 26 energy groups) is generated. Most calculational results for some 600 cases are reported in several JAERI-M reports, corresponding to the different experimental systems.

Considering the importance of criticality safety for the second reprocessing plant planned in Japan, it will be mandatory to validate the code system with the experimental data of uranyl nitrate or plutonium nitrate or their mixture solution systems. There are rather a small number of reports

Fig. 1. Histogram of the effective multiplication factors.

available concerned with this kind of experimental system. Figure 1 is the histogram of results of K_{eff} calculations with the MGCL-137 data library and KENO-IV code. The mean value and the standard deviation of the distribution are given as 0.9809 ± 0.0312 . In the diagram, one can see obviously a principal peak located at about $k_{eff} = 0.995$ and the other lower peak at about $k_{eff} = 0.92$. The latter peak is mainly attributed to the calculations for the unreflected uranyl nitrate solution in slab and cylinder systems.³ Further investigations, including reviews and calculations of other similar experimental systems, are needed.

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Economic Incentives for Additional Critical Experimentation Applicable to Fuel Dissolution, J. F. Mincey, R. T. Primm III (ORNL), W. R. Waltz (AGNS)

Fuel dissolution operations involving soluble absorbers for criticality control are among the most difficult for which to establish economical subcritical limits. The paucity of applicable experimental data can significantly hinder a precise determination of a bias^{1,2} in the method chosen for calculation of the required soluble absorber concentration. Resorting to overly conservative bias estimates can result in excessive concentrations of soluble absorbers. Such conservatism can be costly, especially if soluble absorbers are used in a throwaway fashion. An economic scoping study will be presented that demonstrates that additional critical experimentation will likely lead to reductions in the soluble absorber (i.e., gadolinium) purchase costs for dissolution operations. The results indicate that anticipated savings may be more than enough to pay for the experimental costs.

Dissolution operations with U + Pu oxide fast breeder fuels and light water reactor (LWR) uranium oxide fuels form the basis for this study. The LWR dissolution operation is that anticipated at the Barnwell Nuclear Fuel Plant (BNFP) by Allied-General Nuclear Services (AGNS). The mixed-oxide operation is based on a prototypic consolidated fuel reprocessing program (CFRP) dissolver design. Gadolinium-bearing dissolver outflows are estimated at 4.2×10^6 and 5×10^5 l/yr for the BNFP and CFRP dissolution operations, respectively. The assumed³ gadolinium purchase price is \$118/kg. On the basis of a gram per litre of gadolinium (g-Gd/l) and no recycling of gadolinium, the annual purchase costs are $\sim \$4.9 \times 10^5$ [yr-(g-Gd/l)] and $\sim \$5.9 \times 10^4$ [yr-(g-Gd/l)] for the BNFP and CFRP dissolution operations, respectively.

Figure 1 shows³ existing and projected BNFP gadolinium concentration requirements. The horizontal portion of the existing requirements curve results from the imposition of conservatism in the absence of applicable experimental data for low (i.e., <3%) enrichments. Consequently, it is not possible to precisely define a bias for this portion of the curve. The projected requirements curve reflects a reduction in conservatism that could probably be realized if additional applicable benchmark data were available. If 70 wt% of the BNFP rated throughput is assumed to be 2.0 to 3.1% enriched pressurized water reactor fuel and the remainder

Fig. 1. Projected and existing gadolinium concentration requirements for BNFP's dissolution operation.

1.5 to 2.8% enriched boiling water reactor fuel, an annual 1.25 g-Gd/l reduction in the BNFP dissolver outflow is estimated.³ This corresponds to a $\$6.2 \times 10^5$ reduction in the yearly gadolinium purchase costs projected for the BNFP operation. In contrast, the necessary experiments are projected to cost between $\$2 \times 10^5$ to $\$5 \times 10^5$. Assuming a 30-yr plant life, 2 years to perform and apply the results of the experiments (assumed cost = $\$5 \times 10^5$), and a 15% discount rate, the payback period is <3 years.

A similar argument can be made for the proposed CFRP dissolution operation. The CFRP dissolver relies on mechanical agitation of sheared fuel to enhance dissolution. This feature has resulted in a design where the H/Pu ratio for large masses of mixed oxide (perhaps >50 kg) can vary from ~0 to 400. Hence, the use of both mass control (for the less thermalized neutronic conditions) and gadolinium as a soluble absorber (for more thermal conditions) is anticipated. Calculational studies of expected dissolver slurries (containing 3 to 5 g-Gd/l concentrations) indicate that for a fixed U + Pu mass, K_{eff} is at a maximum in the vicinity of $10 < H/Pu < 50$. The existing experimental data⁴ have a minimum H/Pu of ~196 and a maximum 1.338 g-Gd/l concentration. Therefore, to establish a computational bias at the neutronic conditions of concern, a large extrapolation is required.

For critical experiments⁴ with H/Pu ratios of 196 to 458 (and no gadolinium), calculated K_{eff} values range from ~1.01 to 1.03, respectively. Extrapolated biases against the H/Pu ratio indicate that for the $10 < H/Pu < 50$ region of interest, a calculated K_{eff} of unity could be considered critical. However, for critical experiments⁴ having 0.020 to 1.338 g-Gd/l (H/Pu = 258), calculated K_{eff} values range from ~1.00 to 0.99, respectively. Using the gadolinium concentration to draw the bias, calculated K_{eff} values of

0.96 to 0.97 might be considered critical for systems containing 5 g-Gd/l. From the above trends, it would appear reasonable to expect that the lower limit on calculated K_{eff} values considered critical for the dissolver could perhaps be raised to 0.98 if additional applicable experimental data were available to refine the bias estimate.

The g-Gd/l concentration savings that correspond to increasing the calculated K_{eff} value considered critical by 0.02 (0.96 to 0.98) is, at a minimum, ~ 2 g-Gd/l. This anticipated reduction in conservatism (believed possible if more applicable experimental data existed) represents an annual savings of $\sim \$1.2 \times 10^5$ in gadolinium purchase costs to the CFRP operation. For the same 30-yr plant life, 2-yr experimentation-analysis period, and 15% discount rate, a $\$5 \times 10^5$ series of experiments would have a payback period of ~ 10 years.

The analysis has ignored the fact that the CFRP operation is only pilot-scale. The high throughput of a full-scale plant will further enhance the economics of conducting the mixed-oxide experimental series. A second omission was ignoring the increased high-level waste disposal costs that would result from using excessive gadolinium concentrations. In conclusion, despite these omissions, both series of proposed experiments appear cost effective.

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Computer-Aided Identification of Benchmark Criticality Data, B. L. Koponen (LLNL)

Individuals who prepare benchmark calculations for code validation are well aware that the benchmark data sources are voluminous. The reports collected for our bibliography on criticality experiments occupy three five-drawer file cabinets. Finding and selecting the right data can be a trying task. The difficulty is compounded because, in many cases, the same experimental data are reported in several publications, sometimes with corrections of earlier errors, or in an evaluated "clean" format. To do their jobs properly, the criticality analysts have to be aware of the best sources for their needs. This paper addresses how some tools that are being developed, especially computer data-base systems, can make the job easier. In this paper, we show what we have done to make this task easier, and how it may be of benefit to others.

At Lawrence Livermore National Laboratory (LLNL), we have been using computer data-base management systems to manage benchmark data in two formats: (a) the bibliography of experimental criticality data and (b) the numerical benchmarks in a form ready for use by criticality computer codes. The first of these efforts resulted in a three-volume publication¹ of references relevant to well-documented relatively clean experiments. It contains indexes and concordances that can be used profitably without access to the data base. We will present some examples of how this publication has helped us find benchmark data for a computational

study of water-moderated plutonium arrays.² The second part of this work, the numeric data, has not yet been made available outside of LLNL, but the Oak Ridge National Laboratory has made its series of SCALE/CESAR benchmarks available through the Radiation Shielding Information Center.

We have transferred our bibliographic data files to the U.S. Department of Energy (DOE) Technology Information System (TIS). This dedicated system, operated by LLNL for the DOE Office of Energy Systems Research, is a new generation information machine capable of storing, retrieving, and displaying textual, numeric, and graphic data. The system is being used successfully also by those who are not computer experts. We have started the construction of a general data base for nuclear criticality safety. This includes the bibliography, experimental benchmarks, news items on criticality safety, and a description of the dominant models for criticality calculations. The overall data base will thus permit us to update, extend, and publish with greater ease, and to share the results of our work with a larger user community. The options of electronic mail and automated connections of authorized users to other information centers and computers make TIS especially attractive for this type of rapid technology transfer. When it is completed later this year, we are planning to make the criticality safety data base accessible via TIS to other criticality specialists.

The TIS is available by telephone dialup at 300/1200 bps from commercial and FTS telephone lines, the WATS toll-free service, ARPANET, and shortly also via TYMNET/TELENET, which connect to international communication networks.

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Criticality Considerations of Interim Spent Fuel Storage, C. E. Jewell (SRP)

The objective of the U.S. Department of Energy (DOE) Interim Spent Fuel Management (ISFM) program is to ensure that no domestic commercial power reactor is required to shut down due to lack of storage space for spent nuclear fuel. To accomplish this objective, the DOE accouned in October 1977 that the federal government would, under certain conditions, take title to and store spent nuclear fuel from commercial reactors. Storage would be at a government-owned away-from-reactor (AFR) facility, and the total costs would ultimately be borne by the utilities. In March 1981, the DOE announced that it would discontinue its efforts to provide government-owned AFR storage facilities.

Table I shows that beginning in 1986, a significant need for additional storage capacity over and above current utility plans will likely occur.¹ The need will persist until the inventory is reduced by the availability of either reprocessing or permanent geological repositories. To alleviate this need, the ISFM program currently focuses its activities on the development of alternate spent fuel storage technologies designed to enhance utility at-reactor storage capabilities. The concepts that are currently available or being considered for use include (a) the water pool concepts of reracking and/or double tiering existing basins, building an additional basin,

TABLE I
Fuel Discharge and Storage Projections
(cumulative MTU)

End of Calendar Year	Total Amount Discharged	Amount Stored at Reactor Sites	Additional Storage Requirements
1980	6,880	6,880	0
1981	8,280	8,280	0
1982	9,590	9,590	0
1983	11,480	11,480	0
1984	13,660	13,660	0
1985	16,070	16,070	0
1986	18,690	18,570	120
1987	21,800	21,410	390
1988	24,830	24,210	620
1989	28,540	27,300	1,240
1990	31,970	30,170	1,800
1995	50,940	44,460	6,480
2000	72,050	56,040	16,010

or utilizing consolidated rod storage, and (b) the dry concepts of storage casks, vaults, silos, and drywells. Only reracking and double tiering have been licensed in the United States.

Based on comments received during meetings with the utilities, their principal concern is the probability of extended delays in the licensing process. Therefore, the current program is designed to provide data and equipment that will assist the utilities in licensing various at-reactor storage modes. The ISFM program strategy relating to wet storage is directed toward implementing rod storage technology because basin construction reracking and double tiering are commercially available licensed technologies.

The dry storage program strategy is directed toward implementing storage in casks, drywells, silos, and vaults because all have potential as a viable technology. One programmatic objective applicable to both wet and dry strategies is to provide the utilities with generic licensing documentation, including a detailed nuclear criticality safety analysis for each storage mode.

The early segment of the ISFM program dealt mainly with AFR storage in water pools. The major criticality concerns were the reactivity effects of reracking the storage pools and/or the storage of disassembled and consolidated fuel rods. The Babcock and Wilcox Research Center, under contract to the DOE, conducted benchmark experiments covering both of the above cases.^{2,3} An analysis by Battelle-Pacific Northwest Laboratories is being conducted to determine possible accident scenarios leading to criticality in a spent fuel storage pool and to determine the termination mechanisms and possible radiological consequences. This study does not include an analysis of accidents in pools utilizing rod storage.

Existing standards reflect the criticality data that might be required for the redirected ISFM program. ANSI/ANS-57.7, "Design Criteria for an Independent Spent Fuel Storage Installation" (water pool type), was approved in February 1981. A draft of its counterpart for dry storage, ANS-57.9, is scheduled for issue in 1982. ANSI/ANS-57.7, and probably -57.9, require that the criticality analysis of the storage rack design consider the full range of both fuel and moderator densities. Currently, no benchmark data have been validated for reduced moderator densities in spent fuel storage. Also, no technique has been identified to extrapolate benchmark data from fully flooded to dry conditions.

If storage space at a reactor site is at a premium, a spent fuel reactivity monitor could be used to expand the capacity of some storage modes. ANSI/ANS-57.7 permits credit for burnup in the analysis of storage rack designs. However, an allowable fuel assembly reactivity must be established, and each fuel assembly must be shown by measurement to meet this criterion. These measurements could be accomplished by either static or pulse techniques.

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A Review of Subcriticality Measurements Using ²⁵²Cf-Driven Power Spectral Density Measurements, John T. Mihalczo (ORNL)

The ²⁵²Cf neutron-driven noise analysis method has two advantages over other methods of reactivity determination: (a) the subcriticality of a multiplying system can be determined without knowledge of its neutronic properties at delayed criticality, and (b) the interpretation of the measured data to obtain the subcriticality does not depend on knowledge of the inherent source strength or the detection efficiency. Thus, this method can be used in the initial fuel loading of systems where determination of subcriticality dependent on some calibration near critical is not possible. It can also be used in determining the reactivity of assemblies where sufficient material to achieve criticality is not available or where loading to critical is undesirable. For some subcritical systems, the method may have disadvantages because of the requirements that the neutron source intensities be comparable to those of the inherent sources in the fuel materials, and that the neutron detectors be in close proximity to the fuel materials. This paper discusses the principles on which this technique is based and summarizes the results of typical measurements that have been performed.

The ²⁵²Cf neutron-driven noise analysis method for obtaining the reactivity from cross power spectral measurements was developed to avoid certain difficulties inherent in time-domain measurements, such as Rossi- α and pulsed neutron measurements. The probability of detecting a correlated count at time t after an initial count in the Rossi- α measurement¹ is

$$D R \alpha(t) = \frac{W^2 F X R k_p^2}{\alpha l^2} \exp(-\alpha t), \quad (1)$$

where F is the fission rate, W is the detection efficiency, X is the Diven factor, R is a spatial effects correction factor, and the other quantities have their conventional definitions.

The probability of a count after the injection of ν_c neutrons in a pulsed neutron measurement² is

$$D P N(t) = W \frac{\nu_c I_c k_p}{\nu_p I} \exp(-\alpha t), \quad (2)$$

where I_c and I are the average importances of californium neutrons and reactor fission neutrons, respectively. To avoid the difficulties of pulsed neutron sources, ²⁵²Cf was deposited on one plate of a parallel-plate ionization chamber located in a multiplying system.² The detection of the ionizing fission products from the spontaneous fission of ²⁵²Cf indicates the time of emission of ν_c neutrons per californium fission. Thus,

the ^{252}Cf serves as a randomly pulsed neutron source for time-domain measurements.

By combining these two types of time-domain measurements, an expression for the neutron multiplication factor is obtained.³ The ratio of correlated counts per ^{252}Cf fission in a randomly pulsed neutron measurement with ^{252}Cf to the correlated counts per count in a Rossi- α measurement is

$$\frac{\nu_c}{\nu_p} \frac{I_c}{\text{IRX}} \left(\frac{1 - k_p}{k_p} \right), \quad (3)$$

and thus provides the prompt neutron multiplication factor, k_p , if the other quantities are known. This ratio was used in both types of time-domain measurement with fast metal assemblies at delayed criticality in 1971 at Oak Ridge and Los Alamos³ to obtain values of the effective delayed neutron fraction.

A difficulty with time-domain measurements is brought about by overlap of fission chains, and this precludes, in practice, measurements of systems with large inherent sources and thermal systems. To avoid this difficulty, an expression equivalent to Eq. (3) was obtained in the frequency domain⁴:

$$\frac{G_{12}^* G_{13}}{G_{11} G_{23}} = \frac{\nu_c}{\nu_p} \frac{I_c}{\text{IRX}} \frac{(1 - k)}{k}, \quad (4)$$

where subscripts 1, 2, and 3 denote the ^{252}Cf ionization chamber and an additional pair of detectors positioned close to the system, respectively; G_{ij} is the cross power spectral density between detectors i and j ; and $*$ denotes complex conjugation.

Measurements have been performed with a mockup of the Fast Flux Test Facility (FFTF) reactor core at Argonne National Laboratory,⁴ with several unmoderated, unreflected uranium cylinders and a sphere,⁴ with a plate-type water-moderated research reactor⁵ at Oak Ridge National Laboratory, and with a uranyl fluoride fuel solution tank at Los Alamos National Laboratory.⁶

The experiments with the fast assemblies, the mockup of the FFTF reactor core (1974), which contained ~500 kg of plutonium, and the uranium metal sphere and cylinders (1975) confirmed the theory of the measurement and provided an accurate indication of the subcriticality down to $k = 0.8$.

Calculations⁷ (1978) indicated that the ^{252}Cf neutron-driven method might be feasible for some water-moderated systems. Experiments performed with a small plate-type research reactor⁵ (1979) whose neutron lifetime was ~57 μs showed that (a) the method was practical for thermal systems and (b) spatial effects introduce a frequency dependence to the various ratios of spectral densities, thus providing experimental confirmation of the presence or absence of spatial effects. Measurements with a 5% ^{235}U uranyl fluoride solu-

tion tank⁶ (1980) indicated that the method could determine the reactivity of fuel solution tanks containing as little as half a critical mass.

Since the ratio of spectral densities is independent of detection efficiency and source intensity, it should be a more useful quantity than the inverse count rate in initial loading of reactors or critical assemblies. The solution tank experiments confirmed this, since the spectral density ratio was linear with fuel solution height down to ~0.5 critical height, thus allowing a good prediction of criticality early in the filling of the tank. The inverse count rate versus fuel solution height plots, however, either underpredicted or overpredicted critical height until near-critical height was achieved.

The ^{252}Cf method of measurement appears to have potential application in (a) initial loading of reactors (light water reactors and liquid-metal fast breeder reactors), (b) refueling of reactors, (c) fuel processing and reprocessing facilities, (d) fuel storage facilities, (e) post-accident situations, and (f) basic experiments where sufficient fuel to achieve criticality is not available. Before practical application is implemented, however, additional experiments are required to investigate the following areas: the lower practical limit of measurement, restrictions on placement of detectors, spatial effects, and measurement times.

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Data and Analysis for Nuclear Criticality Safety—I

Continuous Nuclear Criticality Surveillance of Technological Facilities, Tadeusz Żółtowski (Inst Nuclear Research-Poland), Pedro Landeyro (CNEN-Italy), Wojciech Ratyński (Inst Nuclear Research-Poland)

INTRODUCTION AND OBJECTIVES

Due to the progress in technology and development of intensive fuel reprocessing methods, applying centrifugal extractor batteries, there is a strong demand for the method to observe closely some process facilities from the criticality point of view.^{1,2} This paper presents experiments with on-line, anomalous reactivity prediction of some facilities, providing alarm and possible plant protection functions.

CONTROL POLICY STRUCTURE AND EXPERIMENTAL

Recently, Schulze and Wuerz³ described the neutron monitor in the high-activity part of a reprocessing plant. They reported a unique relationship between k_{∞} and epithermal leakage flux multiplication. However, the basic importance has a knowledge of nuclear criticality index (e.g., k_{eff}). Physically, the relationship between M_{epi} and k_{eff} cannot be determined directly due to perturbations introduced by the presence of high flux modes. The problem

is increasing in real technological conditions when the geometry and physicochemical properties of fissile materials are varying with time. Neutron leakage flux multiplication measured during movement of external neutron source is one of epithermal profiles that may be used to achieve additional information about fissile material state and, in consequence, to correlate the M_{epi} value measured in the selected spatial point with the k_{eff} of a system. Moreover, the effect of measuring M_{epi} profiles is to detect overall departure from normal operation even without reference to the *a priori* known training states. The structure of control policy is shown in Fig. 1. Pattern recognition method allows to compare the actual M_{epi} profile with a predetermined training set member. Training set data have to be calculated by source-dependent neutron transport calculations or determined experimentally for typical process perturbation conditions. Finally, we can determine the spatial transient point of external neutron source at which the measured M_{epi} value is correlated with k_{eff} . For this purpose, modern control theory "fuzzy approach" or heuristics can be used.⁴

Continuous, time variable concentration gradients of fissile/poison slurry suspended in n-paraffins were produced in the annular chemical reactor. An external ^{252}Cf source was moved axially during each measurement. The experimental setup was operated under PDP 11/45 computer real-time supervision. An appropriate set of KENO code k_{eff} calculations was also carried out. Three sets of data have been

Fig. 1. Policy of continuous nuclear criticality surveillance.

Fig. 2. Results of predictability study.

collected: (a) M_{epi} training set profiles for ten sedimentation stages, (b) evaluation sets measured in the precise states of the training set, and (c) evaluation sets with time lags uniformly distributed within a 20-s time interval relative to the moments of training set measurements.

Two methods for the pattern recognition of evaluation profiles have been used: (a) statistical discriminant analysis and (b) pattern clustering and nearest neighbor pattern classification. Fourier and Hadamard transform coefficients have been assumed for compact representation of profile shapes.

SAMPLE RESULTS AND CONCLUSION

The effectiveness of the continuous criticality surveillance was very high in all cases when evaluation data have been measured in exact training states. More than 95% of correct k_{eff} determinations have been obtained. It is worth noting that in the experimental conditions, k_{eff} values ranged non-monotonously from 0.3 through 0.7 to 0.6. The results of "predictability study" using the evaluation set with time lags are presented in the Fig. 2. It is also clearly seen that k_{eff} values have also been recognized excellently. The first two diagrams show the result of recognitions in the case of overlapping evaluation states because of comparable time lags with the time difference between neighbor states.

Results presented show that the method allows us to continuously observe k_{eff} value variation of selected process facilities. Modern control theory can be used in that case as a

junction between the physically measured M_{epi} value and the system's k_{eff} . In this light, the method presented should be tested in plant conditions. Hot tests are being planned in the Eurax Saluggia Fuel Reprocessing Plant.

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Criticality Design for Low-Enriched Uranium/Water Systems: A Different Perspective of Homogeneous Versus Heterogeneous, E. Walker, R. I. Smith (Bechtel)

The design of systems and components for fuel fabrication facilities and in fuel reprocessing facilities may specify

process parameters for heterogeneous low-enriched uranium-water systems with reduced "lump" density. Such a situation may occur for pressed pellets prior to sintering in a fuel fabrication facility or in a calciner for nitrate-oxide conversion of facilities such as fuel reprocessing. This study was performed to determine the effect on critical dimensions of reducing the lump density in a water-moderated heterogeneous uranium oxide system.

The optimum moderation condition for uranium oxide pellets in water was determined for three different lump density systems. Optimum moderation for this study was defined as that combination of pellet diameter (POD) and water-to-fuel ratio (W/F) resulting in the maximum material buckling (B_m^2), and hence the minimum critical dimensions. This point does not result in optimum reactivity, which is defined as the condition producing the greatest infinite multiplication (k_∞). The data to determine these optimum moderation conditions were generated using the computer codes GAMTEC (Ref. 1) and THERMOS (Ref. 2). The results of these determinations are shown in Table I.

For each of the three densities, the "smear" density of the system was determined to permit direct comparisons with homogeneous systems. The results of this comparison are shown in Fig. 1. Note that for heterogeneous uranium-water systems with lump density in the range of sintered pellets (10.90 g/cm³), the heterogeneous system will result in a smaller critical dimension than a corresponding homogeneous mixture and is the more conservative condition. However, as the lump density is reduced below $\rho_0 = 8.0$ gUO₂/cm³, crossover occurs for which the homogeneous mixture will result in a smaller critical size than the corresponding heterogeneous system.

The implication of this exercise is that one must use caution in specifying the design basis parameters when attempting to optimize the criticality design of a process system.

Fig. 1. Effects of homogeneous versus heterogeneous UO₂-H₂O system moderation on the minimum criticality size of the system. Points for heterogeneous system correspond to the moderator/fuel volume ratio and fuel rod diameter for maximum material buckling at various fuel lump densities.

TABLE I

Lump Density (gUO ₂ /cm ³)	Optimum Parameters		Max B _m ² (cm ²)	k _∞
	W/F	POD (in.)		
4.72	2.50	0.500	0.01191	1.4813
8.00	3.10	0.400	0.01581	1.5106
10.90	3.56	0.300	0.01841	1.5205

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Neutron and Gamma Attenuation Characteristics of Various Shields During Nuclear Criticality Accidents, R. E. Swaja, R. T. Greene (ORNL)

The effectiveness of reducing radiation doses due to nuclear criticality accidents was determined for various shields fabricated of materials used to construct shipping containers and storage facilities for radioactive isotopes. These data are based on experimental results obtained during seventeen nuclear accident dosimetry (NAD) intercomparison studies conducted since 1965 at the Oak Ridge National Laboratory.¹⁻⁴

During these studies, criticality accidents were simulated by operating the Health Physics Research Reactor (HPRR)⁵ in the pulse mode and resulting neutron and gamma radiation doses were measured at area monitoring stations and on anthropomorphic phantoms. Pulse yields varied from 10¹⁶ to 10¹⁷ fissions which produced neutron doses between 0.2 and 8.0 Grays and gamma doses between 0.1 and 2.0 Grays. Various shields were placed between the HPRR and dose monitoring locations to modify the radiation spectra affecting the dosimeters. Experimental configurations of the reactor, shield, and dosimeters were arranged to minimize effects on measured dose due to side scatter. Perturbations caused by scattering from reactor building walls were also minimal due to low-scatter aluminum sheet metal construction. Using data measured at area monitoring stations, dose reduction factors and neutron-to-gamma dose ratios were determined for eight different shields.

Table I shows measured neutron, gamma, and total dose reduction factors at 2- and 3-m distances from the reactor for the shields used in the seventeen NAD studies. Shield and dosimeter distances are measured from the reactor vertical centerline to the front side of the shield (side facing the HPRR) and to the center of the accident dosimeter, respectively. Indicated errors represent one standard deviation about the mean of all individual dose measurements made by NAD study participants. Lucite and polyethylene shields were the most effective in reducing total and neutron doses from the simulated criticality accidents; steel and graphite shields were the least effective; and concrete and steel-concrete shields were similar in effectiveness at 2- and 3-m distances. The relatively low gamma dose reduction at 3 m

TABLE I

Measured Radiation Dose Reduction Factors for Various Shielding Materials During Criticality Accidents*

Shield	Shield distance from HP RR, m	Dosimeter distance from HP RR, m	Shielded dose/Unshielded dose		
			Neutron	Gamma	Total
5-cm steel	1	3	0.64 ± 0.11^a	0.52 ± 0.08	0.62 ± 0.06
13-cm steel	2	3	0.36 ± 0.01	0.29 ± 0.04	0.35 ± 0.01
20-cm graphite	1	3	0.31 ± 0.09	0.56 ± 0.19	0.35 ± 0.09
20-cm concrete	1	3	0.25 ± 0.05	0.55 ± 0.12	0.29 ± 0.03
5-cm steel-15-cm concrete	1	3	0.24 ± 0.10	0.56 ± 0.27	0.28 ± 0.07
12-cm Lucite	2	3	0.21 ± 0.02	1.13 ± 0.08	0.34 ± 0.03
23-cm Lucite	2	3	0.07 ± 0.02	0.67 ± 0.04	0.15 ± 0.01
25-cm polyethylene	1	3	0.10 ± 0.02	0.59 ± 0.17	0.17 ± 0.03
20-cm concrete	1	2	0.26 ± 0.13	0.61 ± 0.12	0.31 ± 0.12
5-cm steel-15-cm concrete	1	2	0.20 ± 0.10	0.47 ± 0.12	0.24 ± 0.09

*Based on area station dose measurements obtained during seventeen nuclear accident intercomparison studies.

^aOne standard deviation about the mean of all individual dose measurements.

TABLE II

Measured Neutron-to-Gamma Dose Ratios for Various Shielding Materials*

Shield	Shield distance from HP RR, m	n/γ at 2 m	n/γ at 3 m
None	-	6.2 ± 2.1^a	6.2 ± 0.9
13-cm steel	2	-	7.8 ± 1.9
5-cm steel	1	-	7.6 ± 1.4
20-cm graphite	1	-	3.2 ± 1.3
20-cm concrete	1	2.6 ± 1.1	2.7 ± 1.1
5-cm steel-15-cm concrete	1	2.7 ± 1.1	2.4 ± 1.4
12-cm Lucite	2	-	1.2 ± 0.1
23-cm Lucite	2	-	0.7 ± 0.2
25-cm polyethylene	1	-	1.0 ± 0.3

*Based on area station dose measurements obtained during seventeen nuclear accident dosimetry intercomparison studies.

^aOne standard deviation about the mean of all measured dose ratios.

exhibited by the Lucite and polyethylene shields is partly due to enhancement of the shielded gamma dose by 2.2-MeV gamma rays from hydrogen neutron capture reactions. This effect is less pronounced for the 23-cm Lucite and 25-cm polyethylene shields relative to the 12-cm Lucite shield due to attenuation effects of the additional material thickness.

Measured neutron-to-gamma dose ratios at 2- and 3-m from the HP RR are shown in Table II for the bare reactor spectrum and eight shields used in the NAD studies. Lucite and polyethylene shields exhibit dose ratios near 1.0 which are significantly lower than that obtained for the unshielded reactor due to high neutron attenuation and relatively low gamma reduction. Steel shields show neutron-to-gamma values higher than that obtained for the bare reactor spectrum. These data are of interest to personnel concerned with accident dosimeters and criticality alarm monitors since these devices are sensitive to both neutrons and gamma rays.

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Rationale for Sprinkler Operation in a High Assay Gaseous Diffusion Cascade, K. R. Yates (ONWI) R. S. Denning, R. G. Jung (BCL)

The X-326 process building, located at the Portsmouth Gaseous Diffusion Plant operated by Goodyear Atomic

Corp., enriches uranium to high assays as required for a variety of U.S. Department of Energy (DOE) programs. Both criticality safety and fire safety are major concerns in the operation of the facility. Objectives of criticality safety and fire safety can, however, lead to conflicting safety measures. For example, the question arises whether an automatic sprinkler system should be valved off over an area in which a uranium deposit exists in cascade equipment, especially if the equipment has to be removed for maintenance activities. The use of water for fire fighting has the potential to adversely affect criticality safety. Alternatively, large fires have occurred previously in other diffusion plants. A large fire in the X-326 building could have considerable consequences. The intent of this study was to compare the increased risk from fires that would result if automatic sprinkler systems were valved off to the decreased risk of accidental criticality. Based on the evaluation, a rationale has been developed as to whether the valving off of the automatic sprinkler system is appropriate.

In developing the rationale, a variety of information had to be reviewed or developed. For fire safety, this included a review of the cause and consequences of previous large cascade fires, a review of regulatory guidance, and an assessment of the value of cascade equipment in areas in which a sprinkler system would be valved off. For criticality safety, this included a review of previous cascade deposits of high assay material in cascade equipment and their causes, a review of regulatory guidance, and an assessment of the time

of sprinkler operation and uranium mass necessary to cause a critical excursion.

From the review and assessment, several things became clear. First, fires can spread very rapidly, and the only fire fighting system capable of handling such fires is an automatic sprinkler system. Second, if a sprinkler system is valved off, a noncompliance situation very likely exists because the value of equipment in jeopardy exceeds DOE guidelines. Third, criticality safety regulations do not prohibit the use of water for fire fighting. Fourth, the time required to cause a criticality excursion is essentially independent of the starting uranium mass in the equipment of concern for large accumulations at very high assays. In other words, the H/U ratio necessary for criticality, over a wide range of mass, changes in such a way that the quantity of water required is about the same in every case. Figure 1 illustrates this situation for several 97% ^{235}U masses, as UO_2F_2 plus H_2O , assumed to have collected in a centrifugal compressor. Calculations were performed using KENO-IV with 123-group AMPX cross sections. For the worst-case situation, exposure of the compressor openings directly to sprinkler water flow, an unlikely situation, ~ 8.1 min is required for sufficient water entry to cause a criticality.

Risk is composed of two aspects: probabilities and consequences. Therefore, maximum consequences were estimated for a large fire or a critical excursion in the X-326 building. From this comparison, the potential consequences of a large fire or critical excursion can be considerable,

Fig. 1. K_{eff} versus H/U for four 97% ^{235}U masses in a centrifugal compressor.

although neither appears to represent a significant threat to the public. However, the potential consequences of a large fire appear more severe than those of a critical excursion.

Finally, the relative probabilities of a large fire or sprinkler initiated critical excursion were estimated. Sprinkler operation can lead to a critical excursion as a result of two situations: the inadvertent activation of the sprinkler system, or the activation of the sprinkler system as a result of a large fire.

The probability of an inadvertent sprinkler head failure can be estimated based on the history of sprinkler head performance over the past 20 years at GAT. Over this period of time, ~100 000 sprinkler heads have been in operation without a single inadvertent failure. Using binomial distribution theory, the probability of an inadvertent sprinkler head failure can be estimated over the next 20-yr period as $4.6 \times 10^{-5}/20$ yr at the 99% confidence level. By combining this probability estimate with the probability of a sprinkler head activating over a piece of equipment (such as a compressor) being removed for maintenance, containing a deposit, and falling such that its openings are exposed to the water for >8 min produces a combined probability of inadvertent sprinkler activation initiated criticality that is very small and can be considered negligible. Furthermore, this probability can be further reduced by simple administrative controls.

In developing a rationale for sprinkler operation, it is therefore only necessary to consider the case in which sprinklers operate because they have been set off by a large fire. In this case, it is not necessary to estimate the absolute probability of a large fire or the absolute probability of sprinkler operation resulting in criticality to establish a reasonable policy. Given the existence of a large fire, it is only necessary to consider accident consequences and conditional probabilities. If there is a large fire, sprinkler operation is essential. If the sprinkler system in the affected region is valved off, the consequences would be very large and probably exceed regulatory guidelines. Even if the probability of such sprinkler operation leading to a criticality were near unity, the consequences would be no greater than for the uncontrolled fire. Realistically, the probability of sprinkler operation leading to a criticality in the event of a large fire is clearly less than unity.

In conclusion, (a) the likelihood of inadvertent sprinkler operation leading to criticality is negligible and can be controlled through administrative controls, and (b) the operation of the sprinkler system is essential to control consequences of a large fire and should be permitted to function; even though some risk of criticality exists, it is not as great as the opposing risk of an uncontrolled fire.

Criticality Analysis of Low-Density Fissile Material in Radioactive Waste, Arlo D. Summers (EG&G Idaho)

An analysis was performed to show that large arrays of waste contaminated with small amounts of fissile materials are critically safe. An analysis technique was adopted that made expensive time-consuming engineering studies of the behavior of the waste over long periods of time unnecessary.

At the Idaho National Engineering Laboratory (INEL), waste is placed either in interim storage on asphalt pads or in permanent disposal in soil excavations. Most of the waste contaminated with small amounts of fissile materials consist of paper, polyethylene wrap, glass, rags, etc., which is usually received as compacted bales or in standardized containers such as DOT 7A wood boxes.

Criticality control depends on fissile material limits placed on each type of waste container. "Worst-case" conditions considered in the development of these limits are

optimum moderation and reflection, geometry of storage configurations, overbatching, and distribution of fissile materials within the waste packages.

The traditional criticality analysis approach of optimizing those parameters affecting criticality that do not have a control placed on them was not feasible for waste contaminated with small amounts of fissile material. In reality, we are certain from the nature of the waste that neutron-absorbing and diluent materials are always present with the fissile material, but it cannot continuously be guaranteed. This is especially true in the disposed condition over long periods of time.

It is not very probable that the small amount of fissile material will separate out from the neutron-absorbing and diluent materials that constitute the waste, migrate to a collection point, and form an optimum moderated and reflected shape that results in a critical system. Instead of performing costly engineering studies to prove this, criticality engineers, operations engineers and management, and the criticality Review Committee (an independent review agency representing Safety Division management) thoroughly reviewed, discussed, and mutually accepted the residual risk.

The assumption adopted was assurance that the fissile material and significant moderators (other waste materials were ignored) are spread out homogeneously throughout two inside dimensions within storage and disposal containers. The third dimension was varied to optimize the balance between neutron moderation and leakage to obtain the highest k_{eff} values. Calculations were performed using the KENO IV Monte Carlo code¹ and the ANISN (Ref. 2) and SCAMP multigroup transport codes, with Hanson and Roach 16-group cross sections.³

Since storage and disposal arrays for the waste evaluated in this study are ~50 X 300 m in cross-sectional area, most of the calculational models consisted of infinite slabs. Except for some of the storage calculations container walls were ignored. The first models (infinite slabs with spacing between) calculated were layers of fissile material simulating layers of waste containers. The spacing between the fuel was determined by the dimensions of waste containers and optimum fuel thickness within the containers. It became apparent that the same k_{eff} values were obtained when fuel was spread out homogeneously throughout a single slab that had a height the same as the array as for arrays with separation. This situation is shown in the model used to simulate the slumping of an array (assumed to slump to half of its original height) in the disposal condition from deterioration of containers. An infinite slab (fuel region) with optimum H_2O moderation and full H_2O reflection resulted in 0.580 k_{eff} for both 3.66- and 7.32-m slab thicknesses at ^{235}U concentrations of 0.353 and 0.177 kg $^{235}U/m^3$, respectively.

Additional calculations consisted of single slabs. An infinite slab with optimum H_2O moderation and full H_2O reflection containing 0.177 kg $^{239}Pu/m^3$ resulted in k_{eff} values of 0.756, 0.828, and 0.934 for slab (fuel region) thicknesses of 7.32, 4.0, and 12.0 m, respectively. A 7.32-m-thick (fuel region) infinite slab with optimum H_2O moderation and full H_2O reflection resulted in k_{eff} values of 0.756, 0.902, and 1.002 for ^{239}Pu concentrations of 0.177, 0.265, and 0.353 kg $^{239}Pu/m^3$, respectively.

For the storage calculations, credit can be taken for container walls. Two sizes of DOT 7A boxes were evaluated. Calculations of a 152.4 X 274.3 X 4.88-m array of wooden boxes containing an equivalent of 177 kg $^{239}Pu/m^3$ (0.0191-m-thick wood walls, full H_2O reflection) resulted in k_{eff} values of 0.589 and 0.434 for 2.13 X 1.22 X 0.61-m and 2.13 X 1.22 X 1.22-m (exterior container dimensions), respectively.

Since polyethylene has been shown⁴ to be a significantly better moderator than water for minimum critical mass

systems, its effects were evaluated. In our low-density fissile material calculational models, there was an insignificant effect on reactivity from polyethylene.

Results of these studies can be used to determine criticality safety limits for large arrays of waste contaminated with low-density fissile material. The analysis was made possible by the proper identification and acceptance of the assumption that neutron-absorbing and diluent materials will not significantly separate from the low-density fissile materials.

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Criticality Safety of an Annular Tank for Fissile Solution, Robert E. Rothe (Rockwell International)

Annular tanks are a viable method of critically safe storage of fissile solutions. A larger annular thickness may be permitted with equal safety if the space interior and exterior to the storage region contains an effective neutron absorber. This experimental program provides criticality data for uranium solution contained in three configurations of annular tanks, each having a variety of boron-loaded earthen materials inside and/or outside.

Six annular tanks were built in four nesting sizes. Each had a 38-mm-thick X 2.13-m-tall annular solution region held 0.3 m above the floor by a skirting. They were built of 6.4-mm-thick Type 304L stainless steel. Outside and inside diameters are as follows: Tank A—1.04 and 0.94 m, Tanks B and C—0.92 and 0.82 m, Tanks D and E—0.81 and 0.71 m, and Tank F—0.69 and 0.59 m.

The uranyl nitrate solution contained 357 g U/l and had a density of 1.5 kg/m³. The uranium was enriched to 93.2% ²³⁵U, with other isotopes as follows: 5.4% ²³⁸U, 1.0% ²³⁴U, and 0.4% ²³⁶U. The solution contained 0.5 M nitric acid and a total impurity content of <1500 ppm. Important neutron absorbers, boron and cadmium, averaged 10 and 30 ppm, respectively.

Fig. 1. Configuration II. Tanks A, B, and D (background) have a 175-mm-thick boron-loaded plaster interior. Tanks C, E, and F (foreground) have a 72-mm-thick interior of the same material. The two closer wood-sheathed slabs contain 50-mm-thick boron-loaded plaster, while the two at the rear of the figure are 19 mm thick.

Earthen materials were ordinary concrete or Hydrostone Super-X^a plaster with minerals colmanite ($\text{Ca}_2\text{B}_6\text{O}_{11} \cdot 5 \text{H}_2\text{O}$) and ulexite ($\text{NaCaB}_5\text{O}_9 \cdot 2 \text{ to } 8 \text{H}_2\text{O}$) added. Boron loadings were ~2.8, ~0.8, or 0.0 wt% boron. Earthen "interiors" (also annular) were 2.13 m tall, with a radial thickness of ~70 or ~170 mm. Flat slabs of boron-loaded plaster (~0.8% boron) were also cast for use outside the tanks. These were 19 or 50 mm thick, and 1.09 X 2.10 m in the other two dimensions.

Each configuration was closely reflected by ordinary concrete wall panels 0.2 m thick X 2.44 m tall. No reflection

existed above the tanks; the floor was also 0.2-m-thick concrete, ~0.3 m below the solution region.

Configuration I had Tanks A, B, D, and F nested and centered in a 1.23-m square concrete enclosure. A variety of interiors were centered within them, with resulting critical heights reported in Table I. No Configuration I experiments had boron-loaded plaster outside the tanks.

The second and third configurations are described in Figs. 1 and 2 (all dimensions in metres). Earthen materials were centered inside and placed outside the nested tanks as described in Table I. Three cases are included where other fissile materials (table footnotes g and h) contact the tank array. These simulate interaction situations as might occur in a production facility. The resulting critical heights for these two configurations are also reported in Table I.

^aRegistered trademark, U.S. Gypsum Company, Chicago, Illinois.

TABLE I
Critical Uranium Solution Heights (mm)

CONFIGURATION I	EARTHEN INTERIOR			CRITICAL HEIGHT IN TANKS (Number of Tanks)				
	Percent Boron	Material	OD x ID (mm)	A,B,D,F (4)	A,B,D (3)	B,D,F (3)	B,D (2)	D,F (2)
	----- no interior -----			272		397		935
	0	cadmium ^a	559 x 0	285		460		b
	0	concrete	559 x 0	259		357		1065
	0	concrete	559 x 415	256		374	1033	795
	0.8	concrete	559 x 415	290		445		b
	0.8	concrete	559 x 210	291		471		b
	2.8	concrete	559 x 415	294		485		b
	0.8	plaster	559 x 415	290	439	464	b	b
0.8	plaster	559 x 210	296		512		b	
0	water	556 x 0	283		435		b	
5.3 g/l	water	556 x 0	304		534		b	
CONFIGURATION II	0.8% BORON-LOADED PLASTER INTERIOR CENTERED IN:					0.8% B PLASTER SLAB ARRANGEMENT	CRITICAL HEIGHT IN TANKS	
	Tanks A,B,C OD x ID (mm)		Tanks D,E,F OD x ID (mm)			A to F (6)	D,E,F (3)	
	no interior		no interior		none	360	448	
	559 x 210		559 x 415		none	417	554	
	559 x 210		559 x 415		See Fig. 1	463	553	
CONFIGURATION III	EARTHEN INTERIOR CENTERED IN: (Code Described in Footnote c)					SEE FOOTNOTE	CRITICAL HEIGHT IN TANKS	
	Tanks A & B		Tanks C & D		Tanks E & F			
	no interior		no interior		no interior			
	2.8,C,762,450		2.8,C,559,415		a,C,559,0		d	
	0.8,C,762,0		0.8,C,610,262		0.8,C,559,210			
	0.8,C,762,0		0.8,P,559,210		0.8,P,559,415			
	0.8,C,762,0		0.8,P,559,210		0.8,P,559,415		f	
	0.8,C,762,0		no interiors		0.8,P,559,415		f	
	0.8,C,762,0		0.8,P,559,210		0.8,P,559,415		g	
	0.8,C,762,0		0.8,P,559,210		0.8,P,559,415		h	

^aThis ordinary concrete interior was sheathed with 1.6-mm-thick cadmium sheet.

^bWell subcritical even with indicated tanks full of uranium solution.

^cThe four-part code in these three columns gives: the boron content in percent, the principal material (C = concrete and P = plaster), and the o.d. and i.d. of the annular earthen cylinder in millimetres.

^dCritical heights were also measured in Tanks A, C, and D (1181 mm) and in Tanks C and D only (1426 mm).

^eThe critical height for a repeat of this run was 1323 mm.

^fSlabs (19 mm thick) of 0.8% boron plaster between each pair of nested tanks.

^gUranium metal sphere (8.2 kg) (93% ²³⁵U) between each pair of nested tanks, 610 mm above the floor.

^hApproximately 7 l of uranium solution (93% ²³⁵U) between each pair of nested tanks.

Fig. 2. Configuration III. Three nested pair of tanks (identified on the figure) had no boron-loaded earthen materials for this experiment. Others had these materials interior to the tanks. A few experiments also had plaster slabs (footnote f on Table I) between pairs of tanks.

Data and Analysis for Nuclear Criticality Safety—II

Calculated K_{eff} Using ENDF/B-V Data for U + Pu Solution Critical Experiments, R. T. Primm III, J. F. Mincey (ORNL)

Effective multiplication factors for 12 critical experiments have been calculated using multigroup cross sections derived from the ENDF/B-V library.¹ All 12 experiments contained mixed plutonium and uranium nitrate solutions. The range of hydrogen-to-fissile plutonium atom ratios spanned by these experiments was 200 to 2200. A comparison with K_{eff} 's calculated with ENDF/B-IV data² is presented.

The most recent version of the Evaluated Nuclear Data File, ENDF/B-V, was released in July 1979. These point data were collapsed to a 227-group library released in June 1980 (Ref. 3). An "in-house" 27-group library was recently prepared at Oak Ridge National Laboratory from the larger, 227-group master library. This 27-group library was used with KENO-IV (Ref. 4) to calculate K_{eff} 's. The AMPX-II computer code module,⁵ NITAWL, was used to perform the Nordheim integral treatment for the resolved resonance region.

Before 1981, 64 critical experiments had been conducted

with mixed plutonium and uranium nitrate solution.⁶⁻⁹ The K_{eff} 's for these experiments have been calculated with KENO-IV and a 27-group library¹⁰ derived from ENDF/B-IV data. Calculated K_{eff} 's were correlated according to hydrogen-to-fissile plutonium atomic ratios, and a first-order, least-squares, Legendre polynomial fit was made to the multiplication factors for calculated configurations not containing boron or gadolinium. Equation (1) is the fit to these 45 unpoisoned experiments:

$$K_{eff} = 1.00775 + 3.90466E-6[H/(^{239}\text{Pu} + ^{241}\text{Pu})] \quad (1)$$

To evaluate the ENDF/B-V-derived library, a subset of 12 experiments was selected from the larger group of 64. These 12 experiments spanned the range of hydrogen-to-fissile plutonium ratios covered by the larger set. The subset included experiments performed with a solution with a high (23 wt% plutonium) ^{240}Pu content as well as experiments conducted with low (5 wt% plutonium) ^{240}Pu solutions. Two experiments performed with soluble gadolinium in the solution were calculated. Both British and U.S. experiments were included in the subset. The group also included experiments conducted in spherical vessels and some conducted in cylindrical tanks.

Fig. 1. K_{eff} calculation of homogeneous U + Pu experiments using KENO-IV and 27-group cross sections from ENDF/B-V.

Fig. 2. Calculated flux spectra using XSDRNPM and two 27-group cross-section libraries.

A second, linear fit was made to the ENDF/B-IV calculations of this subset of experiments only. (Again only unpoisoned experiments were correlated with hydrogen-to-fissile plutonium atomic ratio.) The linear fit is given by Eq. (2). The good agreement between Eqs. (1) and (2) (maximum difference in calculated K_{eff} 's is 0.004 at $H/Pu = 2200$) indicates that the small subset is a representative sample of the entire group:

$$K_{eff} = 1.00978 + 0.969837E-6[H/(^{239}Pu + ^{241}Pu)] \quad (2)$$

Calculated K_{eff} 's using the ENDF/B-V derived cross sections are shown in Fig. 1. A linear fit to these data, defined in Eq. (3), is also shown:

$$K_{eff} = 1.01023 + 1.60003E-6[H/(^{239}Pu + ^{241}Pu)] \quad (3)$$

The linear fit to the ten ENDF/B-IV calculations, Eq. (2), is plotted as a dotted line. All ENDF/B-V calculated values were within two standard deviations of the values calculated

with ENDF/B-IV-derived cross sections. Thus, it is seen that the cross-section data from the more recent ENDF/B-V library does not substantially improve agreement between calculation and experiment.

Agreement in calculated K_{eff} 's does not guarantee that the same flux spectra would be calculated with each library. The one-dimensional, discrete-ordinates code, XSDRNPM (Ref. 5) (also a module of AMPX-II), was used to calculate flux spectra for the experiments with the smallest and largest hydrogen-to-fissile plutonium atom ratios. Figure 2 shows the calculated, volume-weighted spectra. It is evident that good agreement is seen in all groups except the fastest and most thermal.

Calculated K_{eff} 's have been examined for 12 critical experiments conducted with mixed uranium and plutonium solutions. Essentially the same results are obtained with ENDF/B-V cross sections as are obtained with ENDF/B-IV data. Calculated flux spectra for two experiments are also in good agreement.

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Cross-Section Validation for Fuel Storage Pools (Enrichment 4.29 wt%), M. Carotenuto, P. A. Landeyro, F. Siciliano, Tayyab Abbas (CNEN-Italy)

In the Fuel Cycle Department at the Comitato Nazionale per l'Energia Nucleare (CNEN), our group is devoted to nuclear safety verifications for fuel cycle plants. Our work is mainly directed toward reprocessing cells, storage pools, and transport casks. We analyze these facilities from the criticality and shielding points of view.

In criticality studies, it is quite important to have codes and cross-section libraries validated over the whole range of the design field. Taking into account these premises, an important part of our efforts is devoted to the validation of computational tools. The promising results of the AMPX-II 123-group cross-section library for a 2.35 wt% enrichment¹ have encouraged us to extend the validation to higher enrichments.

In this paper, we present and comment on the calculations of benchmark critical experiments on the critical separation between subcritical clusters of 4.29 wt% ²³⁵U-enriched UO₂ rods in water with and without fixed neutron absorbers. These calculations will be used for validating a 123-fine-energy-group cross-section library for analysis of

fuel storage pools. A detailed description of fuel rod, cladding, neutron absorber plates, and the experimental system of the selected cases (identified as experiment numbers 004, 028, 022, 009, 026, 014, 031, and 024) is given in Ref. 2.

The calculations were performed with the KENO IV code employing a 123-fine-energy-group cross-section library and a 16-broad-group Hansen-Roach cross-section library. The master interface of a 123-multigroup library (including 91 epithermal and 32 thermal groups) was generated from ENDF/B-IV data by an XLACS2 (XLACS for hydrogen) module of the AMPX-II system³ and was subsequently processed by the NITAWL module for obtaining the cross sections in AMPX working format. All experiments were calculated using both the discrete fuel rod and smeared rod models, with 109 batches and 500 neutrons per batch, skipping the initial three batches. In the discrete model, the individual rods, including cladding, interstitial water, and aluminum end pieces, are described. In the smeared model, the cluster is taken as a homogeneous mixture of fuel, cladding, and water. The homogenized unit cell average 123-group cross sections produced by the AMPX-II system XSDRNPM module were used in the homogenized cluster region. The homogenized unit cell average 16-group cross sections were produced by ANISN.

TABLE I
Results of Discrete Model Calculations*

Experiment ID	123-Group AMPX-II		16-Group Hansen-Roach	
	K _{eff}	Sigma	K _{eff}	Sigma
004	0.99601	0.00359	1.01884	0.00335
028	0.99742	0.00359	1.01805	0.00341
022	0.99351	0.00352	1.01571	0.00338
009	0.9954	0.00359	1.02102	0.00352
026	0.99058	0.00294	1.01556	0.00344
014	0.99069	0.00314	1.01817	0.00373
031	0.99184	0.00327	1.01497	0.00349
024 ^a	1.00124	0.00362	1.01315	0.00332

*Average K_{eff} 123-group = 0.99459; average K_{eff} 16-group = 1.01693.

^aType of starting distribution = flat.

TABLE II
Results of Smeared Rod Model Calculations*

Experiment ID	123-Group AMPX-II		16-Group Hansen-Roach	
	K _{eff}	Sigma	K _{eff}	Sigma
004	0.99778	0.00387	1.00722	0.00332
028	0.99001	0.00340	1.02202	0.00330
022	0.99523	0.00292	1.02072	0.00304
009	0.99671	0.00332	1.01947	0.00299
026	0.99198	0.00295	1.01498	0.00330
014	0.99781	0.00291	1.02125	0.00319
031	0.98994	0.00348	1.01747	0.00337
024	0.99472	0.00323	1.01624	0.00279

*Average K_{eff} 123-group = 0.99427; average K_{eff} 16-group = 1.01742.

The results of discrete and smeared model calculations are summarized in Tables I and II, respectively.

The 123-fine-group discrete and smeared model calculations give average multiplication factors of 0.99459 and 0.99427, respectively. This shows that the discrete and smeared model results are in good agreement within the 0.0055 statistical error limits. The 16-broad-group discrete and smeared model calculations give average multiplication factors of 1.01693 and 1.01742, respectively. The discrete model average multiplication factor value obtained using 123-fine-group cross sections is lower than the critical value by $\sim 0.55\%$ (in terms of K_{eff}), whereas 16-broad-group discrete model calculations give an average K_{eff} higher than the critical value by 1.7%.

The discrete model calculations using 18-broad-group cross sections indicate a conservative bias of $\sim 1\%$ in terms of K_{eff} (see Ref. 2). From this comparison, it appears that the criticality calculations with the KENO IV code using 123-fine-group cross sections give more accurate results for the systems simulating fuel storage pools.

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2. S. R. BIERMAN, B. M. DURST, and E. D. CLAYTON, "Critical Separation Between Subcritical Clusters of 4.29-wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," NUREG/CR-0073, Battelle-Pacific Northwest Labs. (1978).

3. "AMPX-II Modular Code System for Generating Coupled Multigroup Neutron and Gamma Ray Cross Section Libraries from Data in ENDF Format," ORNL-PSR-63, Oak Ridge National Lab. (1978).

Comparisons of Experiments and Calculations for LWR Storage Geometries, B. F. Cooney, T. R. Freeman, M. H. Lipner (Westinghouse NES)

This paper presents the results of a comparison between measured and calculated reactivities for two sets of experimental data. One set of data¹ was generated for the U.S. Department of Energy (DOE) at the Babcock and Wilcox Lynchburg Research Center, where various critical assemblies were constructed simulating close-packed light water reactor (LWR) fuel storage configurations. The second set of data^{2,3} was generated for the U.S. Nuclear Regulatory Commission (NRC) at the Battelle Pacific Northwest Laboratories (PNL). Criticality calculations were performed using the AMPX-KENO code package developed by the Oak Ridge National Laboratory (ORNL). The results of the calculations are compared with the experimental data to verify the methodology.

The Westinghouse calculational methodology employs the AMPX (Ref. 4) modules NITAWL and XSDRNPM for cross-section generation and KENO-IV (Ref. 5) for reactivity calculation. The energy group structure employed was the

TABLE I
Benchmark Calculation Results

DOE Criticals			PNL Criticals	
Spacing, cm	Core	ΔK_{eff}^a	Experiment	ΔK_{eff}
0.0	I	-0.008	005	-0.010
	II	-0.012	012	-0.005
	III	-0.008	015	-0.013
1.636	IV	-0.012	016	-0.001
	XI	+0.002	017	-0.012
	XIII	-0.007	020	-0.006
	XIV	-0.013	021	-0.009
	XV	-0.016	022	-0.014
	XVII	-0.015	026	-0.010
	XIX	-0.016	027	-0.010
	V	-0.011	028	-0.003
	VI	-0.011	029	-0.013
3.272	XII	-0.013	031	-0.011
	XVI	-0.015	032	-0.012
	XVIII	-0.020	033	-0.013
	XX	-0.011	034	-0.014
	VII	-0.011	041	-0.005
	VIII	-0.013	043	-0.005
	X	-0.008	044	+0.001
	XXI	-0.017	047	-0.005
	IX	-0.016	048	-0.004
4.907				
6.543				

^aCalculated-measured.

27-group structure⁶ developed by ORNL for mixed-oxide applications. The 27-group cross sections were generated through NITAWL and XSDRNPM from the ORNL 218-group master library.⁷ A corrected version of NITAWL was used in which the scattering self-shielding in the resonance region was approximated by the absorption self-shielding. Resonance self-shielding was provided for in ²³⁵U, ²³⁸U, and ⁵⁵Mn. Individual Dancoff factors were also input for edge, corner, and interior pins. A unit cell XSDRNPM was then used to group collapse the libraries with resonance parameters. The KENO-IV representation of the DOE criticals explicitly modeled each fuel pin, each B₄C pin, the isolation sheets, the moderator, and the core tank in the X-Y directions. The fuel and support structure above the water was not modeled. The KENO-IV representation of the NRC criticals explicitly modeled each fuel pin, the poison plates, and the moderator.

With this calculational model, values of K_{eff} were obtained that seem to be insensitive to spacing (see Fig. 1) and average ~ 0.99 for each spacing between arrays. In results previously reported,¹ K_{eff} was underestimated when

the separation between arrays was greater than one pin pitch, and the degree of underestimation increased with wider spacing. Because the KENO models employed in both studies were essentially identical, the only differences between the methods are in the cross-section library and method of generation.

Also included in this critical experiment evaluation was a group of 21 criticals from experiments done at PNL. These 21 criticals were performed using LWR-type enrichments and geometries. The results of these calculations display a high degree of consistency when compared to the results for the DOE criticals. The NRC criticals provide an average value of K_{eff} of 0.990, while the DOE criticals provide an average value of 0.989. The results of both sets of Westinghouse calculations are tabulated in Table I.

The results of the Westinghouse calculations indicate that, with ENDF/B-IV cross sections and a resonance scattering self-shielding approximation, there is a consistent bias between experiment and prediction of $\sim 0.01 \Delta K$. This bias is not sensitive to changes in the spacing between fuel arrays, the enrichment of the fuel rods, the water-to-fuel

Fig. 1. Benchmark calculation results.

volume ratio or the presence of boron in the water. On this basis, these calculations provide verification for the use of the AMPX-KENO methodology for criticality calculations.

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3. S. R. BIERMAN et al., "Critical Separation Between Subcritical Clusters of 4.29 wt% ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons," PNL-2615, Battelle-Pacific Northwest Labs. (Mar. 1978).
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Reactivity Sensitivity of a Poisoned High-Density PWR Spent Fuel Storage Rack as a Function of Inside Cavity Opening, C. Ho, R. L. Becker, R. J. Weader,* J. R. Fisher, E. D. Kendrick (Nucl Assoc Int)

The design of a high-density poisoned storage rack for pressurized water reactor (PWR) spent fuel assemblies requires careful consideration of the opening clearance of the inside of the canister. At a minimum, the opening must be

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Fig. 1. Typical PWR spent fuel rack geometry (diagram not to scale).

larger than the actual physical size of the fuel bundle, plus sufficient tolerance for bowing, such that a bundle can be easily slipped in and out of a cavity. At the other end of the spectrum in cavity clearance design, the choice is much less clear, and the dimensions are often not carefully chosen. Studies performed by Nuclear Associates International (NAI) show that the reactivity of a spent fuel rack is sensitive to the inside clearance of a cavity. The difference in subcritical reactivity margin can vary 3% in Δk for a 0.2-in. variation in cavity clearance.

Figure 1 shows a typical PWR storage rack design. The

assembly is surrounded on all four sides by four poison plates sandwiched between two concentric stainless steel canisters. Depending on the design of the rack, the poison compartment formed by the inside and the outside canisters may be either flooded with water or voided. The rack, when installed in the storage pool, is completely submerged in water that fills all the open spaces of the rack.

An intercavity clearance sensitivity study was performed by NAI using a four-group diffusion theory model. The basic cross-section data and other cell-dependent parameters were generated by CHEETAH (Ref. 1), which is a modified

Fig. 2. Reactivity and absorption fractions as a function of cavity clearance.

LEOPARD code. The constants representing the poison material (B_4C) were generated by the CORC code,² utilizing the equivalent diffusion parameter theory. The sensitivity results were calculated by PDQ-7 (Ref. 3), which has been demonstrated to be a reliable tool in calculating finite perturbations in a typical SFP analysis.

Figure 2a shows the pool reactivity as function of the canister inside dimension. The results show that the reactivity has a strong dependency on the inside canister opening when all other factors or parameters are the same, and the correlation is essentially linear in the range useful for spent fuel rack designers.

The results presented in the following figures display the source of this behavior. Figure 3 gives the pointwise neutron

flux for the third energy group ($0.625 \text{ eV} \leq E_n \leq 5.530 \text{ keV}$) and the thermal neutron group ($E_n < 0.625 \text{ eV}$). Case A represents the design with a smaller can opening (8.50 in.) and Case B, a larger one (8.70 in.). The neutron spectrum is slightly harder in Case A because less edge water is present in the supercell calculation. The large water gap in between two adjacent Case A cavities produces a larger thermal flux trapped in that water space that, in turn, enhances the neutron absorption by both the water and the adjacent Boral plates. Figures 2b and 2c give the normalized Boral absorption fractions as a function of canister inside opening or cavity clearance. The absorption fractions for both groups 3 and 4 decrease as the canister inside opening increases. Also shown in the same figures are the absorption fractions

Fig. 3. Pointwise neutron flux versus distance from center of fuel assembly.

in the fuel, which take an opposite trend because of the neutron spectra in the fuel.

In conclusion, from a criticality point of view, the inside opening of a storage cavity for a PWR spent fuel rack should be made as small as physically possible, taking into consideration the requirement for bowing and seismic effects. A smaller inside canister opening will provide a significant benefit in pitch or poison loading reduction or a combination of both. This logic does not in general apply to boiling water reactor (BWR) spent fuel rack design because the center-to-center spacing requirement for BWR cavities is generally smaller, and most designs have a common poison plate shared by two adjacent cavities. Therefore, no large water gap exists between two adjacent cavities to act as a neutron flux trap.

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3. W. R. CADWELL, "PDQ-7 Reference Manual," WAPD-TM-678, Bettis Atomic Power Lab. (Jan. 1967).

Analysis of Critical Experiments with Monte Carlo and ENDF/B-V Data, *Laurance D. Eisenhart, Michael Todosow (BNL)*

Calculations of critical experimental benchmarks are often used to qualify a specific computational approach for the solution of a class of problems whose characteristics are similar to those of the experimental configuration. Comparison of the calculated and experimental results helps to quantify the deviation of the calculated result from experimentally measured "truth." This paper presents the results of Monte Carlo simulations with two state-of-the-art Monte Carlo programs, SAM-CE (Ref. 1) and RCP01 (Ref. 2), using ENDF/B-V nuclear data, for two critical experiments that are typical of configurations encountered in light water reactor fuel storage pools.³ The results are also compared to previously published results obtained during the validation of the SCALE system.^{4,5}

The experiments that were simulated involved the determination of the critical separation between three clusters of UO_2 rods arranged in a linear array, both with and without 1.1 wt% borated stainless steel absorber plates located

between the clusters.³ The clusters consisted of 15 X 8 arrays of 4.31 wt% enriched UO_2 rods (clad in aluminum) at a lattice pitch of 2.54 cm. With this fuel and lattice spacing, a critical configuration can be achieved with a single 8 X 16.37 ± 0.03 cluster. The configurations that were examined therefore represent a loosely coupled system, and are a stringent test for any calculational approach.

The analyses were performed with two continuous energy Monte Carlo programs, SAM-F (the Monte Carlo component of the SAM-CE system) and RCP01, using nuclear data based on ENDF/B-V. The cross-section library for SAM-F was obtained from the National Nuclear Data Center. The experimental configurations were modeled in detail with both programs (i.e., fuel-gap-cladding, absorber plates, and the surrounding water-filled tank). The structural materials within the tank and the tank itself were not explicitly modeled, but this omission is expected to have a negligible effect on the calculated results. Approximately 20 000 neutron histories were processed for each simulation.

The results obtained for the critical configuration in the absence of the poison plates and with the borated stainless steel plates present are presented in Tables I and II, respectively. The SAM results utilize the Maximum Likelihood Estimator.⁶ The number of neutron histories processed per minute on a CDC-7600 are presented for the SAM and RCP results as a measure of the efficiency of the Monte Carlo simulation. Results obtained with KENO-IV (from Ref. 5) using three cross-section libraries (27-group ENDF/B-IV, 123-group GAM-THERMOS, and 16-group Hansen-Roach) are also presented. The KENO calculations involved the processing of ~30 000 neutron histories, but no information is available on the computing times.

The major results of this study can be summarized as follows:

1. Both SAM and RCP, with ENDF/B-V nuclear data, perform well relative to experiment.
2. RCP is approximately twice as fast as SAM (for comparable accuracy in the simulation for the unpoisoned critical).

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TABLE I

Monte Carlo Results for Simulation of Unpoisoned Critical

Method	$K_{eff} \pm 1\sigma$	Number of Neutron Histories min
RCP01	0.9996 $\pm$ 0.007	1413
SAM-F	1.0052 $\pm$ 0.007	694
KENO-IV		
Library 1 ^a	0.989 $\pm$ 0.005	Not available
Library 2 ^a	0.984 $\pm$ 0.004	
Library 3 ^a	0.997 $\pm$ 0.004	

^aKENO-IV libraries are all multigroup (results are from Ref. 5):

Library 1—27-group ENDF/B-IV
Library 2—123-group GAM-THERMOS (ENDF/B-II)
Library 3—16-group Hansen-Roach.

TABLE II

Monte Carlo Results for Simulation of Poisoned Critical

Method	$K_{eff} \pm 1\sigma$	Number of Neutron Histories min
RCP01 ^a	1.003 $\pm$ 0.009	1404
SAM-F	0.996 $\pm$ 0.004	734
KENO-IV		
Library 1 ^b	0.982 $\pm$ 0.004	Not available
Library 2 ^b	0.987 $\pm$ 0.004	
Library 3 ^b	1.001 $\pm$ 0.004	

^aResults following processing of 9500 neutron histories.

^bKENO-IV libraries are all multigroup (results are from Ref. 5):

Library 1—27-group ENDF/B-IV
Library 2—123-group GAM-THERMOS (ENDF/B-II)
Library 3—16-group Hansen-Roach.

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Criticality of Normal Compacted Spent Fuel,
S. Y. Hong, P. E. Netusil, D. R. Harris (RPI),
R. R. Calabro, D. S. Reed (Northeast Util)

Compaction of spent fuel offers potential benefits in storage and shipment. A number of nuclear power plant operators already have increased the capacities of their spent fuel pools by reracking for closer storage of fuel assemblies.¹ Considerable further increases in spent fuel storage could be achieved by removing the fuel pins from their assemblies and storing the pins more compactly in (a) storage canisters² or (b) storage skeletons.³ Spent fuel racks and shipping casks already are licensed to be quite subcritical ($K_e < 0.95$) when ordinary fuel assemblies are present. These are somewhat water-deprived, and compacted fuel can have an even lower water-to-fuel ratio. Consequently, storage and shipment of compacted fuel is potentially safe.

It must be considered, however, that because of pin distortion and other problems, the spent fuel pins may not load uniformly into the canisters or skeletons. Thus, criticality analyses must be carried out for off-nominal pin configurations in canisters or skeletons spanning the range of possible cases, some possibly more reactive than ordinary fuel assemblies. This analysis is described here for normal compacted spent fuel, where normal signifies that the pins are confined in normally handled canisters or skeletons, although possibly not in uniform close-packed arrays. A companion paper treats accident situations.⁴

Specific analyses were carried out for compacted storage of Westinghouse Electric Corp. fuel pins in the Millstone-2 spent fuel pool⁵ and for shipment of these pins in the General Electric Co. (GE) IF-300 shipping cask.⁶ The fuel analyzed had an enrichment of 3.35 wt% and dimensions of 0.19025, 0.00375, and 0.026 in. for fuel radius, air gap thickness, and cladding thickness, respectively. Compacting operations were hypothesized to take place in a region of the Millstone-2 pool from which a rack section would be removed. The top-end fitting would be cut away from a fuel assembly, fuel rods removed using a remote pulling device, and the fuel rods loaded into a $\frac{1}{8}$ -in. Type 304S stainless steel canister (8.2-in. square i.d.), which itself would fit into a Millstone-2 rack cavity ($\frac{1}{4}$ in. thick). Maximum hexagonal close packing would result in 368 fuel rods in the canister compared with 176 fuel rods in the usually stored assembly. If the canister were tipped during loading, the rods would assume an 18 X 18 square array of 324 rods at maximum packing. It is more likely, however, that the rods be less than close packed, so the off-nominal conditions to be analyzed range from 18 X 18 to 8 X 8, including the most reactive condition near 12 X 12. This range includes pin configurations in storage skeletons.

Criticality analyses were carried out using KENO-IV (Ref. 7) Monte Carlo calculations, and few-group PDQ (Ref. 8) diffusion calculations were benchmarked against these. The KENO-IV calculations employed a 123-group cross-section set that has sufficient fine groups to model thermal and fast neutron disadvantage effects, but in the

TABLE I
 Multiplication Factors for Westinghouse Fuel Pins Compacted in Stainless Steel
 Canisters in the Millstone-2 Pool and in a GE IF-300 Shipping Cask

Square Pin Array	Pitch (in.)	Effective Multiplication Factor k_e			k_∞
		Millstone-2 Pool KENO-IV	PDQ-7	IF-300 Shipping Cask PDQ-7	
18X18	0.4540	0.704 ± 0.007	0.7172	0.6253	1.1732
16X16	0.5125	0.805 ± 0.008	0.8169	0.7464	1.3305
15X15	0.5461	---	0.8511	0.7881	1.3800
14X14	0.5857	0.855 ± 0.008	0.8742	0.8192	1.4126
12X12	0.6833	0.878 ± 0.009	0.8875	0.8473	1.4370
10X10	0.8200	---	0.8597	---	1.3820
8X8	1.025	---	0.7643	---	1.2495

important resonance range, effective group cross sections were determined by NITAWL (Ref. 9). The Nordheim resonance self-shielding calculations in NITAWL require an input Dancoff factor, which usually is determined from Sauer's fit to Fukai's Monte Carlo two-region fuel-moderator calculations.¹⁰ The approximate treatment of cladding in Sauer's expressions is more important for close-packed rods, so Fukai's cases were reanalyzed by Monte Carlo and extended to include cladding. Changes in the Dancoff factor were found to be <10%, with accompanying changes in K_e being <1%.

KENO-IV calculations for the Millstone-2 rack are shown in Table I. The PDQ results also shown in Table I utilized three-group cross sections prepared by CASMO (Ref. 11) and EPRI-CELL (Ref. 12). The IF-300 shipping cask utilizes poison pins to ensure subcriticality, and effective three-group cross sections for these were obtained from Ref. 6. Values of k_{∞} are listed in Table I for reference purposes. Satisfactory agreement was found between the PDQ and KENO-IV values of K_e and with previously reported results for fuel assemblies.^{5,6} The range of cases listed in Table I is sufficiently subcritical that even when other uncertainties are included, $k_e < 0.95$. For properly compacted fuel, $k_e < 0.75$. It is inferred that normal compacted fuel does not present a criticality hazard in the Millstone-2 racks and in the IF-300 shipping cask, even if the pins were to become loosely arrayed in the canister or skeleton.

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Accident Analyses of Compacted Spent Fuel, S. Y. Hong, P. E. Netusil, D. R. Harris (RPI), R. R. Calabro, R. A. Isakson (Northeast Util)

Compaction of spent fuel offers the potential of considerably increased capacity for licensed and proposed spent fuel pools and shipping casks. The compaction process consists of cutting away or otherwise removing the top-end fitting from the spent fuel assembly, pulling the fuel rods, and reloading the fuel rods into a metal canister¹ or skeleton.² The compacted fuel may be hexagonal close-packed or square close-packed in the canister, but it is likely

a. Near-Cavity Spill

b. Gap-Center Spill

Fig. 1. Geometries for two configurations of spilled fuel pins.

that rod bowing and other effects will result in looser and more irregular arrangements. It is shown in a companion paper³ that such arrangements are quite subcritical ($k_e < 0.95$) for Westinghouse Electric Corp. fuel rods (3.35 wt% enriched) compacted in stainless steel canisters in the Millstone-2 spent fuel pool⁴ and in the General Electric Co. IF-300 shipping cask.⁵ Accident analyses for compacted spent fuel are described here.

Accident analyses for storage and shipment of ordinary fuel assemblies take into account possible rearrangements of entire assemblies. An assembly might be dropped onto the top of the rack, for example, or four assemblies might be closer than nominal because of manufacturing imperfections in the rack or cask. For compacted fuel, it is additionally necessary to take into account the possibility that fuel pins may become rearranged accidentally. We find that

Fig. 2. K_{eff} versus number of pins spilled for various spilled pin pitches.

certain fuel pin rearrangements indeed can become supercritical, but that these can be prevented by suitable mechanical modifications.

Compaction operations were hypothesized to take place in a region of the Millstone-2 pool from which a section of rack might be removed. Fuel pin rearrangement then could occur in the compaction area, in the pool, or in the shipping cask. The most severe condition appears to result from spilling the contents of a canister (for example, by upending) above the pool. Conceivably, the pins could rearrange between the rack cavities in the pool, thus adding their own reactivity as well as coupling the previously decoupled fuel regions. The Millstone-2 spent fuel cavities are on a 12.3125-in. pitch and are sufficiently spaced that fuel pins could drop between them. Survey estimates based on the values of k_{∞} listed in Ref. 3 indicated potentially hazardous conditions. These were verified by a series of PDQ-7 (Ref. 6) calculations. (The PDQ-7 three-group method was described and verified against Monte Carlo results in Ref. 3.) The PDQ-7 calculations assumed that 104, 232, or 368 (the maximum number of hexagonal close-packed rods in a single canister) fuel pins become rearranged into square arrays near a cavity or between cavities in the Millstone-2 pool (Fig. 1). The cavities were taken to be filled with uncompacted fuel or with 18 X 18 compacted fuel. The PDQ-7 calculations employed a nine-cavity repeating array in most cases (Fig. 1). This approximation was confirmed by calculations for a 25-cavity repeating array, which was found to diminish k_e by only 0.4%. Moreover, a zero flux rather than zero current boundary condition on the nine-cavity problem decreased k_e even less. The results of these calculations are shown in Fig. 2. As few as 100 spilled pins can present a hazardous condition ($k_e \approx 0.95$). Similar analyses were carried out for the shipping cask, assuming that all baskets but one were already filled. These results indicated that k_e could exceed 0.95 if certain poison pins were

knocked away and if the contents of an entire canister were present in the empty basket and between baskets. In actuality, however, the spacer disks in the IF-300 cask prevent the pins from spilling into the interbasket regions.

It is concluded that fuel pin rearrangements such as those treated here must be prevented. This can be accomplished by several means. Spacers between the cavities in spent fuel pools and between the baskets in shipping casks can exclude fuel pins from these regions. Such spacers can add to the mechanical strength required by the additional mass of compacted fuel. In addition, the integrity of fixed poisons and of the canisters and skeletons can be ensured by careful mechanical design.

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Standards for Nuclear Criticality Safety

ANS/ANSI Standards, *Dixon Callihan*

This contribution will address the production of voluntary national standards under the processes developed and administered by the American National Standards Institute (ANSI). Illustrative examples will be largely drawn from the nuclear standards program.

A Standard is defined as "A prescribed set of rules, conditions, or requirements concerning definitions of terms; classification of components; specification of materials, performance, or operations; delineation of procedures; or measurement of quantity and quality in describing materials, products, systems, services, or practices. The word 'standard' does not include federal, state, or local laws or statutes enacted to adopt or reference a standard."

A national standard is one reasonably expected to have a significant effect on a substantial number of U.S. citizens.

National standards perhaps most familiar to the general public are those in photography and building construction. The light sensitivity of photographic film is unified by national standards establishing the well-known film speed as, for example, ASA 64 (ASA denotes American Standards Association, a predecessor of ANSI). National standard ANSI C1-1981 recommends good electrical wiring practices that become legally binding and enforceable when adopted into a building code.

A standard has its genesis with the identification of a need followed by the preparation of a proposed draft by a group of knowledgeable individuals representative of all concerned with the subject—producers, manufacturers, users, regulators, etc. Technical societies have traditionally been these producing bodies. Once prepared to the satisfaction of the writing group, the proposal is shown to have achieved a consensus through one of several prescribed methods.

Final action is by the Board of Standards Review (BSR), an ANSI arm, which first puts the document before the public for an extended review and comment period and lastly, after the address and reconciliation, by the writing group of any comments, approves the document as a national standard. Regardless of the acceptance and use of any standard, it must be formally acted on again with five-year intervals following its first adoption or be withdrawn.

Within nuclear criticality safety, standards are proposed and drafted by American Nuclear Society Standards Subcommittee ANS-8 and its several writing groups. That a consensus is represented is established by National Standards Committee N16. Thence, as noted above, proposals go to the Board of Standards Review. Comments and criticisms are welcomed at every stage of development. Consideration of comments must be established in the record. Although unanimity is not a requisite, every reasonable attempt shall be made to reconcile differences.

ANS Standards and Regulatory Guides, *Norman Ketzlach (NRC)*

The U.S. Nuclear Regulatory Commission (NRC) sets broad policy to protect the public health and safety and to

establish performance requirements for the nuclear industry. This paper is devoted to the NRC's interest in standards for nuclear criticality safety. Section 70.22, "Contents of Applications," of 10 CFR Part 70 ("Domestic Licensing of Special Nuclear Material") of the Code of Federal Regulations requires that applications for a specific license to own, acquire, deliver, receive, possess, use, or transfer special nuclear material contain, among other things, proposed procedures to avoid accidental conditions of criticality. To serve as a regulatory tool for facilitating the licensing process, the NRC develops regulatory guides that provide guidance for complying with this portion of the Commission's regulations by describing procedures acceptable to the NRC staff for the prevention of criticality accidents. The American National Standards Institute (ANSI) standards for nuclear criticality safety serve as the basis for most of the regulatory guides in this area.

After a standard has been approved by ANSI, it is reviewed by the NRC for acceptance in the regulatory process. The NRC may accept a standard as written, accept it with exceptions, or not adopt it at all in the preparation of regulatory guides. In adopting a standard in a regulatory guide, the NRC is notifying the license applicants that it will use the method described in the guide in evaluating a license application for compliance with specified portions of the Commission's regulations unless the applicant proposes an acceptable alternate method. A guide provides a convenient reference for the licensee or license applicant in designing nuclear criticality safety programs to meet the regulatory requirements and standards of good nuclear criticality safety practices.

The NRC has endorsed in regulatory guides seven of the nine standards in nuclear criticality safety approved by N16. Table I lists the standards incorporated in guides.

The relationship between ANS-8 (Standards Subcommittee for "Fissionable Material Outside Reactors"), N16 (American Standards Committee for "Nuclear Criticality Safety"), and the regulatory guides of the NRC should be noted. ANS-8, sponsored by the American Nuclear Society (ANS) Nuclear Criticality Safety Division, is the subcommittee that develops the standards. N16, the American National Standards Committee for which ANS holds the secretariat as defined by the Nuclear Standards Management Board of ANSI, is the consensus body for processing and approving national standards. The NRC is the regulatory body that develops guides used in evaluating license applications for compliance with the Commission's regulations.

It is obvious from the above that cooperative relationship is necessary for the development of standards and guides to define methods for good nuclear criticality safety practices and the establishment of acceptable methods for their implementation. The NRC participates at all levels of the standards development process. Members of the NRC staff have participated, as individuals and as members of Subcommittee ANS-8, in the development of standards for determining the potential for nuclear criticality of fissionable material outside reactors, for the prevention of accidental criticality, and for coping with accidents should they occur. The NRC is represented on National Standards Committee

TABLE I
Standard-Regulatory Guide Relationship

Standard No.	Title	Regulatory Guide No.
1. ANS-8.1/N16.1-1975	Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors	3.4
2. ANS-8.3-1979	Criticality Accident Alarm System	8.12
3. ANS-8.5-1979	Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material (Revision of N16.4-1971)	3.1
4. ANS-8.7/N16.5-1975	Guide for Nuclear Criticality Safety in the Storage of Fissile Materials	3.43
5. ANS-8.9-1978	Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate	3.45
6. ANS-8.11/N16.9-1975	Validation of Computational Methods for Nuclear Criticality Safety	3.41
7. ANS-8.12-1978	Nuclear Criticality Control and Safety of Homogeneous Plutonium-Uranium Mixtures Outside Reactors	3.47

N16. Lastly, the NRC has a liaison representative on the ANS Standards Committee, which is comprised of six consensus bodies, including N16. This working relationship between the NRC and the committees responsible for the development and approval of nuclear criticality safety standards has been in existence almost since their inception.

To expedite the development of standards and advance good practices in the application of nuclear criticality safety in fuel cycle facilities, the NRC has funded a number of research projects. These included a project for the development and publication of the "Nuclear Safety Guide,"¹ development of a better understanding of neutron interaction between storage and processing units in a nuclear fuel facility,^{2,3} and the development of a standard for the administrative practices in nuclear criticality safety.

Thus, the NRC has been working together with all levels of the standards developing process to implement good nuclear safety practices in the nuclear industry.

1. "Nuclear Safety Guide," Rev. 2, NUREG/CR-0095, U.S. Nuclear Regulatory Commission (June 1978).
2. J. T. THOMAS, "Solid Angle and Surface Density as Criticality Parameters," NUREG/CR-1615, U.S. Nuclear Regulatory Commission (Oct. 1980).
3. J. T. THOMAS, "The Effect of Reflector Location on Array Criticality," NUREG/CR-1616, U.S. Nuclear Regulatory Commission (Nov. 1980).

Overview of Standards Subcommittee 8 and Criticality Safety Administrative Practices—ANS-8.19, J. D. McLendon (Union Carbide, Nucl Div)

OVERVIEW

American Nuclear Society (ANS) Standards Subcommittee 8 has worked continuously for the past 20 years to prepare and promote, through the ANS/ANSI consensus process, standards on nuclear criticality safety for the handling, processing, storing, and transport of fissionable materials outside of reactors. The historic perspective, functions, scope of activities, personnel resources available, and accomplishments are briefly presented. ANS-8, "Fissionable Materials Outside Reactors," is sponsored by the Nuclear Criticality Safety Division.

ANS-8 membership is as follows:

J. D. McLendon, Chairman, Union Carbide Corporation
 E. B. Johnson, Secretary, Oak Ridge National Laboratory
 F. M. Alcorn, Babcock & Wilcox Company
 H. K. Clark, Savannah River Laboratory
 E. D. Clayton, Pacific Northwest Laboratory
 D. M. Dawson, General Electric Company
 Norman Ketzlach, U.S. Nuclear Regulatory Commission
 W. G. Morrison, Exxon Nuclear Idaho Company, Inc.
 David R. Smith, Los Alamos National Laboratory
 J. T. Thomas, Oak Ridge National Laboratory
 G. E. Whitesides, Oak Ridge National Laboratory
 F. E. Woltz, Goodyear Atomic Corporation

Subjects for criticality safety standards may arise from any source in which a sufficient need is identified. In the past, topics for consideration have been suggested by members of ANS-8, ANSI Subcommittee N16, the ANS Standards Steering Committee, criticality safety specialists, and others. To accomplish preparation of draft standards, ANS-8 utilizes work groups led by interested and dedicated group leaders. Because of direct sponsorship and support by the Nuclear Criticality Safety Division of ANS, a broad base of national and international criticality safety expertise is available to draw upon in accomplishing assigned tasks and developing consensus on each project.

ANSI N16 is the body responsible for determining that an appropriate cross section of those having interest have had the opportunity to review and have input. To expedite the establishment of consensus, it is common practice within ANS-8 to send copies of drafts in late and final forms to N16 and other recognized groups, organizations, or individuals for their early input.

Following establishment of consensus for approval by ANSI N16, the proposed standard is submitted to ANSI Board of Standards Review for final approval as an American National Standard. When this last step is accomplished, the Standards Department of ANS undertakes the publication of the new American National Standard. Availability of new standards is included on a sales list that contains ~2500 addresses. In addition, announcements are carried in *Nuclear Standard News*, *Nuclear News*, and in the *ANSI Reporter*. Copies of newly published standards are distributed to all members of ANS Information Center on Nuclear Standards (ICONS), a subscription service.

Nine approved and published standards have been prepared by this Subcommittee. They are:

- ANS-8.1/N16.1-1975 Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
- ANS-8.3-1979 Criticality Accident Alarm System
- N16.3-1975/ANS-8.6 Safety in Conducting Subcritical Neutron-Multiplication Measurements *in Situ*
- ANS-8.5-1979 Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material
- N16.5-1975/ANS-8.7 Guide for Nuclear Criticality Safety in the Storage of Fissile Materials
- ANS-8.9-1978 Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate
- N16.8-1975/ANS-8.10 Criteria for Nuclear Criticality Safety Controls in Operations Where Shielding Protects Personnel
- N16.9-1975/ANS-8.11 Validation of Calculational Methods for Nuclear Criticality Safety
- ANS-8.12-1978 Nuclear Criticality Control and Safety of Homogeneous Plutonium-Uranium Fuel Mixtures Outside Reactors

Currently, 12 writing groups are active under the guidance of ANS-8. Of these, six are concerned with extension or mandatory maintenance of existing standards (ANS-8.1, 8.6, 8.7.1, 8.9, 8.10, and 8.12.1).

American National Standards resulting from the work of ANS-8 have been recognized and incorporated into guides and regulations of the U.S. Department of Energy, the U.S. Nuclear Regulatory Commission, the International Atomic

Energy Agency, and the International Organization for Standardization.

The extensive and consistent participation of nuclear criticality safety specialists, both within the United States and from foreign countries, gives ANS-8 standards broad recognition and acceptance in the specialty field of nuclear criticality safety outside reactors.

ADMINISTRATIVE PRACTICES FOR NUCLEAR CRITICALITY SAFETY-ANS-8.19

This standard provides criteria for the administration of a nuclear criticality safety program for operations outside of reactors in which there exists a potential for criticality accidents. Responsibilities of management, supervision, and the nuclear criticality safety staff are addressed. Objectives and characteristics of operating and emergency procedures are included.

Work group membership is as follows:

- D. R. Smith, Chairman, Los Alamos National Laboratory
- Dixon Callihan, Union Carbide Corporation (retired)
- R. W. Carson, Babcock & Wilcox Company
- J. D. McLendon, Union Carbide Corporation
- H. C. Paxton, Los Alamos National Laboratory (retired)
- J. T. Thomas, Oak Ridge National Laboratory

Effective nuclear criticality safety programs include cooperation among management, supervision, and the criticality safety staff and rely on the knowledgeable conformance with operating procedures by all employees. Although the extent and complexity of safety-related activities may vary greatly with the size and type of operation with fissile material, certain safety elements are common. This standard represents a codification of such elements related to criticality safety.

The standard addresses these elements:

1. responsibilities of management, supervisors, and staff specialists
2. general nature and content of operating procedures
3. guides to planned response to criticality accidents.

International Standards and Nuclear Safety, David R. Smith (LANL)

International standards provide many virtues, services, and advantages. The development of an international standard may avoid, or at least reduce, duplicative effort on the part of many national standards bodies. In that safety standards in particular are frequently developed from experience, the broader base of experience existing internationally may be expected to lead to a better product. Consensus standards facilitate cooperative international ventures and international trade while enhancing trust and understanding among nations.

Development of international standards is carried out through one of two organizations. The International Electrotechnical Commission (IEC) is responsible for worldwide standardization in the electrical and electronics fields. The Commission operates through some 70 technical committees and more than 100 subcommittees. About 100 000 electrical and electronics experts participate annually in IEC standardization programs.

Technical Committee 45 of IEC has as its scope nuclear instrumentation. Subcommittee 45A deals with reactor instrumentation, while Subcommittee 45B is concerned with radiation protection instrumentation. A writing group sponsored by Subcommittee 45B is currently developing a Recommendations for Warning Equipment for Criticality Accidents.

The IEC produces consensus "recommendations" that are intended to provide the basis for national standards and are frequently adopted verbatim for this purpose.

Membership in the IEC is held by 43 national committees representing ~80% of the world's population. The organization is financed by dues from national committees and by the sale of IEC publications.

The International Organization for Standardization (ISO) has a broader scope than the IEC, essentially being concerned with "everything else." The ISO comprises some 160 technical committees, 600 subcommittees, and 1200 working groups.

Technical Committee 85 of ISO concerns itself with nuclear energy and has four subcommittees.

SC 1: Terminology, Definitions, Units, and Symbols

SC 2: Radiation Protection

SC 3: Power Reactor Technology

SC 5: Nuclear Fuel Cycle.

SC 5 has seven working groups, one of which (WG 6) has for several years been developing a draft standard for criticality accident alarm systems. This effort has now reached the status of a "Draft Proposal"—DP7753. It is now being distributed for ballot and, if the result is successful, will become a Draft International Standard for a period of time before being redesignated as an International Standard.

Meeting in May 1981, Subcommittee 5 of ISO/TC 85 acted to initiate a writing group that would address "Standardization of Calculations, Procedures, and Practices Related to Criticality Safety." If this proposal is endorsed by the member countries participating in SC 5, it is expected that the American National Standards Institute (ANSI) will serve as convenor for this new writing group.

Membership of ISO now consists of 88 countries, each represented by a member body that is "most representative of standardization in its country." ANSI serves as the U.S. member body for ISO, as well as being the U.S. National Committee for IEC.

Close cooperation exists between ISO and IEC, even to the point of sharing a common address in Geneva. Both have close ties with the agencies of the United Nations, which demonstrate significant interests in international standardization. Strong efforts are made to avoid duplication of efforts and to prevent conflicts or inconsistencies between the products of ISO and IEC. As an example, the chairman of the ISO/TC 85/SC 5 writing group developing the standard for criticality accidents alarm systems was designated to provide liaison to IEC/SC 45B/WG 5, developing the standard for Warning Equipment for Criticality Accidents. This was very effective in enhancing the consistency of these two complementary efforts.

The role of international standards is gaining increased recognition, with such standards frequently preceding national developments. While standards writing is tedious work, it can also be very rewarding. Persons interested in such activities should make themselves available to the standards committees of their technical or professional organizations.

The American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, *Hugh K. Clark (SRL)*

The American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors is a basic standard that contains general guidance for the prevention of criticality accidents. The guidance is in the form of required and recommended administrative

and technical practices and the specification of "maximum subcritical limits" for some simple systems. The limits incorporate margins sufficiently large to compensate for uncertainties in calculations, but contain no additional safety factor. The standard has an appendix, not officially part of the standard, that contains a listing of the types of contingencies that need to be considered in satisfying the standard's requirement that before beginning an operation, it shall be determined that the operation will be subcritical under normal and credible abnormal conditions. A number of other standards have been written to amplify particular aspects of nuclear criticality safety considered too specialized for this basic standard, such as the use of borosilicate glass Raschig rings, and references are made to all such standards at appropriate points.

The standard was originally drafted in 1958, and after several revisions, was proposed as American National Standard N6.1-1962 under the joint sponsorship of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers. (it was eventually issued as ASA N6.1-1964.) In June 1963, a work group of ANS-8 (Subcommittee 8 of the Standards Committee of ANS) was appointed to undertake a revision. One of the criticisms of the original version was its lack of numerical values. Whether these should be critical values was the subject of much discussion, but it was decided that maximum subcritical limits would be worth more to the user. The work group produced its first draft in October 1963, and by June 1965 had modified it some six times in response to comments. Rather than submitting written comments, some members of ANS-8, who objected to the approach being taken by the work group, prepared an alternative draft in October 1965. The two drafts were reconciled at a meeting held in February 1966. After minor revisions, the standard was finally issued as American National Standard N16.1-1969. The mandatory five-year review of the standard was begun in January 1974. Minor revisions were made in wording, and the concentration limit for ^{239}Pu solutions was reduced from 7.3 to 7.0 g/l, because support for the larger limit was considered inadequate. The revision was issued as American National Standard N16.1-1975.

The American National Standard for Validation of Calculational Methods for Nuclear Criticality Safety is an amplification of the requirement in the basic standard that calculational methods be shown to be valid by comparison with experimental data. An example of a validation for a fictitious system is given in its appendix. The work group that developed this standard was formed in 1969, and its efforts culminated in the issuance of the Standard as American National Standard N16.9-1975.

As 1980 approached, preparations were made for the mandatory review of both standards. Proposals for improvements in the ANSI-N16.1 (henceforth to be termed ANS-8.1) Standard were circulated to the members of ANS-8 in June 1979 and elicited little comment. At the November 1979 meeting of ANS-8, the suggestion was made to incorporate the validation standard in the general standard. A draft of the combined standard was circulated to ANS-8 for comments in June 1980. This time more comments were forthcoming. A draft attempting to reconcile the comments was prepared in September and was circulated to ANS-8 in October. Ballots on this draft were distributed to ANS-8 in March. The ballots were returned with comments, the comments were resolved at a June meeting of ANS-8, and the standard was forwarded to American National Standards Committee N16 for processing as an American National Standard.

An important contribution to this revision was the examination of limits in the standard and the proposal of slightly more restrictive limits where there were doubts as

to subcriticality under stated conditions. This work was instigated, in part, by doubts expressed by McNeany and Jenkins¹ as to the subcriticality of some of the ²³³U limits. Additional limits were provided where it was felt they would be useful. The basis for the revised and additional limits is being published in the open literature. It was considered essential that this documentation conform to the calculational methods validation portion of the standard and serve as an example of the validation process.

1. S. R. McNEANY and J. D. JENKINS, "An Evaluation of ENDF/B-IV and Hansen-Roach Uranium-233 Cross Sections for Use in Criticality Calculations," *Nucl. Sci. Eng.*, 65, 441 (1978).

ANSI/ANS-8.3-1979—American National Standard Criticality Accident Alarm System, *David R. Smith (LANL)*

In two of the eight criticality accidents that are known to have occurred in processing fissionable material, lives were saved because criticality accident alarm systems were installed. The American National Standard Criticality Accident Alarm System provides guidance regarding the need for alarm systems, characteristics of such systems, maintenance and test programs, alarm characteristics, and emergency planning.

The Standard calls for a survey of each activity involving more than threshold quantities of fissionable materials and the installation of an alarm system wherever it is judged that such a system will provide a reduction in total risk. Here, the Standard recognizes that false alarms occur and thus introduce the possibility of injury to those who respond to the alarm signal.

It is required that the alarm signal be used for no other purpose than immediate evacuation, and this signal shall be audible in all areas that are to be evacuated. A means of manual initiation of the alarm is acceptable, and visual signals may be used to supplement the audible signal in areas of very high background noise. Simple designs are endorsed to encourage reliability, and high radiation levels are not to render the system inoperable.

Among the more significant features of this Standard is a definition of a minimum accident of concern. Based on accident experience and consideration of accident mechanisms, the radiation level to be detected from such an accident is stated as an absorbed dose in free air of 20 rad at 2 m from the reacting medium within 60 s. This corresponds to a dose rate of 3.3 mGy/s, and it is required that such a dose rate shall be detected immediately. Here, "immediately" is usually interpreted as meaning a time less than human reaction time. This dose rate specification assumes the absence of significant shielding and corresponds roughly to a yield of 10^{15} fissions in the first minute of a reaction. This detection criterion, coupled with the alarm trip point, permits an evaluation of the radius of coverage of a detector.

It is recommended that concurrent response of two detector channels be required to initiate an alarm. This is intended to reduce the frequency of false alarms. A further requirement, that failure of any single detector channel not prevent compliance with the detection criterion, results in having each point in a process area within the radius of coverage of three detectors.

Test requirements are specified. It is recommended that the response of each channel to radiation be determined at least monthly and that the entire system be tested periodically. Audible signal generators are to be tested at least once every three months, with a determination that the signals be audible above background for all areas that are to be evacuated.

Test procedures should be established to minimize false alarms and to ascertain that the system is in normal operation after the test.

One section of the Standard addresses emergency procedures. This was included because no other standard addressed this subject, and it was believed that if an alarm system is necessary, such procedures are also appropriate.

Two appendixes are associated with, but not part of, the Standard. The first supports the selection of the detection criterion and the definition of the minimum accident of concern. The second appendix provides sample calculations of the radius of coverage for a detector.

This Standard presents a comprehensive set of criteria for the need, design, installation, and use of criticality accident alarm systems.

Borosilicate-Glass Raschig Rings as Neutron Absorbers—ANS-8.5, *Norman Ketzlach (NRC)*

A number of American National Standards Institute (ANSI) standards have been prepared that provide guidance for the prevention of criticality accidents in the handling, storing, and processing of fissionable materials. One of these standards, ANSI/ANS-8.5-1979, provides guidance for the use of borosilicate-glass raschig rings in preventing accidental criticality in processing solutions of fissionable materials. The standard provides acceptable fissile concentrations in homogeneous solutions of uranium and plutonium in vessels of unlimited size packed with raschig rings having specified characteristics. In the absence of the rings, solutions can become critical at relatively low concentrations of fissile material. The maximum subcritical concentrations of fissile material as a function of isotopic content are given in Table I for unpoisoned aqueous solutions.

The standard specifies the chemical and physical environment, properties of the rings and packed vessels, maintenance inspection procedures, and acceptable fissile concentration limits. The concentration limits are 400 g ²³⁵U/l for highly enriched ²³⁵U solutions, 200 g ²³³U/l for 100 wt% ²³³U solutions, and 200 g Pu/l for plutonium solutions. In the presence of the rings, no concentration of ²³⁵U in solution enriched to ≤ 5 wt% ²³⁵U can be made critical.

Although the general use of fixed neutron absorbers, including raschig rings, for criticality control in fuel processing with solutions of fissile materials dates back to 1958, some applications were recorded as early as the mid-1940s. The first attempt to codify in a single document a series of recommendations derived from experience and technology on this subject was in 1965 (Ref. 3). The proposed standard was reviewed by the American Nuclear Society (ANS) Standards Committee and submitted to ANS members and

TABLE I
Maximum Subcritical Concentration

Fissile Solute	g Fissile Concentration/l
²³⁵ UO ₂ (NO ₃) ₂	11.5 ^a
²³³ UO ₂ (NO ₃) ₂	10.8 ^a
²³⁵ U(s)O ₂ (NO ₃) ₂	~14.0 ^b
Pu (NO ₃) ₄	7.0 ^a

^aFrom Ref. 1.

^bFrom Ref. 2.

other interested persons for comment. In the next five years, a study was made gathering together the background material, references, and supporting data for the proposed standard.⁴ Additional experiments were also performed in support of the proposed standard. After modification of the proposed standard for consistency with the data, it was adopted as an ANSI Standard in 1971 (Ref. 5). The Standard concerned only highly enriched uranium and plutonium solutions and was limited to an acidic or neutral environment.

After the prescribed five-year review was made of the 1971 Standard, it was revised.⁶ Besides reaffirming the application of the 1971 Standard, the revision provides guidance for the use of the rings as a secondary criticality control and for their application to caustic solutions (under limited conditions) and aqueous solutions of uranium containing ≤ 5.0 wt% ^{235}U (Ref. 7).

Large tanks packed with borosilicate-glass raschig rings have been and are being used as an alternative to geometrically safe tanks for solution storage. These tanks accommodate large volumes of solution more efficiently than long, limited-diameter cylinders or thin slab-like containers. In addition to primary criticality control, as for in-process storage, raschig rings in auxiliary tanks may be used to protect against accidental criticality resulting from inadvertent transfer of fissile solution to those tanks. They are also used in sumps and on floors beneath large vessels to prevent inadvertent criticality should the vessels develop leaks.

The present Standard sets a limit of 2 W/g on the equivalent radiation exposure rate to the glass by absorption of energy from alpha particles. This restriction is based on available data³ on the effects of alpha radiation on borosilicate glass. Light water reactors may generate plutonium containing up to 3.5% ^{238}Pu . Since ^{238}Pu generates ~ 0.56 W/g, the plutonium solutions could generate more than 2 W/g. Therefore, additional experimental data will be needed to determine the structural integrity of the borosilicate glass under these conditions. If the rings are to be exposed to other more severe environments than those specified in the standard, additional data will be necessary to determine the chemical and structural integrity of the rings. A more severe environment may also call for more frequent maintenance inspections.

The fissile concentration limits specified in the Standard should be adequate for all postulated solution processing operations. It is unlikely that the allowable concentration limits would unduly restrict solution processing operations.

In summary, the Standard provides guidance for the storage of large volumes of solutions of fissile materials.

1. "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI N16.1-1975.
2. R. D. CARTER, G. R. KIEL, and K. R. RIDGWAY, *Criticality Handbook* (May 23, 1969).
3. "Use of Borosilicate-Glass Raschig Rings as a Fixed Neutron Absorber in Solutions of Fissile Material," *Nucl. Eng. Bull.*, **3** (Nov. 1965).
4. J. P. NICHOLS, C. L. SLINSKI, with D. W. MAGNUSON, "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," Y-CDC-8 (July 1971).
5. "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," ANSI N16.4-1971.
6. "Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," ANSI/ANS-8.5-1979.

7. N. KETZLACH, "Proposed Extension of Raschig Ring Standard to Lower Enriched Uranium Fuel," *Nucl. Technol.*, **42**, 65 (1979).

Safety in Conducting Subcritical Neutron Multiplication Measurements *In Situ*—ANS-8.6, H. C. Paxton (LANL)

The standard "Safety in Conducting Subcritical Neutron Multiplication Measurements *In Situ*" refers to measurements of fissile configurations without benefit of shielded or remote operation. The purpose is to safely establish or confirm subcriticality and to approximate the margin to criticality. It codifies practices developed at the Rocky Flats Plant, mostly in the decade from 1955 to 1965, and used at critical facilities for stacking parts of critical assemblies by hand. These practices are directed toward maintaining, with confidence, a safe margin below criticality in the course of meaningful measurements.

This standard, designated ANS-8.6, ANSI N16.3-1975, is undergoing a second revision and is expected to be republished in 1982. (Original publication was in 1969.) The current revision should clarify the original intent without changing requirements.

Required administrative practices include written procedures, reviewed by an experienced experimenter and approved by management; a minimum of two experimenters, one experienced in subcritical or critical experiments; records with emphasis on safety; and emergency procedures.

Equipment criteria call for verification of mechanical integrity of equipment and proper functioning of instrumentation prior to the experiment, a neutron source, two independent neutron-counting devices, and a signal indicating neutron level, either audible or otherwise apparent. It is proposed that "for example, a flashing light" be added to the last item.

Operational practices require suspension of the experiment if doubt of safety is expressed or there is unexpected behavior, adherence to a reactivity limit defined in the written procedure, and plots of reciprocal neutron multiplication from at least two neutron-detector channels, maintained independently by at least two persons. A proposed addition to the last requirement is: "These plots shall have a sufficient number of points to provide confidence in extrapolation." Further requirements are that reactivity additions be guided by the plots of reciprocal neutron multiplication ("reciprocal" is a proposed addition) and authorized by the person assigned primary safety responsibility (this person is to be identified more clearly); the method of reactivity addition will neither interfere with extrapolation of reciprocal multiplication plots nor be accomplished by change of spacing in arrays; inadvertent reactivity additions are to be avoided; source and detector locations will be such that observed neutrons are predominantly from the fissile assembly, and that effects of changed source-detector location or sensitivity be measured before further reactivity addition. It is proposed that the last item be made a requirement instead of a recommendation.

The early *in situ* measurements at Rocky Flats and some at Los Alamos applied to plant equipment such as melting crucibles and associated molds and vessels packed with raschig rings for solution storage. They also applied to shipping containers and to storage arrays of fissile units including weapon components and assembled weapons. This type of measurement for plant criticality guidance became relatively infrequent after the Rocky Flats critical facility began operation in about 1965. It is still used, however, in some plant process areas where equipment and surroundings

are too complex to be mocked up in a critical facility. Of course, the standard still applies frequently to setups in critical facilities.

ANSI/ANS-8.9-1978—Criticality Safety Pipe Intersection Standard Review and Applications, C. M. Hopper (Union Carbide, Nucl Div)

INTRODUCTION

Basic parameters and guidance for nuclear criticality safety control outside reactors are described in American National Standards Institute Standard N16.1, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors." Since N16.1 provides general guidance and limited single-unit parameter limits, the American Nuclear Society (ANS) Standards Subcommittee 8 developed the ANSI ANS-8.9-1978 standard, "Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate." Four years have elapsed since the acceptance and publication of the pipe intersection standard, and it is appropriate that the standard and its development be reviewed, the applications be highlighted, and current efforts for further development be explained.

EXPLANATION OF THE STANDARD

The pipe intersection standard provides general design guidance for limiting intersecting pipe geometries containing homogeneous aqueous solutions of uranyl nitrate in which the uranium is enriched to 93.5 wt% ^{235}U . The standard is specific to a single, unlimited length large-diameter pipe (column) intersected by numerous lesser diameter pipes (arms) of lengths not to exceed ten times the column diameter, all of which may be surrounded by a partially or fully reflected environment.

DEVELOPMENT AND TECHNICAL BASES OF THE STANDARD

In 1968, members of the Criticality Safety Division within ANS Subcommittee 8 appointed a work group to examine the status of data and develop the scope of a proposed standard for pipe intersections. In 1970, the work group limited the scope of the standard to 93.5 wt% enriched uranyl nitrate solutions due to the paucity of critical experiment data available for pipe intersections containing solutions of fissile material.

By the end of 1976, several critical experiments had been performed^{1,2} to validate numerous computer calculations,³⁻⁵ no less than two analytical models^{6,7} had been developed for pipe intersections, and seven draft standards had been balloted for acceptance. On January 26, 1977, the work group successfully obtained written confirmation of consensus acceptance and submitted the draft for the publication approval process.

ANSI/ANS-8.9-1978 was approved and published by the American National Standards Institutes on February 15, 1978. Since the issuance of the 1978 standard, work has continued on the standard to expand its application to other solutions of fissile material and variable configurations of pipe intersections within differing neutron-reflecting environments.⁸ Since the original issuance of the standard, no less than five draft revisions have been considered.

APPLICATIONS OF THE STANDARD

The applications of the pipe intersection standard are self-evident. Wherever highly enriched uranyl nitrate solutions are collected in manifolds for transfers and storage, batch pour-up stations are consolidated into feed columns,

etc., the standard may be applied. Limitations for the application of the standard are clearly delineated and include the following:

1. The fissile material shall be a homogeneous aqueous solution of uranyl nitrate. The ^{235}U content shall be no more than 93.5 wt%.
2. The intersection areas of arms with the central column shall not exceed tabulated values.
3. Any portion of the system of intersecting pipes having moderating material among the arms shall be considered as fully reflected. The minimum surface separation of arms shall be 80 mm (3 in.).
4. Pipes <300 mm (1 ft) from a concrete surface shall be considered fully reflected.
5. Independent and validated analyses must be performed on systems of intersecting pipes having multiple arm lengths greater than ten times the central column.

FUTURE STANDARD DEVELOPMENTS

The future efforts of the ANS-8.9 work group include the extension of the current standard to

1. any homogeneous aqueous solution of uranium and plutonium in standard steel pipes of schedule 10 or thicker wall
2. specific information for uranium solutions having a ^{235}U content not exceeding a 5 wt% of the total uranium, solutions of plutonium, ^{233}U , and highly enriched uranium.

SUMMARY

It is readily observed that the pipe intersection standard can be useful in the design of solution pipe intersection systems. The review of the standard and its development efforts (past and present) provide an understanding of the "standard effort," its dynamic character, and the opportunity for individuals to enhance the standard with constructive comments or suggestions.

1. J. K. FOX, L. W. GILLEY, and D. CALLIHAN, "Critical Mass Studies, Part IX, Aqueous ^{235}U Solutions," ORNL-2367, Oak Ridge National Lab. (1958).
2. E. B. JOHNSON, "The Nuclear Criticality of Intersecting Cylinders of $\text{U}(5)\text{O}_2\text{F}_2$ Aqueous Solution," *Trans. Am. Nucl. Soc.*, **14**, 678 (1971).
3. DEANNE DICKINSON, "Calculations for Pipe Intersections Containing Fissile Solution," RFP-1499, Dow Chemical U.S.A., Rocky Flats Division (1970).
4. N. F. CROSS, G. E. WHITESIDES, and R. J. HINTON, "Monte Carlo Analysis of Experimentally Critical Pipe Intersections," *Trans. Am. Nucl. Soc.*, **17**, 268 (1973).
5. DEANNE DICKINSON, "Nominally Reflected Pipe Intersections Containing Fissile Solution," *Nucl. Technol.*, **26**, 265 (1975).
6. B. B. ERNST and C. L. SCHUSKE, "Empirical Method for Calculating Pipe Intersections Containing Fissile Solutions," RFP-1197, Dow Chemical U.S.A., Rocky Flats Division (1968).
7. DEANNE DICKINSON and C. L. SCHUSKE, "An Empirical Model for Safe Pipe Intersections Containing Fissile Solution," *Nucl. Technol.*, **10**, 179 (1971).
8. J. T. THOMAS, "Reflectors, Infinite Cylinders, Intersecting Cylinders, and Nuclear Criticality," *Nucl. Sci. Eng.*, **67**, 279 (1978).

Nuclear Criticality Safety in Shielded Facilities—ANS-8.10, Bryan F. Gore (PNL)

This joint American National Standards Institute/American Nuclear Society (ANSI/ANS) criticality safety standard applies to facilities constructed and operated so that personnel would be protected from the effects of an accidental criticality if it occurred. When specified protection criteria are met, reduced conservatism in equipment design and process operation is allowed compared to operations where protection is not provided.

A major factor in the achievement of the excellent criticality safety record of the nuclear industry has been broad application of the "double-contingency" principle of criticality control. This principle requires that equipment, processes, and operations be so designed and carried out that two independent and unlikely "contingencies" (accidents or limit violations) must occur before criticality can result. The standard ANS-8.1 (not ANS-8.10) recommends general use of the double-contingency principle in criticality control. However, it permits the use of less stringent control when facilities provide protection to personnel. ANS-8.10 defines the protection criteria that must be met, and permits "single-contingency" criticality control in facilities meeting these criteria.

Facilities such as fuel reprocessing plants and certain fuel fabrication plants that handle highly radioactive materials provide protection in the forms of thick shielding against direct radiation and ventilation systems that prevent release of radioactive material from the plant. ANS-8.10 applies to any facilities that meet its protection criteria. This standard recognizes that in such facilities the "... consequences of criticality accidents are primarily damage to equipment, disruption of processes, spillage of product, etc." When this is the case, the standard states that, "Accordingly, one may be willing to accept an increased risk of accident when compensating process benefits accrue. This may be reflected in reduced conservatism in the process analysis. However, plant design and operations should be based on good engineering practices, which dictate that criticality not occur under normal operating conditions or likely abnormal conditions."

ANS-8.10 allows facilities that meet its protection criteria to base criticality prevention on certain single primary methods of control that have a low probability of failure. This is single-contingency control. Examples of acceptable primary criticality controls include:

1. geometry and spacing between fissile process materials
2. isotopic composition of fissile material
3. concentration of fissile material in solutions
4. concentration of soluble neutron absorbers in fissile solution.

It should be noted, however, that more restrictive controls are required where improper performance of a routine operation might lead to criticality. In such situations, the standard requires that "... procedures shall be established to ensure that no single administrative error on the part of any one individual leads to criticality."

The criteria that a facility must satisfy to qualify for single-contingency operation are based on the radiation dose that might result from an accidental criticality. The shielding and the ventilating system of an acceptable facility must "... be such that the whole body radiation dose received by an individual located at any point outside the shield will not exceed 25 rem (Ref. 1) following a criticality accident." (This may be compared to fatal doses, estimated

at $\sim 10^4$ rem, received during two of the very few criticality accidents that have actually occurred.²) In addition, the standard also requires that "The quantity of fission products or fissionable material which may be released from the facility should not exceed the limit specified by the regulation applicable to such an accidental release." Facilities meeting these criteria may use single-contingency criticality control.

Work on this standard was begun in 1968 by a work group containing representatives of industry, research, and government organizations. The first draft was submitted for comment in June 1979. Twelve subsequent drafts were prepared over a five-year period, some in response to recommendations from the American National Standard Committee N16. Subcommittee 8 of the ANS Standards Committee approved this standard in 1974. It was then adopted by the ANSI in May 1974 as ANSI N16.8-1975/ANS 8.10, "Criteria for Nuclear Criticality Safety Controls in Operations Where Shielding Protects Personnel."

ANSI standards must be reviewed and reaffirmed every five years. ANS 8.10 is presently undergoing review for the first time. This review has produced a recommendation to change the criterion concerning the quantity of material that might be released from a qualifying facility due to an accidental criticality. A new criterion has been proposed stating, "In addition, the shielding and confinement shall be such that the whole body radiation dose received by an individual outside the restricted area surrounding the facility will not exceed 0.5 rem (Ref. 3)." The 0.5-rem dose corresponds to the total calendar year whole-body dose limit specified in 10 CFR 20 for unrestricted areas around a nuclear facility. The basis for this proposed revision is the philosophy that a reduction in design conservatism should only be allowed when the consequences of accidental criticality are indeed primarily confined to plant operations or personnel.

The proposed revised standard is presently undergoing balloting by ANS Standards Subcommittee 8. Present indications are that it is likely to be adopted. If so, it will be forwarded to the ANSI Standard Committee N16 for consideration.

1. "Basic Radiation Protection Criteria," NCRP Report No. 39, p. 101, National Council on Radiation Protection and Measurements (Jan. 15, 1971).
2. H. C. PAXTON, "Criticality Control in Operations with Fissile Material," LA-3366 (Rev.), Los Alamos National Lab. (Nov. 1972).
3. "Standards for Protection Against Radiation," 10 CFR 20, Sec. 20, 105, U.S. Code of Federal Regulations (June 15, 1979).

Use of the Solid Angle Method in Nuclear Criticality Safety, F. M. Alcorn (B&W)

Work Group ANS-8.13.1 of American Nuclear Society (ANS) Standard Subcommittee 8 is currently writing a consensus standard for use of the solid angle technique. The Standard will state the requirements for establishing a solid angle method and the criteria for application of that method. The solid angle method may then be used to demonstrate the nuclear criticality safety of an array of individually subcritical units of fissionable material.

The original solid angle method was developed¹ at the Oak Ridge Gaseous Diffusion Plant and reported in 1956 (over 25 years ago). The area of application was initially for small planar arrays of highly enriched uranium solution.

Because of the relatively simple nature of the method, the ease of application, and the belief that the method was conservative in nearly all situations, the original method has been widely used outside its originally intended area of application. Several variations of the method have evolved over the years and are in use. The description of one solid angle method² and guidance for its use are available on pp. 97-99 of TID-7016, Rev. 2.

The solid angle model relates the neutron multiplication factor of a uniform array of units to the multiplication factor of the units and the total solid angle subtended at a unit by all other units of the array. The solid angle method in application is an empirical procedure. The use of such is permitted by the American National Standard for Nuclear Criticality Safety, N16.1-1975, provided the procedure has been validated. The American National Standard N16.9-1975 outlines the requirements for establishing the validity and area(s) of applicability of such a calculational method. This standard on solid angle will permit use of the solid angle technique in compliance with the requirements of N16.1 and N16.9.

Work Group ANS-8.13.1 initially sought to provide a generally applicable solid angle method resulting in known subcritical configurations. While the development of a generally applicable model is possible, it would, in all likelihood, prove impracticably restrictive. Describing and providing a unified justification for the various solid angle methods then in use appeared to present an insurmountable task and would also prove overly restrictive by not providing for new variations in the solid angle model. The current thrust of the work group is the one that will surely be of maximum utility to industry: that thrust is to permit each user to develop and use a solid angle method tailored to his needs.

Use of a solid angle method developed according to requirements of the standard will permit allowances to be incorporated for the shape of individual units, array shape, vessel walls, interspersed moderation, reflector material, and reflector location. Of special concern to those who make use of the solid angle technique is the problem of "mixed arrays," i.e., arrays with units of different fissionable material, different shape, different size, different orientation, etc. The standard will provide guidance in dealing with application of the solid angle method to "mixed arrays." The standard will contain a bibliography of pertinent background and reference documents.

Work group ANS-8.13.1 at the current time is composed of Wilbur Goodwin, Westinghouse Electric Corp.; Bill Peters and Dave Dawson, General Electric Co.; Ed Walker, Bechtel Incorporated; Bruce Ernst, American Nuclear Insurers; Joe T. Thomas, Oak Ridge National Laboratory; Norm Ketzlach, U.S. Nuclear Regulatory Commission; and Fred Welfare, Babcock and Wilcox. Francis Alcorn of Babcock and Wilcox is chairman. The work group plans to submit a draft for ballot by the ANS Standards Subcommittee 8 before the winter 1981 meeting of ANS.

In summary, this paper has summarized efforts of ANS-8.13.1 to develop an American National Standard on the use of the solid angle method. The thrust of the standard will be on presenting criteria necessary to developing and applying specific qualified solid angle procedures tailored to the user's needs.

1. H. F. HENRY, J. R. KNIGHT, and C. E. NEWLAN, "General Application of Theory of Neutron Interaction," K-1309, Oak Ridge Gaseous Diffusion Plant (Nov. 15, 1956).
2. J. T. THOMAS, Ed., "Nuclear Safety Guide TID-7016, Revision 2," pp. 97-99, NUREG/CR-0095, U.S. Nuclear Regulatory Commission (June 1978).

Use of Solutions of Neutron Absorbers for Criticality Control—Part I: ²³⁵U-Boron, W. G. Morrison (Exxon Nuclear Idaho), R. D. Carter (Rockwell Hanford), R. C. Lloyd (PNL)

An American National Standard on the use of soluble neutron absorbers for criticality control (ANS-8.14) has been prepared to supplement N16.1-1975 (Ref. 1), in particular, by providing subcritical limits for aqueous homogeneous mixtures of ²³⁵U and boron.

Part I is applicable to the use of soluble boron for criticality control of homogeneous aqueous solutions of ²³⁵U with no limit on the enrichment of the uranium in ²³⁵U. The fissile material may be present as a solid or as a precipitate in macroscopically uniform slurries, provided the H/²³⁵U atomic ratio does not exceed that of an aqueous solution having the same fissile material concentration. However, Part I does not provide details or specifications for neutron absorber monitors or for administrative control on the process operations. Those requirements are assigned to and established by the user of this method of criticality control; this allows requirements to be precisely fitted to the specific design and operation of the process or handling facility.

Part I presents subcritical limits as a function of the fissile material concentration (in grams per litre) in homogeneous aqueous solution at constant neutron absorber concentration; this is presented as a family of curves for mass, volume, cylinder diameter, and slab thickness. Limits are presented for ²³⁵U solutions containing as much as 8.0 g of boron per litre. The boron covered in Part I is naturally occurring boron containing ≥ 17.9 wt% ¹⁰B. The curves are calculated on the basis of homogeneous mixtures of ²³⁵U and water with infinite water reflector. Subcriticality assurance is provided by using a value of 0.95 for the value of the *k*_{eff}. Figure 1 presents subcritical mass limit curves. Other curves are presented in Part I for use in cylindrical or slab geometries or with limited volume. The limiting subcritical ²³⁵U concentration in an infinite system corresponding to the boron concentration in the ²³⁵U mass curve is presented in Part I (and can also be determined as the asymptotic concentrations from Fig. 1).

When applying the limits to mixtures of slurries or precipitates, an H/²³⁵U atomic ratio curve is presented in Part I for use in determining that the H/²³⁵U atomic ratio for such mixtures does not exceed that of an aqueous solution having the same fissile material concentration. In all cases, the neutron absorber must be uniformly mixed with the slurry or precipitate mixture. These limits, however, are not to be used for any mixtures or combinations of ²³⁵U and ²³³U or uranium and ²³⁹Pu.

The limits or maximum subcritical values for boron poisoned U(100)-water solutions were determined through a series of parameter calculations using the DTF-IV code with GAMTEC-II cross-section sets.² Validations were made with eleven bare sphere critical experiments: five ²³⁵U nitrate, and six ²³³U nitrate. The fissile material densities ranged from 13 to 28 g per litre, and boron densities ranged from zero to 0.23 g per litre. Other calculations were made with unpoisoned solution experiments: with U(93.2)O₂F₂ at 831 g ²³⁵U per litre and with ²³³U nitrate at 160 g ²³³U per litre. Accuracy of the results calculated for both poisoned and unpoisoned experiments indicate that the poison effect was calculated satisfactorily. Once the accuracy of the correlations was sufficiently proven, the DTF-IV code with the GAMTEC-II cross-section sets was used to calculate a series of maximum subcritical (*k*_{eff} = 0.95) parameters for U(100)-water solutions; these were determined for uranium densities ranging from 20 to 1000 g per litre and for boron ranging from 0 to 8 g per litre.

Fig. 1. Subcritical mass limits as a function of boron and uranium concentrations in spherical geometry for ^{235}U - H_2O mixtures with an infinite water reflector.

As affirmed by ANS-8.14 Part I, soluble neutron absorbers can be readily used as the primary criticality control in all phases of work with large inventories of fissile material. In other aspects, the use may be as a secondary criticality control to guard against accidental criticality incidents. Part I of ANS-8.14 provides the general guidance for achieving the desired criticality control with ^{235}U -water solutions by use of soluble neutron absorbers. It is planned that ANS-8.14 will eventually cover other neutron absorbers such as cadmium and gadolinium in other fissile materials such as ^{233}U and plutonium.

1. ANS-8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI N16.1-1975.
2. R. D. CARTER, "Subcritical Limits for Boron-Poisoned ^{235}U - H_2O Solutions," RHO-SA-63, Rockwell Hanford Operations (1978).

Criticality Control of Special Actinide Elements—ANS-8.15, John E. Bigelow (ORNL), Gordon E. Hansen (LANL)

Most standards for nuclear criticality safety deal with ^{235}U and, to a lesser degree, ^{233}U and ^{239}Pu . However, increasing production of other transuranium nuclides that can be made critical has necessitated consideration of the fissile properties and critical masses of these special materials. A work group (Table I) was formed to draft a new standard that would be an extension of the standard for criticality safety in operations outside of reactors (N16.1-1975/ANS-8.1). Subcritical mass limits for 14 new nuclides (Tables II and III), were developed; however, the major application of this new standard is expected to be in setting operational limits in facilities that handle six important nuclides, namely, ^{237}Np , ^{238}Pu , ^{240}Pu , ^{241}Pu , ^{241}Am , and ^{244}Cm .

Neptunium-237 is of importance as a means of producing

TABLE I

Work Group ANS-8.15 of Subcommittee 8 on
Fissionable Materials Outside Reactors

E. D. Clayton, Chairman	Pacific Northwest Laboratory
H. K. Clark	Savannah River Laboratory
C. R. Richey	Pacific Northwest Laboratory
D. R. Smith	Los Alamos National Laboratory
G. E. Hansen	Los Alamos National Laboratory
J. T. Thomas	Oak Ridge National Laboratory
R. M. Westfall	Oak Ridge National Laboratory
J. E. Bigelow	Oak Ridge National Laboratory
F. E. Woltz	Goodyear Atomic Corporation

TABLE II

Subcritical Mass Limits for Fissile* Actinide Nuclides

Nuclide	Mass Limit of Nuclide with Water Reflector (g)
^{239}Pu	450 ^a
^{241}Pu	200
$^{242\text{m}}\text{Am}$	13
^{243}Cm	90
^{245}Cm	30
^{247}Cm	900
^{249}Cf	10
^{251}Cf	5

*Nuclides having an odd number of neutrons.

^aFor comparison.

^{238}Pu of high isotopic purity. Many uses for ^{238}Pu take the form of a heat source for radioisotopic thermoelectric generators that provide reliable electric power for many space missions as well as numerous terrestrial applications. The other plutonium isotopes are most notable as constituents of the plutonium generated in light water reactor (LWR) fuels, the ^{240}Pu being a significant neutron absorber and the ^{241}Pu providing additional reactivity. The ^{241}Pu also is the parent of ^{241}Am , the nuclide that has a large number of uses as an alpha, gamma, or neutron source. The ^{241}Am can be used to produce isotopically pure ^{238}Pu (through ^{242}Cm). The heavier isotope, ^{244}Cm , has already been produced in amounts of 10 kg or more in U.S. Department of Energy programs. Its even greater availability from future power reactor fuel reprocessing and its potential for replacing ^{238}Pu in some

TABLE III

Subcritical Mass Limits for Nonfissile* Actinide
Nuclides (Metal Form)

Nuclide	Mass Limit of Nuclide with Reflector of:	
	Water (kg)	Steel (kg)
^{237}Np	30	20
^{238}Pu	4	3
^{240}Pu	20	15
^{242}Pu	60	40
^{241}Am	24	16
^{243}Am	35	25
^{244}Cm	5	3

*Nuclides having an even number of neutrons.

applications where penetrating radiation can be tolerated make it an important nuclide.

Although critical masses have not been measured for any of these isotopes, sufficient nuclear data exist to permit reasonable estimations and, thence, to set mass limits that assure subcriticality, taking into account the uncertainties in the calculations. For the fissile isotope ^{241}Pu and others that contain an odd number of neutrons, the smallest critical masses occur for aqueous solutions, while for the nonfissile isotopes— ^{237}Pu , ^{238}Pu , ^{240}Pu , ^{241}Am , and ^{244}Cm —the smallest critical masses occur for unmoderated metals and oxides. Usually, the actinide elements being processed contain a mixture of isotopes, e.g., americium may contain sufficient fissile $^{242\text{m}}\text{Am}$ and curium may contain sufficient fissile ^{245}Cm that the smallest critical masses again occur for aqueous solutions. Sufficient nuclear data exist for these less abundant isotopes to set multiparameter mass limits for such isotopic mixtures.

Future efforts will involve extending the standard to new nuclides, e.g., ^{246}Cm and ^{252}Cf , and to other commonly encountered isotopic mixtures, e.g., the plutonium product from LWRs. In addition, as the nuclear data base improves, less stringent mass limits can be set that still assure subcriticality.

The subcritical mass limits listed here along with multiparameter limits on four different isotopic mixtures have been submitted by the work group for approval as an American National Standard.

Nuclear Criticality Safety of Reactor Fuel Elements—ANS-8.17, Hans Toffer (UNC Nuclear Industries), G. E. Whitesides (Union Carbide, Nucl Div)

This standard, ANS-8.17, currently under development provides nuclear criticality safety criteria and design guidance for the handling, storage, and transportation of light water

reactor (LWR) fuel elements and/or components of fuel elements such as individual rods outside reactors. Both spent and unirradiated fuel elements are being considered. The need for the standard arose when standards related to fuel cycle facilities design started incorporating sections pertaining to criticality control. Such sections contained some misinterpretation of generic basic nuclear criticality standards ANSI N16.1. It became apparent that a standard was needed that provided a detailed checklist related to nuclear criticality control for specific applications extracted from more generalized standards such as Nuclear Criticality Safety Outside Reactors (ANSI N16.1-1975), Validation of Computation Methods (ANSI N16.9-1975), and Criticality Accident Alarm System (ANSI/ANS-8.3-1979). The applications would be to the design of facilities or operations in fuel cycle facilities where the fissile material is in the form of rods or assemblies of rods. The facilities range from fuel element manufacturing plants to dry storage buildings or shear leach process head-end operations in a separations plant. Fuel inside a reactor core is excluded from consideration.

In the late 1970s, a work group was established to draft such a standard. The work group, under the chairmanship of G. E. Whitesides, consists of representatives from the private industry, from government contractor personnel, and scientists and engineers from the national laboratories. A draft of the standard's main body has been prepared and is currently undergoing final revision. Appendixes containing detailed specific application checklists to cover facilities such as an away-from-reactor storage pool (AFRs), a dry storage building, fuel storage pools, and shipping casks are being prepared. The standard will provide guidance and criteria for basic criticality control, the use of neutron absorbers and control over them, the need for and use of criticality alarms, allowance for verifiable presence of burnable poisons and/or fuel burnup.

One section of the standard deals with the criteria to establish subcriticality. The calculated multiplication factor is defined in terms of quantities considering calculational and experimental uncertainties and allowing for operating variables due to normal or credible abnormal conditions or events. Requirements for benchmarking information, use of codes not directly calculating k_{eff} , direct use of experimental data, and *in situ* measurements are discussed.

The main future effort in this standard development will be to develop the specific appendixes containing detailed checklists applicable to design standards for an AFR, a dry storage facility, and other fuel cycle components. With these appendixes, the standard will become a very valuable complement to the more generic standards pertaining to nuclear criticality control. It will provide a much needed information base to support design efforts and fuel handling procedure development in the nuclear fuel cycle.

CPVC as a Neutron Absorber in Solutions/ Fissile Material—ANS-8.18, E. L. Maceda (B&W)

Within nuclear fuel facilities, borosilicate-glass raschig rings are commonly used as neutron absorbers in solutions of fissile material.¹ A problem with raschig rings is their chemical incompatibility to some solutions found in fuel plants. Chlorinated polyvinyl chloride (CPVC) has been shown to withstand these chemical environments better than raschig rings and is now used as a secondary criticality control in some process areas in fuel plants. This paper describes work being done to develop a standard for the use of CPVC and its areas of application.

In February 1980, members of work group ANS-8.18 first met in Washington, D.C. to discuss the proposed standard. The committee consists of B. Ernst (American Nuclear

Insurers), J. Feuerbacher (Goodyear Atomic Corp.), N. Ketzlach (NRC), D. Sgarlata (B&W), and E. Maceda (B&W); also, R. Wilging (B. F. Goodrich Chemical Co.) is advising the committee on the chemical properties of CPVC. The scope of the ANS-8.18 committee's work is: "This standard presents criteria for the use of CPVC as a neutron absorber for secondary criticality control in solutions containing ^{235}U . The chemical and physical environment, properties of the pipes, rings and packed vessels, maintenance inspections and procedures, and criticality operating limits are specified." It was decided that the standard would address the present use of CPVC, which is that of a secondary criticality control in low-enriched uranium solutions.

CPVC is used in two configurations as a neutron absorber. First, it can be used the same way raschig rings are, that is, the vessel to be poisoned is filled with randomly oriented "rings" of CPVC.^{2,3} These "rings" are hollow cylinders with a height-to-diameter ratio of ~ 1 . A typical dimension for these rings is $1\frac{1}{4}$ in. Second, CPVC pipe can be packed into vessels as shown in Fig. 1. Concentric rings of pipe are placed inside the vessel until it is filled.⁴ This method has an additional advantage of allowing visual checks for sludge buildup in the vessels.

The main committee work is to develop an acceptable standard. Presently, a second draft of the standard is being prepared for the subcommittee's comments. This fall, a draft will be given to the ANS-8 committee members for comments. Another project that is being done by D. Sgarlata

Fig. 1. Vessel packed with CPVC pipes.

and E. Maceda is to write a paper containing the chemical compatibility tests performed on CPVC before it was placed in the process streams at the plant. This is in support of the standard writing effort.

In addition to extending the permissible chemical environments that can be poisoned from those allowed by raschig rings, CPVC can withstand temperatures to 180°C. This is 60°C beyond those allowed for raschig rings. Thus, an extension in environments that can be poisoned will be achieved by the new standard, which will make it useful in fuel processing plants. However, the restrictions presently placed within the standard does not allow substitution for raschig rings in every case. These restrictions are that of secondary criticality control, which means that some other parameter (such as concentration) must serve as primary criticality control, and that only low-enriched uranium solutions are treated within the framework of the standard. Hence, future work could involve extensions of the standard into areas not presently considered. This would require more data about CPVC both as to its use as a neutron absorber and its chemical resistance to other environments. Also, experiments to produce benchmark data would be useful in any extensions of this standard.

In this paper, I have presented work being done by ANS-8.18 to develop a consensus standard for the use of CPVC and described some work that, if done, would be useful in extending the range of applicability for the use of CPVC.

1. "American National Standard, Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material," ANSI/ANS-8.5-1979, American Nuclear Standards Institute (1979).
2. J. L. FEUERBACHER, "Relative Efficiency of Various Means of Adding Nuclear Absorbers to Direct-Steam-Heated Vaporization Bays for 10-Ton and 14-Ton UF₆ Cylinders," GAT-DM-1136, Goodyear Atomic Company (1970).
3. J. L. FEUERBACHER, "Computer Check of 2½-Ton UF₆ Cylinders at 5% Assay in Direct-Steam-Heated Vaporization Bays Containing Chlorinated Polyvinyl Chlorine Pipes Filled with Boric Acid," GAT-DM-1137, Goodyear Atomic Company (1970).
4. E. L. MACEDA, D. K. SGARLATA, and F. G. WELFARE, "Chlorinated Polyvinyl Chloride as a Secondary Criticality Control," *Trans. Am. Nucl. Soc.*, **32**, 336 (1979).

Criticality Accident Alarm Systems

The Need for Radiation Field Measurements from a Low-Enriched Critical Assembly, G. E. Whitesides (ORNL)

In the course of study to evaluate the radiation field that would result from criticality accidents, a series of calculations was made to simulate a variety of critical systems. As a part of the study, an attempt was made to establish the validity of the computational method by calculating systems for which experimental data existed. Experiments involving highly enriched uranium metal systems were most plentiful due to the extensive experiments involving the Godiva reactor at Los Alamos National Laboratory (LANL) and the Health Physics Research Reactor at Oak Ridge National Laboratory (ORNL). Measurements had also been made with a highly enriched uranium solution that simulated one of the early criticality accidents. The search, however, revealed that while critical experiments existed for low-enriched uranium solution systems, no radiation field measurements had been made for these systems.

Table I shows the results of dose calculations for the variety of systems studied. An examination of these data reveals that the neutron dose is principally affected by the degree of moderation. While the dose data are integral quantities over energy, some confidence in judging the neutron spectra for low-enriched systems is gained

by the data shown. Further confidence is derived from our ability to accurately compute the k_{eff} for these systems, since computing k_{eff} accurately probably means that the neutron leakage was calculated correctly.

The gamma dose is observed to vary principally as a function of enrichment. This, coupled with the lack of any other confirmations, indicated that no validation of these calculations were possible for low-enriched systems without obtaining experimental data.

A study of the spectral data from our calculations showed a considerable variation as a function of the parameters studied. Indeed, as one would expect, not only did the magnitude of the leakage change, but there was also considerable shift in the spectra as a function of energy. Since the response of interest in detecting and measuring the radiation from a criticality is a product of the radiation and response function integrated over energy, it is important that the radiation as a function of energy be known accurately.

Due to the absence of information on which we could base our validation efforts, it was our recommendation that an experiment be carried out with the expressed purpose of measuring the neutron and gamma-ray leakage spectra. This recommendation led to the design and operation of the SHEBA reactor at LANL, with the radiation spectral measurement being carried out by personnel from Lawrence Livermore National Laboratory (principally), ORNL, and LANL. With the data thus obtained, it will now be possible to validate calculations of radiation detection and measurements with a known degree of confidence.

TABLE I
Computed Surface Dose (rads) for 2×10^{15} Fissions

System	N Dose	γ Dose	N/ γ Dose Ratio	Leakage Fraction
93% Uranium Metal	16350	1150	14.2	0.56
93% Solution H/ ^{235}U = 40	5840	2760	2.1	0.45
93% Solution H/ ^{235}U = 1000	498	835	0.6	0.25
10% Solution H/ ^{235}U = 400	1340	905	1.48	0.36
5% Solution H/ ^{235}U = 496	450	289	1.56	0.22
5% Solution H/ ^{235}U = 616	385	286	1.35	0.21
3% Solution H/ ^{235}U = 494	128	45	2.84	.22
1.5% Solution H/ ^{235}U = 494	72	17	4.24	.13

The Function and Characteristics of Criticality Accident Alarm Systems, David R. Smith (LANL)

The purpose of a criticality accident alarm system is, or should be, to reduce the risk associated with fissile material operations. Experience has shown that such a system can effect a significant reduction in radiation exposures, even to the extent of saving lives, should a criticality accident occur. One ought to be aware, however, that alarm systems present problems as well as virtues. Even though the need for a system may be marginal, once it is installed there is a common attitude that operations in that area should cease if the system becomes inoperative. An installed system requires frequent tests and maintenance and introduces the potential for false alarms and the associated risk of injury to personnel.

The criticality accident that is expected to be signaled by the alarm system may have widely varying characteristics, depending on the form and type of fissile material and on the initiating mechanism. Solution accidents, clearly the most probable for process areas, have a rise time limited by the neutron lifetime, which is on the order of milliseconds. While accidents involving metal or dry compounds are easier to avoid, the shorter neutron lifetime and faster rise time of the excursion lead to much

higher peak power densities and higher radiation dose rates. Solutions always have inherent shutdown mechanisms as by boiling or radiolytic gas formation. Most possible accidents involving metal would be terminated by expansion or melting of the metal. An accident involving a large accumulation of fissile material as a dry powder could have little or no terminating mechanism until the power density became high enough to vaporize the powder, resulting in a relatively large excursion.

The size of a criticality accident is also dependent on the neutron population at the time the system achieves criticality. If few neutrons are present, a high degree of supercriticality may be reached before a reaction is initiated, resulting in a fast rise time and high peak power.

The eight process criticality accidents that have occurred can be divided into two broad categories. Four of these events were critical only briefly, being quickly terminated by the changing physical configuration of the system, while the other four reactions endured for periods of 20 min to 37 h (Ref. 1).

For an accident in which the reaction is terminated promptly, an alarm system will have only a modest impact on personnel exposures, though even this can be important. Alarms are truly valuable when the accident endures for substantial times, and in the two such cases in unshielded operations, the alarm system is credited with saving lives. These events were at the Oak Ridge Y-12 Plant in June 1958 and the Recuplex Plant at Hanford in April 1962.

In principle, there is no lower limit to the magnitude of a criticality accident. However, any accident that is self-limiting must generate sufficient energy to cause a reduction in the multiplication factor of the reacting system. In addition, very small accidents require that the excess reactivity involved be so small that the probability of such events is limited. Designing a reasonable criticality accident alarm system requires the assumption of a minimum accident that must be detected. A consensus has been established² that such an accident will result in $\sim 10^{15}$ fissions in the first minute it endures and provide a dose of ~ 20 rads at 2 m from the reacting medium in this time if no significant shielding is present. These values are based on accident experience supported by considerations of accident mechanisms and effects of energy releases. The minimum accident assumption determines the alarm set point and the detector spacing.

Consideration of the distance from process areas for which evacuation of personnel is desirable requires estimate of the size of sustained reactions. Experiments performed at the critical experiments center in France³ are very useful. Data from these experiments suggest that criticality accidents in 300-mm-diam cylinders containing highly enriched uranium in solution will probably not result in more than 10^{18} fissions in an hour. This would correspond to a dose equivalent of <10 rem at 100 m. While accidents in larger vessels might have higher yields, the dose equivalent would increase slowly because of increased self-shielding.

The alarm standard² addresses the need for alarm systems and also characteristics of alarm signals, dependability, testing procedures, and emergency planning. Appendixes are concerned with supporting the selection of a minimum accident of concern and with calculations of the radius of coverage of a detector.

The International Standards Organization has under development a document directed toward criticality alarm systems, while the International Electrotechnical Commission is preparing guidance for the characteristics of the equipment used in these systems.

Regulations of the U.S. Nuclear Regulatory Commission (NRC) direct that alarm systems be installed purely on the basis of the quantity of material in a facility, while the American National Standard recommends that all areas containing more than threshold quantities of material be

surveyed to determine the need for an alarm system and that systems be installed when it is judged that they will provide a reduction in total risk. An NRC regulatory guide generally endorses the American National Standard, but requires application for a waiver when it is believed that installation of an alarm system will not result in risk reduction.

1. W. R. STRATTON, "A Review of Criticality Accidents," LA-3611, Los Alamos National Lab. (Jan. 1967).

2. "Criticality Accident Alarm System," ANSI/ANS-8.3-1979, American Nuclear Society.

3. P. LECORCHE and R. L. SEALE, "A Review of the Experiments Performed to Determine the Radiological Consequences of a Criticality Accident," Y-CDC-12, Oak Ridge Y-12 Plant (Nov. 3, 1973).

Facility Description of a Solution Critical Assembly: SHEBA, R. E. Malenfant, H. M. Forehand, Jr. (LANL)

SHEBA has been described previously¹ as a 4.95% clean geometry critical assembly employing fuel of enriched uranium as the fluoride. The primary applications of the machine have been to evaluate accidental criticality alarm detectors for enrichment plants, to provide radiation spectra and intensity measurements, to benchmark calculations on a low-enrichment solution system, and to provide radiation fields to calibrate personnel dosimetry.

Goodyear Atomic Corporation had requested the assistance of the Oak Ridge National Laboratory (ORNL)

TABLE I
Total Neutron Leakage Distribution for SHEBA

Group	Lower Energy Limit	Fraction of Leakage in Group	Leakage per Fission
	14.0 MeV		
1	3.0 MeV	0.105	0.0682
2	1.4 MeV	0.201	0.1305
3	0.9 MeV	0.090	0.0584
4	0.4 MeV	0.124	0.0805
5	0.1 MeV	0.116	0.0753
6	17.0 keV	0.075	0.0487
7	3.0 keV	0.043	0.0279
8	0.55 keV	0.039	0.0253
9	0.10 keV	0.034	0.0221
10	0.03 keV	0.022	0.0143
11	0.01 keV	0.020	0.0130
12	3.0 eV	0.019	0.0123
13	1.0 eV	0.017	0.0110
14	0.4 eV	0.014	0.0091
15	0.1 eV	0.030	0.0195
16	Thermal	0.051	0.0331
		1.000	0.6491

Fig. 1. Geometry of SHEBA core vessel. All material is Type 304 stainless steel except as noted.

Fig. 2. Dose rates at 3 m from center of SHEBA core vessel.

in the evaluation of the spacing for nuclear criticality alarm detectors for enrichment plants. Due to uncertainties in the calculation and the high cost of safety through conservatism, Goodyear and ORNL requested that Los Alamos National Laboratory run a solution critical to evaluate neutron and gamma leakage from a solution of relatively low enrichment uranyl fluoride. An extension of the experiments to evaluate personnel dosimetry was obvious. It is interesting to note that all accidents since 1958 have involved solutions, although dosimetry is evaluated with metal assemblies such as the Health Physics Pulsed Reactor.

Normalized neutron leakage distribution presented in Table I was obtained with a TWODA computer calculation using the Hansen-Roach 16-group cross sections and the system geometry as shown in Fig. 1. The fuel solution, $U(4.95)O_2F_2$, has a density of 2.16 g/ml, with a $U(4.95)$ content of 1.04 g/ml. Total neutron leakage in each group per fission event was obtained from the calculated value of 0.6491 leakage neutrons per fission.

The measured dose rate at 3 m from the active center

of the SHEBA core is presented in Fig. 2 as a function of the current output from our standard polyethylene-moderated BF_3 ionization chamber delivered to the linear amplifier readout in the control room.

SHEBA is perhaps unique today in that there is no control rod to perturb the flux. Control is effected solely by varying solution level with a simple combination of pressure and vacuum through a single control handle. There are no pumps in the solution system, and only binary valves are employed. This simplicity has provided a clean system with a very simple control scheme.

We are currently operating SHEBA at various ramps and steps to measure shutdown mechanisms for eventual burst operation. Evaluation of dynamic characteristics is being extended by time-dependent measurements of neutron level with the system at and near critical when pulsed with a neutron generator.

Initial indications are that radiolytic gas formation and thermal expansion provide a fast and reliable shutdown.

Burst operation will allow extension of the measurements with SHEBA to a wider range of accident simulations.

SHEBA continues to be a valuable tool for dosimeter evaluation, accident simulation, and student instruction.

I. R. E. MALENFANT, H. M. FOREHAND, J. J. KOELING, "SHEBA: A Solution Critical Assembly," *Trans. Am. Nucl. Soc.*, **35**, 279 (1980).

Neutron and Gamma-Ray Spectrum Measurements at the Los Alamos Critical Assembly SHEBA, John H. Thorngate, Duane W. Rueppel, Richard V. Griffith (LLNL)

In the summer of 1980, SHEBA, a critical assembly intended as a facility for evaluation of criticality monitors,

Fig. 1. SHEBA neutron spectra at 5 m.

was completed at the Los Alamos National Laboratory. The energy-dependent response of criticality monitors affects how they respond to radiation fields. Since these fields will be affected significantly by local shielding, it was important to have data on the neutron and gamma-ray spectra from the bare assembly, as well as transmitted through shields (13.3-cm-thick steel and 40-cm-thick concrete) used to modify the neutron spectrum. Because of recent work on radiation spectrometry techniques at the Lawrence Livermore National Laboratory, we were asked to make such measurements.

Our measurements were made at 5, 14, and 69 m from the assembly with and without the steel or concrete shields between the assembly and the detectors. Gamma-ray spectra were measured up to 10 MeV with a 2- X 2-in. NaI(Tl) detector. The small size of the detector limited its high-energy response, but a larger detector would have produced an unacceptably high count rate. Above 0.3 MeV, neutron energy spectra were measured with a 2- X 2-in. NE213 scintillator used with a pulse-shape discriminator

to suppress gamma-ray response. Measurements were made at two detector gains, covering the energy ranges from 0.3 to 2 MeV and from 1 to 12 MeV. A multisphere system comprised of a 0.5- X 0.5-in. ^6LiI detector with a cadmium shield and 3-, 5-, 8-, 10-, and 12-in.-diam spheres of polyethylene was used to measure the entire neutron energy spectra, but with poorer resolution than the NE213 detector.

To take full advantage of the NE213 resolution and the multisphere energy range, we began reducing the neutron data by unfolding the NE213 response using a program that differentiates the proton recoil response and corrects for detector characteristics such as nonlinear light output and multiple scattering. Figure 1 shows the neutron spectra obtained with the NE213 detector at 5 m from SHEBA with and without the shields. The NE213 results were also an input parameter for the evaluation of the multisphere responses used to calculate the spectra below 0.3 MeV. Figure 2 presents a comparison of the full range composite neutron spectra thus obtained at 5 m.

Fig. 2. SHEBA neutron spectra at 5 m.

Fig. 3. SHEBA gamma spectra.

The gamma-ray spectra were calculated with a program that fits calculated detector responses to the data. A comparison of the results obtained 5 m from the unmoderated assembly and at 69 m with the steel shield at 12 m is given in Fig. 3. The quality of the data is limited by the number of bins in the reduced data. Although there is limited information, the 69-m spectrum does show relatively more response in the 5- to 7-MeV region due to neutron interactions in the air.

Neutron and Gamma-Ray Dose Measurements at the Los Alamos Critical Assembly SHEBA, Dale E. Hankins (LLNL)

The SHEBA critical assembly, located at the Los Alamos Critical Assemblies Laboratory, was constructed during 1980 to provide dosimetry and spectral information from a low-enrichment uranium solution system. The assembly consists of a stainless steel cylindrical tank 56 cm in

diameter, with a 6.35-mm-thick wall. The tank is filled with a fuel solution of 4.8% enriched uranium as a fluoride to a depth of ~61 cm where criticality is attained. The SHEBA assembly is housed in a thin metal shed.

Measurements were made of the neutron and gamma-ray dose rates at distances ranging from 4.5 to 69 m from the assembly. The effect of shielding materials on the dose rates and spectra were determined using a 13.3-cm-thick steel shield positioned at 4 or 12 m from the assembly and concrete shields 20.3 and 40.6 cm thick placed at 12 and 4 m, respectively.

Neutron doses were determined using the 9-in. sphere PNR-4 rem meter and from the readings of thermoluminescent detectors (TLDs) (^6Li and ^7Li) placed inside the 9-in. sphere. Thermal neutron doses were measured using the BF_3 tube of the PNR-4 instrument and with pairs of ^6Li and ^7Li TLDs. We used albedo neutron dosimeters of the "Hankins" type and applied the 9- to 3-in. sphere ratio technique to evaluate the albedo readings. Gamma-ray dose measurements were made using TLDs

TABLE I
Results Obtained at a Distance of 5 m from the Assembly Being Operated at a Power of 0.209 W

	Neutron Dose Rate (mrem/h)			Gamma Dose Rate (mR/h)			Sodium Activation per Rad (pCi/mg)	Thermal Neutrons (Percent of Total Neutron Dose Rate)	Ratio of Neutron of 9-in. Sphere w/TLDs to (in Air)
	9-in. Sphere Rem Meter Instrument	9-in. Sphere TLDs in Sphere	NE213	Multisphere	TLDs in Air	TLDs on Phantom	Personnel TLDs Badges		
Bare	57.2	66.4	65.2	68.9	5.42	6.32	6.54	0.77	12.3
Steel at 4 m (13.3 cm thick)	24.2	24.8	20.9	23.5	0.42	0.58	0.60	0.36	59.0
Concrete at 4 m (40.6 cm thick)	8.73	8.55	5.9	7.3	0.69	0.81	0.84	2.21	12.4

located in air and on phantoms. Two types of phantoms were used, one consisting of the head and torso section from a Remab phantom (containing bones) and the other being a Lawrence Livermore National Laboratory water-filled torso phantom. A saline solution of Na_2CO_3 (sodium carbonate) was placed in both phantoms, and the activity of the ^{24}Na was determined. Sulfur pellets 38 mm in diameter were exposed on the phantoms, and the ^{32}P activity was used to determine the fast neutron ($>2.9\text{-MeV}$) component of the neutron fluence. The ratio of the gamma dose in air to the gamma dose on the phantom and the gamma-to-neutron ratio was calculated.

Table I shows a portion of the results obtained at a distance of 5 m from the assembly operating at a power of 0.209 W. The neutron dose rates include those obtained by Thorngate et al.¹ using both the multisphere neutron detectors and an NE213 scintillator. The neutron dose rates agree reasonably well, except for the results with the concrete shield where the 9-in. sphere results are higher and the NE213 lower than the multisphere. This is probably caused by an overresponse of the 9-in. sphere and underresponse of the NE213 to the low-energy neutrons leaking through and around the concrete. The neutron dose rates are lower behind the steel shield by ~60% and behind the concrete by ~90%.

The gamma dose rates given in Table I indicate higher readings on the phantoms, which is caused by neutron capture in the phantom. The gamma dose rates are reduced behind the iron and concrete shields by factor of ~13 and 8, respectively. The neutron-to-gamma ratio varies from ~12 for the bare and concrete shield to ~60 for the steel shield. These values are higher than those usually quoted for similar assemblies because the neutron dose rate in this study are in units of mrem rather than mrad. A quality factor of between 8 and 9 applies for this assembly.¹

One application of the SHEBA assembly is for dosimetry studies following an accident. The blood sodium activation of a person exposed to this assembly (bare and shielded) would be between 10.7 and 19.0 pCi/mg of ^{23}Na , which is higher than studies with other assemblies.² The leakage neutron spectrum has few neutrons above 2.9 MeV, which makes the sulfur activation per rad for this assembly low compared to other assemblies. When combined with the high sodium activation, these results give new data points on the curve used to evaluate accident doses using only blood and hair activation² that extend the curve and confirm its accuracy.

Thermal neutrons contribute little to the dose received around the assembly (see Table I), the largest contribution being only 2.2% of the total neutron dose rate.

The neutron dose rates as a function of distance from the bare assembly out to 69 m follow the inverse square law. Behind the steel shield, they decreased more rapidly than the inverse square out to ~9 m and then less rapidly out to 35 m. The decrease behind the concrete shield was considerably less rapid than the inverse square (from 5 to 9 m).

The 9- to 3-in. sphere ratio became smaller as the distance from the bare and steel-shielded assembly was increased, but remained fairly constant behind the concrete shield. The albedo results obtained were within 25% of the expected dose received by the phantoms.

1. JOHN H. THORNGATE, R. V. GRIFFITH, and D. W. REUPPEL, "Neutron and Gamma-Ray Spectrum Measurements at the Los Alamos Critical Assembly SHEBA," *Trans. Am. Nucl. Soc.*, **39**, 557 (1981).
2. DALE E. HANKINS, "Dosimetry of Criticality Accidents Using Activations of the Blood and Hair," *Health Phys.*, **38**, 529 (1980).

A Criticality Alarm System for Plutonium Process Operations, William H. Tyree, Henry Urano (Rockwell International)

Rocky Flats is operated by Rockwell International for the U.S. Department of Energy (DOE). The plant has 11 buildings involved directly with fissile material processing and storage. These buildings have detection systems utilizing neutron detection, with two detector coincidence criteria required before a general building alarm is sounded. Each of these buildings includes multiple detectors with coincidence control, local and remote display, and multiple annunciation. This concept developed as an alternative to the former system, which used a gamma-sensitive scintillation system operating in a single-detector criteria alarm configuration.

The basic detection process uses the neutron-alpha reaction in ${}^6\text{Li}$. The 4.78-MeV reaction energy produces an easily detected alpha particle with low sensitivity to gamma radiation and good electrical noise immunity. A prototype neutron detection system using a 20-mm² gold window surface barrier detector coupled to a ${}^6\text{LiF}$ foil was tested at the Los Alamos National Laboratory (LANL) Critical Test Facility in the fall of 1968. The detector unit produced an alarm when placed 580 m from a metal assembly producing 5×10^{16} fissions in 33 μs . This detector is presently installed in the 11 neutron criticality alarm systems at Rocky Flats.

A new detector unit has been designed to add an internal check of radiation detection sensitivity by including a ${}^{230}\text{Th}$ source in the detector ${}^6\text{LiF}$ foil assembly. This provides continuous alpha-particle emission to test the ability of the foil detector to sense the neutron alpha-particle interaction in the ${}^6\text{LiF}$ foil. An internal clock circuit forces the SCR to trigger if the detector amplifier fails for longer than ~ 1 min. The unit produces a radiation alarm with any detected pulse rate from the ${}^6\text{LiF}$ foil > 4 pulse/s. Mechanical and electrical compatibility of the new detector design with the existing detection system has been a basic requirement not only to maintain the existing radiation sensitivity but also to minimize installation costs.

Both detector designs were tested at the LANL facility with a slow rise time liquid assembly. This assembly was operated at a continuous fission output of 1.65×10^{12} for each of several tests on November 12, 1980. The detector units using a 100-mm² gold window surface barrier detector produced an alarm up to 200 m away from the assembly.

In the existing production building systems, detectors are located at distances ranging from ~ 8 m to not greater than 30 m from glove boxes involving fissile material operations and storage areas.

The criticality detection sensitivity tests conducted in 1968 and 1980 at the LANL Critical Facility have provided important operational performance benchmarks for the Rocky Flats criticality alarm systems. These tests confirm that the response of these systems meet or exceed the performance requirements of ANSI 8.3-1979 and DOE Manual Chapter 0530.

A Study of Some Current Technologies for Criticality Alarm Systems, Norman L. Pruvost (LLNL), John C. Tsitouras (EG&G Energy Meas Group)

We have recently completed a study of criticality alarm systems. The purpose of the study was to propose a new design for alarm systems about to be replaced

at the Lawrence Livermore National Laboratory (LLNL). Our approach was to visit some of the other U.S. Department of Energy (DOE) facilities and some commercial firms to inform ourselves of recent developments in technology. Our study did not progress to the stage of completing detailed engineering designs. Nevertheless, we found it useful to begin our study by selecting the following design criteria.

1. High reliability. A reliable system will, with high reliability, sound the Klaxon warning when there is a criticality. In addition, the system will not, with high reliability, sound the Klaxon when there is no criticality. We chose the criterion that, on a component-by-component basis, a new system component would provide performance as high or higher than the analogous component in the alarm systems presently installed at LLNL.

2. Low maintenance cost. A system whose lifetime is 10 to 15 years can incur maintenance cost as high as three to five times the initial system cost. Consequently, a reduction in maintenance cost can significantly reduce total system cost. We took as a design criterion a reduction in maintenance cost compared to cost for systems presently installed at LLNL. However, we consider lower maintenance cost to be of second priority to system performance.

3. Compliance with ANSI 8.3. This compliance, along with DOE compliance, was an in-house requirement.

4. Standardization of alarm system hardware. Our design criterion was a single standardization system with hardware capable of servicing all laboratory facilities. The most important aspect of standardization is assuring that the criticality warning Klaxon (or sound) will be common throughout all laboratory facilities.

5. Low voltage electronics. We adopted the design criterion that a new system should utilize electronics that operate in the low (12- to 24-V) range insofar as practical.

6. Uninterruptible power supply (UPS). We adopted the design criterion that the criticality alarm system should, during power failure, operate from a UPS dedicated to the alarm system.

7. Commercially manufactured units. We adopted the criterion that we would incorporate off-the-shelf hardware when available.

8. Self-checking features. We chose to incorporate as many self-checking features into the design as appeared practical. Our design criterion was that a self-checking feature must result in higher overall system reliability.

In addition to the above criteria, we note that other factors may be important in system design. As examples, we mention purchase and startup cost, hardening against shock, fire, and electronic interference, reliability of system hardware suppliers, and compatibility for computer interfacing. Computer interfacing can be an important consideration if the alarm system is connected to a larger laboratory-wide emergency response network. The study allowed us to identify three options that appear to satisfy most of our design criteria. A brief description of each option is given below.

1. The first option consists of Tyree detectors (see option two below) integrated directly into a commercially available data acquisition system made by Eberline Corp. of Santa Fe, New Mexico. This option operates in a digital mode by detecting individual neutrons, transmitting pulses over a twisted shielded pair to the Eberline data acquisition module (DAM), which then interfaces with the central control terminal (CT-1). The CT-1 contains two microprocessors. The DAM and CT-1 units carry out the data

processing and alarm decision-making functions in a digital mode. Reliability is increased by the twin CT-1 microprocessors alternating control of the system. The entire detection system operates at <20 V. In addition, the CT-1 can interface to a smart terminal with disk storage and color monitor display. This system is configured to assure that no interference due to the smart terminal can take place with the process of detection, alarm decision-making, and sounding of the Klaxons.

2. The second option is the system developed and currently in use at the DOE Rocky Flats Plant. The system has a record of about ten years of successful operating experience. The detector (developed by William Tyree) is a solid-state neutron sensing unit utilizing the ${}^6\text{Li}(n,\alpha)$ reaction. The reaction cross section is sufficiently large to provide some detector sensitivity up to ~ 1 MeV. A small alpha source is incorporated into each detector as part of a continuous self-checking feature. The entire system is low voltage and operates in a mixed digital and analog mode.

3. The last option examined was the current system offered by Nuclear Measurements Corporation of Indianapolis, Indiana. Three of these systems are installed at LLNL and have performed well over the last 5 to 10 years. The system technology utilizes high-voltage (2-kV) gamma-sensing detectors and operates entirely in an analog mode. The detector utilizes a photomultiplier tube that requires the high voltage.

Our study leads us to a preference for option one. This preference stems from the advantages we believe result from a low voltage and digital mode of operation. In addition, this option appears to satisfy more of our design criteria than the other options. Our study indicates

that most reliability and maintenance problems result from difficulties in providing high voltages (1 to 3 kV) in the detectors over periods of months and years. The low-voltage solid-state components utilized the Tyree detector and throughout options one and two are relatively more reliable and trouble free. In addition, a low-voltage (low-power) system reduces the size and complexity of the system UPS.

We believe all three options will satisfy the compliance criterion. Standardization leads us to prefer neutron over gamma detectors. This comes about because some facilities could accidentally produce gamma fluxes capable of setting off the criticality alarm system without a criticality. Personnel in these facilities prefer neutron to gamma detection to guard against such a false criticality alarm. The Tyree detector, in addition to being a neutron detector, can be packaged into a unit about the size of a bar of soap. The small size provides for ease of installation and some hardening to mechanical shock. The self-checking features of the Tyree detector and the Eberline CT-1 reinforce our preference for option one. Finally, the smart terminal capability of option one does allow for information about system performance and variations in detector background levels to be saved and displayed. Options two and three do not have this capability. We believe that the additional information provided by a smart terminal could lead to an enhancement in criticality safety. This enhancement could come about by increased awareness of how the system is operating or by alerting facility personnel to a potential criticality by observing unexplained changes in detector background levels.

We conclude that criticality alarm systems can now be designed and built that utilize low-voltage solid-state detectors and circuitry. These systems can operate in a digital mode and provide more information than the sounding of Klaxons for a criticality.

Key Problems in Thermal Reactor Physics

Benchmark Data Testing for Thermal Reactor Applications, P. F. Rose, S. Pearlstein (BNL), J. Hardy (BAPL)

The purpose of the thermal reactor data testing effort is to determine the adequacy of basic cross-section data in thermal reactor applications, particularly for beginning-of-life criticality calculations.

This data testing effort has:

1. pointed up inadequacies in Evaluated Nuclear Data (ENDF/B) sets, and thus has stimulated new differential data measurements and improved evaluations

2. provided standard calculated results against which approximate or newly developed accurate methods can be tested

3. identified inadequacies in integral experiments

4. identified problems in specific calculational methods.

In preparation of ENDF/B-V, a great deal of testing has been done with ENDF/B-IV as well as preliminary versions of ENDF/B-V. This data testing, covering the major fissile and fertile materials, highlighted a number of discrepancies and contributed to a number of data improvements by the nuclear evaluators that have significantly

TABLE I
Benchmarks Characteristics

System Type	Benchmark Coverage
U235-H ₂ O	Good. Small assemblies should be added to benchmark set.
U235-U238-H ₂ O	Good.
U235-U238-D ₂ O	Poor. MIT-4, 5, 6 are old and exponentials. LTRIIA is complex and has some inconsistencies with ρ^{28} . ZEEP is being added.
U233-H ₂ O	Good. Full range of ORNL and small assemblies should be added to benchmark set.
U233-Th-H ₂ O	Good. BNL's are exponentials. Would be better to have simple critical lattices.
Pu-H ₂ O	Poor. Need some well determined experiments.
UO ₂ -PuO ₂ -H ₂ O	Fair. Could use additional experiments to resolve inconsistencies and provide lattice parameters.

upgraded the performance of ENDF/B-V for thermal reactor analysis.

Data testing has centered around a series of benchmark assemblies for the principal fissile and fertile materials. These include homogeneous aqueous assemblies and simple rod lattices. The Cross Section Evaluation Working Group (CSEWG), a cooperative effort of more than 20 laboratories, has specified a number of thermal benchmarks that are either H_2O - or D_2O -moderated, and most are critical assemblies. Detailed benchmark specifications are provided in ENDF-202 (Ref. 1).

In the initial CSEWG testing of ENDF/B-IV, effort was concentrated on the homogeneous assemblies and the simple lattices of slightly enriched uranium rods in a moderator of H_2O or D_2O . Integral transport theory with special resonance treatments and Monte Carlo methods were used. Results of these efforts are collected in an ENDF/B-IV Data Testing Report.²

The Oak Ridge National Laboratory aqueous ^{235}U - H_2O assemblies covered a wide range of neutron leakage through variations of the $H/^{235}U$ fraction. The calculated k_{eff} compared to experiment for these assemblies significantly increased with leakage. This was attributed to (a) a leakage underprediction caused by the soft fission spectrum in ENDF/B-IV, and (b) the high value of η for ^{235}U in ENDF/B-IV which is emphasized in the harder flux spectrum of these high-leakage assemblies. Evidence such as this strongly suggested to the evaluator a need for a harder fission spectrum for ^{235}U that led to the adoption for ENDF/B-V of a prompt Watt spectrum with $\bar{E} = 2.03$ MeV. (The ENDF/B-IV spectrum was Maxwellian with $\bar{E} = 1.985$ MeV.) The analysis also suggested the need for an increase of thermal $\bar{\nu}$ for ^{235}U . A 0.74% increase in $\bar{\nu}$ for ^{235}U was adopted in ENDF/B-V compared to ENDF/B-IV.

The TRX assemblies have been extensively analyzed both from data and methods points of view. The chief problem has been to properly calculate shielded ^{238}U resonance capture. The strong correlation between calculated k_{eff} and the epithermal-to-thermal ratio of ^{238}U capture (ρ^{28}) for these assemblies has been the chief source of discrepancy. Since the shielded resonance integral is only ~ 15 b, unshielded components of the capture, which are small compared to the dilute capture integral, become very important in these lattices. The calculated values of ρ^{28} decreased $\sim 2\%$ from ENDF/B-IV to -V, mostly due to the impact of new measurements³ reducing the capture width of the first three s-wave levels of ^{238}U and increasing the neutron widths at 20.9 and 36.9 eV.

The calculation is now $\sim 2\%$ above experiment on the average and is consistent with the overall uncertainty assignments. The effective multiplication factor increased from 0.987 (ENDF/B-IV average) to $k_{eff} = 0.997$ with ENDF/B-V, due primarily to the 0.74% increase of the ^{235}U thermal $\bar{\nu}$ normalization.

The aqueous plutonium (nitrate) $-H_2O$ assemblies are mostly unreflected homogeneous spheres modeled as a sphere in a void. These assemblies were analyzed with ENDF/B-IV by a number of laboratories. Average k_{eff} results are high ($k_{eff} \approx 1.011$), and show a considerable amount of scatter. The overall picture using ENDF/B-V data is unchanged from Version IV. The large scatter of k_{eff} (2.7%) shows no convincing trends and is probably attributable to uncertainties in plutonium content. In any case, sensitivities are sufficiently small that it is difficult to explain the high k_{eff} on the basis of obvious data deficiencies. There remains a need for several highly accurate benchmark critical experiments and careful studies to identify the source of high k_{eff} values.

An ENDF/B-V Data Testing Report has been prepared,⁴ and it is instructive to complete our comments about the thermal benchmark coverage with the information from that report presented in Table I, which summarizes the current status of the experiments.

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3. G. DE SAUSSURE et al., "Evaluation of the ^{238}U Neutron Cross Sections for Incident Neutron Energies up to 4 keV," ORNL/TM-6152, ENDF-257, Oak Ridge National Lab. (1978).
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Measurement and Calculation of Reactor Parameters—II

Determination of the k_{eff} of a Subcritical Assembly Using the Asymmetric Source Method, H. Ocampo, S. H. Levine (Pennsylvania State Univ)

It has been proposed^{1,2} that we measure the k_{∞} of a spent fuel assembly by using a subcritical assembly wherein a neutron source multiplication measurement, made with the fuel assembly of an unknown k_{∞} , is compared with the same measurement performed with a fuel assembly of known k_{∞} . The device in which this measurement is performed is called a k_{∞} -meter, and it requires that the k_{eff} of the initial subcritical assembly be measured. The purpose of this paper is to report the results of subcritical experiments performed with the Pennsylvania State University Breazeale TRIGA reactor to determine the k_{eff} of a subcritical assembly using the asymmetric source method³ (ASM). Similar experiments would be performed with the k_{∞} -meter to establish the k_{eff} of the initial subcritical assembly.

For these experiments, the TRIGA reactor is initially arranged as shown in Fig. 1 minus four fuel elements in positions H-4, H-14, J-4, and J-14. Two sets of experiments have been performed, all using core configurations with half-core symmetry. The first set involved measuring the reactivity of critical and slightly supercritical cores to provide data for calibrating the numerical calculations and to measure the reactivity change for a few corresponding changes in the core. These latter measurements were used to determine the k_{eff} of a subcritical core to check the ASM.

The excess reactivity of three supercritical cores was determined using the period method and control rod calibrations.⁴ The initial core of Fig. 1 had an excess reactivity of 22.9 cents. Adding four fuel elements to positions H-4, H-14, J-4, and J-14 increased the core excess reactivity to 112.4 cents. Replacing two 12-wt% uranium fuel elements with two 8.5-wt% uranium fuel elements (positions I-8 and I-9) reduced the core reactivity back to a just critical condition, i.e., this change was worth 112.4 cents. Hence removing the four fuel elements in positions H-4, H-14, J-4, and J-14 from

Fig. 1. TRIGA core configuration.

Fig. 2. Count rate ratio versus effective multiplication.

the latter just critical core reduced the core reactivity by 89.5 cents and produced a subcritical assembly having $k_{eff} = 0.993774$. This core was the first in a series of subcritical cores with which the ASM was used to measure their k_{eff} .

In the subcritical measurements, two fission detectors were placed on the core line of symmetry, each ~25 cm from the edge of the core. A ^{252}Cf neutron source was placed in the central thimble (CT) 3.7703 cm off the core centerline. A series of subcritical configurations was attained by replacing in steps two 12-wt% uranium fuel elements with 8.5-wt% uranium fuel in a manner that preserved half-core symmetry. After each change, the count rates of the two fission detectors were taken. The count rate ratio, R , was determined for each configuration for which $R < 1$. Figure 2 shows the count rate ratio as a function of k_{eff} for both experimental and calculated results. The LEOPARD-EXTERMINATOR-2 codes were used to calculate the k_{eff} and R for each configuration.²

The major part of the discrepancy between the experimental points and the calculated data can be explained by an offset of 0.15 cm in the flux symmetry. The two detectors were positioned to give the same count rate for the initial subcritical assembly. Correcting for a 0.15-cm displacement of one of the fission detectors brings the experimental data in line with the calculated data, as shown in Fig. 2. The remaining difference between the experimental and calculated data can be explained by errors in the LEOPARD cross sections and in the experimental data. Errors in the cross sections, used in a numerical calculation, do not change the slope of the line. Such errors only change the relative position of the points on the line; hence, the calculations can be used to provide the correct slope and correct for any offset in the detector position. The experimental count rate ratio, on the other hand, can then be used to determine the k_{eff} . Based on these measurements and calculations, the asymmetric source method can be used to determine the k_{eff} of the subcritical core within $\pm 0.003 \delta k/k$ in reactivity.

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Data and Analysis for Nuclear Criticality Safety

Development of the Computer Code System JACS for Criticality Safety, J. Katakura, Y. Naito, Y. Komuro (JAERI-Japan)

Calculational techniques for criticality safety assessment are required to be able to treat

1. various moderation conditions
2. various geometrical shapes of a system
3. considerable neutron leakage from a system
4. various phases of fuel materials.

To satisfy the above requirements, a computer code system named JACS has been developed in Japan, in which neutron transport calculations are performed with the Monte Carlo code KENO-IV (Ref. 1) and a modified KENO-IV code MULTI-KENO, S_n codes ANISN-JR (Ref. 2) and DOT3.5 (Ref. 3), or the diffusion code FEDM (Ref. 4) with the finite element method. The Monte Carlo code is a powerful tool for criticality safety evaluation. For the Monte Carlo calculation, the KENO code was selected, for which the Hansen-Roach library⁵ is often used, and a simple P1 approximation was applied to scattering. The number of energy groups in the Hansen-Roach library, e.g., 16, is too small to satisfy requirement 1 above, and is especially inadequate for calculation of thermal neutron behavior. A simple P1 approximation used in KENO-IV is also inadequate to satisfy requirement 3 above. Taking this into consideration, the cross-section library and treatment of scattering of KENO-IV were improved.

A new multigroup cross-section library, MGCL (Ref. 6), with 137 energy groups, has been generated from the ENDF/B-IV by the MGCL-ACE code system,⁶ in which temperature-dependent ultra-fine (~70 000 energy groups) cross sections are produced by the computer code RESEND-D, an improved version of the RESEND code,⁷ with two kinds of weighting neutron energy spectra available for collapsing the ultra-fine cross sections. One is a prepared standard spectrum that consists of three parts: fission spectrum, 1/E, and Maxwellian. The other is obtained by solving the neutron slowing down equation. The upscattering is taken into consideration for energy below 1.855 eV (45 groups of 137 groups).

Treatment of the angular-dependent scattering cross sections of KENO-IV was improved as follows: the angular dependency of the scattering cross section of hydrogen is treated exactly, and those of the other nuclides are obtained by a random number that is selected to show the expected value of the angle, and becomes the P1 coefficients of a Legendre polynomial.

A MULTI-KENO code, including the above improvements, has been developed. An improvement for geometry identification is accepted in the code, in which a system is divided into several superboxes, each of which may contain a different type of geometry specification. Some superboxes

are divided into many boxes. When stacking boxes, the mesh sizes of boxes in each superbox can be selected independently. By introducing the idea of a superbox, it became easier to specify the system geometry with box types, and required computing time became shorter.

The general flow diagram of JACS is shown in Fig. 1. As shown in this figure, a cell calculation is used before a core calculation when a system configuration to be calculated is too complex to analyze the phenomena with only one transport code. The group constants collapsed over the cell on energy and space are arranged for core calculation by S_n codes (ANISN-JR or DOT3.5), Monte Carlo codes (KENO-IV or MULTI-KENO), or diffusion codes [FEDM or ADC (Ref. 8)]. The YENMA code in Fig. 1 is an evaluator of criticality safety for the system to be checked with a data pool that contains many computed results—more than 700 cases of benchmark problems. With YENMA and the results computed on the system by JACS, the probability of the system being subcritical and its reliability are obtained.

JACS has been evaluated with many benchmark calculations, and the first version will be published in this year.

Fig. 1. General calculation flow of JACS.

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A Study on a Nuclear Criticality Safety Evaluation Technique, Y. Naito, J. Katakura, M. Yokota (JAERI-Japan)

A system whose criticality should be checked is often too complex to find directly applicable experimental measurements. In such a case, computer codes are used to evaluate the criticality safety. The codes and data used must be validated by many benchmark calculations. A computer code system named JACS (Ref. 1) has been developed at the Japan Atomic Energy Research Institute (JAERI) for evaluating nuclear criticality safety, and more than 700 benchmark calculations, as shown in Table I, were carried out to validate it. The computed k_{eff} 's of the calculations were widely spread (0.90 to 1.03), and in some cases absolute values of k_{eff} were far from 1.0. Hence, two problems are submitted, one on how to improve the calculation method and another on how to evaluate the criticality safety of a system from the result computed by the code. A study has been carried on to answer the latter problem.

The effective multiplication factor, k_{eff} , of a system is expressed as

$$k_{eff} = \frac{\iint \nu \Sigma_f \phi dE dr}{\iint \Sigma_a \phi dE dr + \text{Leakage}}$$

The above relation shows that k_{eff} depends on three important quantities: neutron flux distribution, nuclear cross sections, and neutron leakage. Neutron flux distribution

Fig. 1. Histogram of effective multiplication factors.

is continuously variable with every change of neutron energy and space, so that it is inadequate to use for an index to determine the characteristics of neutron reaction phenomena. The most important neutron energy spectrum is that in the fuel regions of the system. Then, fission rates of each energy group are chosen for indices against the neutron flux distribution, that is, fast fission rate, F_1 , epithermal fission rate, F_2 , and thermal fission rate, F_3 . To take into consideration nuclear cross-section uncertainties, benchmark problems are classified by the kind of fissile materials, that is, uranium, plutonium, and so on. The benchmark problems are also classified by the ratio of neutron leakage to absorption. The above three important parameters—fission rate, kinds of fissile materials, and leakage ratio—are chosen for grouping a system. In consideration of results computed for benchmark problems, the boundary values of fission rate are determined, and benchmark problems are classified into five groups:

- Group 1 $F_3 \geq 0.7$
 Group 2 $0.4 < F_3 < 0.7$, $F_2/F_1 \geq 5.0$
 Group 3 $0.4 < F_3 < 0.7$, $F_2/F_1 < 5.0$
 Group 4 $F_3 \leq 0.4$, $F_2/F_1 < 5.0$
 Group 5 $F_3 \leq 0.4$, $F_2/F_1 \geq 5.0$

Classification of the benchmark problems by leakage ratio is carried out in the same manner. Finally, the number of classification groups becomes 30. Each group has many benchmark problems, with a mean value of k_{eff} for each group. As shown in Fig. 1, the distribution of the number of cases between k_{eff} and $k_{eff} + \Delta k_{eff}$ has a noncentral normal distribution. With this distribution, the bias and limiting values of k_{eff} are obtained for evaluating the criticality of a system that belongs to a specific group. If the computed k_{eff} of a system with the bias is smaller than

TABLE I

No.	Title	Number of cases
1	A Pile of PuO ₂ -UO ₂ -Polystyrene Compact Fuel	51
2	UF ₆ Container Cylinder Critical Array	81
3	Slab-Cylinder Critical Configuration of Uranium Nitrate Solution	128
4	Cylinders or a Tank Containing Uranyl Nitrate Solution	76
5	Plutonium Fuel in various shape	33
6	Water Moderated Triangle Lattices of Plutonium enriched Rods	28
7	Intersecting Cylinders of Aqueous Uranyl Fluoride Solutions	51
8	Three Clusters of UO ₂ Rods Lattice	48
9	A Light Water Reactor Critical Assembly	125
10	A Critical Mixed Oxide (UO ₂ -PuO ₂) Fuel Lattice Moderated by Borated Water	12
11	Water Storage of Power Reactor Fuel in Close Proximity	21
12	A NxNxN Array of Compacted Lowenriched Uranium Oxide (U ₃ O ₈) Packages	10
13	An Array of Beryllium Oxide Fuel Pins in Uranyl Nitrate Solution	18
14	Plutonium Nitrate Solution of High Fissile Content Poisoned with Gadolinium	15
15	A Critical Assembly of Mixed Oxide Fuel Pins Immersed in Water	5
16	An Array of Tin Cans Containing UO ₂ -Alcohol Slurry	35
17	A Subcritical Assembly of Slightly Enriched Tubular Uranium Slugs	21
18	A Homogeneous System of U+Pu Nitrate Solution Containing Gadolinium	40
19	Subcritical Clusters of 2.35wt% and 4.25wt% U-235 Enriched UO ₂ Rods in Water with Uranium or Lead reflecting Walls	20

the limiting value, the system is judged to be subcritical. This method is embodied in the YENMA code of JACS. To validate the YENMA code, a system was chosen from the above benchmark problems that is just critical; therefore, the computed result must not show subcritical. Conversely, we set limiting values with which all benchmark problems are judged to be not subcritical. These values are not so conservative as to impede economical design. We think, therefore, that these limiting values may be accepted for determining a realistic limit value of k_{eff} for safety evaluation.

I. J. KATAKURA et al., "Development of the Computer Code System JACS for Criticality Safety," *Trans. Am. Nucl. Soc.*, **41**, 329 (1982).

MONK 6—A Revised Version of the U.K. Criticality Code, D. E. Bendall, R. J. Brissenden (AEE-U.K.), P. Hague, G. Walker (UKAEA-U.K.)

INTRODUCTION

Accurate and reliable prediction of the reactivity of complex geometries involving fissile materials is important

in nuclear criticality safety. The U.K. MONK Monte Carlo code has been widely employed for this purpose for a number of years, during which time it has been progressively developed, culminating in MONK 5 (Refs. 1 and 2) in 1978. MONK 6 is the product of a further phase of development aimed principally at improving the nuclear data, although a significant effort has also been applied to improving the code performance and user image. This paper describes these developments and the results of some of the validation calculations that have been made.

CODE DEVELOPMENT

The principal feature of MONK 6 is the incorporation of a new "point" form nuclear data base. The POND system of collision processing in MONK 5 has been replaced by the DICE modular coding package, which achieves enhanced accuracy via a histogram representation of all primary reaction cross sections using a fixed mesh. DICE is linked, via the program MOULD, with the UKNDF Library, which contains up-to-date evaluations of differential cross sections.

A new exact point energy treatment of neutron thermalization has been developed for scattering by monatomic gases. This model has been modified to take into account molecular binding effects when applied to scattering by

TABLE I
Results of MONK 6 Validation Calculations

EXPERIMENTAL SYSTEM	k_{eff}	REFERENCE
URANIUM		
Bare 93% U_{235} Metal Sphere	1.0093	LA 4208
93% U_{235} Metal Sphere in Water	1.0025	Trans ANS <u>27</u> , 412-413
Bare 93% U_{235} Nitrate Cylinders	1.0120	ORNL-TM-719
4.75% U_{235} Oxide lattices in water	1.0044	Nucl. Sci. & Eng. <u>71</u> , 154-163
2.35% U_{235} Oxide lattices in water	1.0012	Nucl. Tech. <u>47</u> , 51-58
5% U_{235} Fluoride Solution; in Plexiglass	1.0146	Y-DR-129
3% U_{235} UF_4 /wax compacts in wax	1.0288	Y-DR-14
PLUTONIUM		
Bare 20% Pu_{240} Metal Sphere	1.0022	LA 4208
Metal Sphere in water	0.9875	Nucl. App & Tech, <u>7</u> (5)
4.6% Pu_{240} Nitrate	1.0110	Nucl. Sci & Eng, <u>25</u> (2) July 66
Nitrate Sphere in concrete	1.0399	Nucl. Sci & Eng, <u>25</u> 165-173
Oxide/polystyrene compacts in Plexiglass	1.0283	Nucl. App. <u>6</u> , Jan 69
MIXED URANIUM/PLUTONIUM		
Oxide lattice in water	1.0363	Nucl. Tech. <u>44</u> , 141-145
Oxide lattice in water & concrete	1.0397	PNL-SA-7498
Nitrate cylinder in water	1.0023	Trans ANS, <u>17</u> , 269
Oxide/polystyrene compacts in Plexiglass	1.0364	Nucl. Sci & Eng, 61 pp370 - 376

light and heavy water and graphite. Studies made to establish the accuracy of this treatment include calculations of the mean angle of scatter as a function of incident energy and temperature, and also diffusion area. In the latter case, results of calculations have been compared with experimentally derived values.

A hierarchical stabilized random number generator system is now employed in MONK 6, and the use of double-precision arithmetic for both geometry and fission scoring has been introduced to ease problems associated with code utilization on different computers and the checking of minor modifications that are always required in a code of this size and complexity. It is now possible to check code operation by running a range of calculations for a very small number of stages.

Via a combination of

1. individual fission scoring
2. Fibonacci powering techniques
3. track-length-type estimation, which will work in "hole" regions where actual track lengths are not computed
4. modifications to the form of the k_{eff} estimator,

variance reduction by a factor of between 2 and 3 has been achieved with MONK 6 for the same computational cost level (by comparison with MONK 5) for most problems, and significantly larger economies have been achieved in some instances.

Improvements to the code geometry package have been made in MONK 6 to permit free orientation of simple bodies, and a wider range of initial neutron source distributions has been introduced to facilitate reduction in the number of settling stages.

To ensure that code input is correctly specified, an interactive color graphics geometry and material specification package has been added, and code output edit facilities have also been improved.

Further developments in hand include various group options, with direct links to Hansen-Roach, WIMS, ENDF/B, and other data sets, automatic k_{eff} search and optimization techniques, and other work on data and modeling aimed at the improvement of MONK 6 predictions for plutonium and mixed uranium/plutonium systems.

VALIDATION OF MONK 6

A range of validation calculations has been made for a number of experimental critical systems involving different fissile materials. The main emphasis has been on water-moderated and -reflected systems, although others have been examined. A selection of the results of these calculations is given in Table I.

Predictions for uranium systems are generally good, and the overestimation of k_{eff} that was generally a feature of calculations made with earlier versions of MONK has been significantly reduced. MONK 6 also performs better on most plutonium systems that have been studied thus far, but previous MONK 5 overprediction levels for mixed systems are only slightly reduced.

The k_{eff} levels quoted in Table I are mean values, and associated standard deviations are within the range from 0.005 to 0.01 for tracking 2.0×10^4 neutron histories.

CONCLUSION

MONK 6 represents a significant advance over previously available versions of this code, and both performance and accuracy have been improved. Continuation of the present U.K. development program is foreseen to yield further enlargement of the range of code user options, and to provide further improvements in the predictions of reactivity for certain fissile material combinations.

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Fault Tree Methods in the Criticality Assessment of the Transuranic Waste Treatment Facility, H. A. Worle (EG&G Idaho)

Large, complex facilities that process fissile materials present a special challenge to criticality safety specialists. As fissile materials pass through the process chain, failure of controls in one area may propagate to other areas. One such facility is the Transuranic Waste Treatment Facility (TWTF) planned for the Idaho National Engineering Laboratory (INEL). This paper discusses the use of fault tree methods in the criticality assessment of such facilities by the example of the TWTF.

The TWTF will receive waste contaminated with transuranic nuclides (TRU waste) presently stored at the INEL Radioactive Waste Management Complex. Waste packages are opened and their contents are sorted, placed into charging containers, and charged into an incinerator. Combustibles will be burned and noncombustibles converted to a chemically inert, basalt-like substance. This substance is cast into ingots and shipped to a federal waste repository. Because of the wide variety of shapes, sizes, and compositions of the waste, heavy reliance is placed on fissile material mass limits in the various process areas. The individual areas interact in the sense that an overload or accumulation of fissile material in one area may be propagated to the next area in the process chain.

Fault tree methods were used to analyze this complex facility to reduce the required assay points and to ensure full coverage of the facility by the double-contingency principle. That principle states that defense against a criticality accident must incorporate sufficient barriers so that no single failure can by itself produce the accident.

In a fault tree diagram, the final outcome (in this case, a criticality accident somewhere in the facility) is placed at the top of the diagram and the causes leading to this outcome are traced back to the smallest level of detail desired. The flow of events backward from the final outcome quickly spreads out into many branches. Individual events (or causes) are connected with each other by logic "gates" of two types. An "AND" gate produces an outcome only if all of the inputs are present; an "OR" gate produces its outcome whenever any of the inputs is present. The first step in diagramming a complex system such as the TWTF is to place the final outcome at the top and then to list all the areas where a criticality is possible. These "events" (a critical system in any one of these areas) are connected by an OR gate to the final outcome (a critical system somewhere in the facility). At this point, the tree can help decide if all areas are covered, or if areas can be eliminated from the list by ensuring criticality safety via physical controls. Each event below the OR gate on the initial tree is then made the outcome of a further developed branch of the tree. Figure 1 is an example of this development. Causes can be traced to any desired degree of detail, as illustrated by Fig. 2.

Coverage by the double-contingency principle is ensured when every pathway from basic cause to final outcome goes through at least one AND gate. Sometimes events are not true "faults" in the sense that conservative safety philosophy demands that worst-case conditions be assumed as normal operating conditions in the absence of controls.

Fig. 1. Fault tree branch for a single area.

Fig. 2. Further development of a subbranch.

An AND gate that joins such an event with just one other fault must be ignored in the analysis for double-contingency coverage.

Where mass limits are the main defense against criticality, assay and inventory records are the important control measures. Fault tree analysis shows where the most effective point of application for an assay system is to cover several areas with one system. This is done by tracing the source of an overbatch to the lowest level cause, which often is the same for different areas. For complex facilities with many interacting process areas, fault tree analysis can be a most valuable tool of the safety professional.

Analysis of an Oil-Uranium Deposit in Gaseous Diffusion Equipment, K. R. Yates, R. G. Jung, R. S. Denning (BCL)

As part of an effort to develop a final safety analysis report of the Portsmouth Gaseous Diffusion Plant, operated by Goodyear Atomic Corporation, criticality safety studies of the high-assay portion of the cascade were conducted. An event of particular interest to the safety analysis, which occurred several years ago, is described and analyzed herein.

One of the major concerns of nuclear criticality safety is the unplanned accumulation of a significant mass of well-moderated fissile material. Events of this type typically have a low probability of occurrence, are rarely uncovered in a safety analysis, and may or may not lead to a critical excursion, depending on a variety of factors. In 1973, such an unplanned accumulation of a significant quantity of high-assay uranium occurred as a well-moderated oil-uranium mass within cascade equipment. The analysis that follows shows why criticality did not occur and examines the effects of uranium density and depth of material on k_{eff} .

A brief description of the event follows. The deposit of oil-uranium was detected due to operational problems that developed as a result of restricted flow within a cascade cell containing UF_6 enriched to $>97\%$ ^{235}U . Beginning near a control valve in the 8-in.-diam interstage piping, the deposit essentially filled ~ 6 linear ft of piping and extended into a compressor. Investigation revealed that, due to vibration, a cascade interstage piping instrument line had rubbed against a hydraulic oil supply line, creating a small hole in each line. Over a period of time, oil slowly leaked into the cascade through the damaged instrument line. Removal of the equipment revealed 55.68 kg (122.5 lb) of a black

tar-like substance containing 11.6 kg (25.5 lb) of uranium, primarily as UF_4 . Approximately 36.8 kg (81 lb) of material containing 7.64 kg (16.8 lb) of uranium was in the pipe, and ~ 19.1 kg (42 lb) containing 3.95 kg (8.7 lb) of uranium was distributed within the compressor. As a result of this event, each cell in the X-326 process building, containing the high assay portion of the cascade, was inspected, and instrument lines and hydraulic oil lines were rerouted to effectively preclude future events of this type.

To analyze the uranium-oil accumulation, several KENO-IV (Ref. 1) calculations were made. XSDRN 123-group cross sections and uranium, as UF_4 , at 97.65% ^{235}U were used in all calculations. For calculations involving either the pipe or the compressor, a concrete reflecting plane 6 in. from the equipment surface, simulating the concrete floor, was included. Actual compressor shell geometry was used, with internal parts simulated by disks.

Since $k_{eff} \approx 0.96$ for the 8-in. pipe, with contents as specified above [36.8 kg (81 lb) U-oil], an area of concern regarding the oil-uranium accumulation was the relationship of k_{eff} to different levels of moderation. The results of KENO-IV calculations are presented in Fig. 1. These results indicate that the mixture was at, or near, optimum moderation for the pipe geometry, and that changes in moderation did not significantly increase k_{eff} .

The 19.1 kg (42 lb) of uranium-oil contained within the compressor was distributed over a large portion of the internal compressor surface. If during compressor removal and disassembly the uranium-oil material settled to the bottom of the compressor housing, it would approximately occupy a segment of a 20-in.-diam sphere having a depth of ~ 5 in. The k_{eff} of such an accumulation was found to be well below critical. Because significantly more uranium-oil mixture could conceivably accumulate in a compressor, several additional KENO-IV calculations were made with

Fig. 1. The k_{eff} versus relative uranium concentration for 6 ft of 8-in. pipe filled with UF_4 -oil mixture.

Fig. 2. The k_{eff} versus enrichment for a compressor filled to depths of 7 and 9 in. with a UF_4 -oil mixture.

a 7- and 9-in. depth of uranium-oil mixture in the bottom of a compressor. Figure 2 presents the relationship of k_{eff} to ^{235}U assay for both 7- and 9-in. depths of uranium-oil mixture. Criticality appears possible at 48% ^{235}U assay for a 9-in. depth of material. For a 7-in. depth of material, criticality may not be possible.

The accumulation of a uranium-oil deposit in cascade equipment has shown that the interaction of auxiliary systems with the cascade may be very subtle. Criticality was prevented because of the relatively small diameter of the cascade piping and the absence of extensive close reflection. Detection and prevention of subtle auxiliary system failures is a difficult task. Nevertheless, careful consideration of auxiliary system failures must be undertaken during any facility design or subsequent safety analysis to avoid hazardous situations with fissile material.

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Close-Packed Critical Experiments Analysis, E. L. Maceda, F. G. Welfare (B&W)

This paper describes a series of calculations performed on a set of newly completed experiments designed to simulate high-density storage of spent fuel pins removed from the fuel bundles. This work has been sponsored by the U.S. Department of Energy (DOE) as part of its away-from-reactor fuel storage program. Since the ability of the calculational techniques to reproduce critical data is important safety information, the analysis of the experiments was included as part of the DOE-sponsored work. In these experiments, the critical assemblies consist of a 5×5 array of specially designed modules containing low-enriched (2½%) uranium fuel.¹⁻²

Critical experiments simulating pin storage were performed with modules having three different packing densities. Arranged in order of decreasing density, they are as follows:

1. triangular pitch (T) equal to fuel pin o.d. (pins touching); 15 X 17 fuel pin matrix each module; intermodular spacing 1.78 to 3.81 cm (volume fraction water = 0.10).
2. square pitch (S) equal to fuel pin o.d. (pins touching); 15 X 15 fuel pin matrix each module; intermodular spacing of 1.78 cm (volume fraction water = 0.22).
3. square pitch (SO) 16.8% greater than fuel pin o.d.; 13 X 13 fuel pin matrix each module; intermodular spacing of 1.78 cm (volume fraction water = 0.43).

The neutron multiplication factor (k_{eff}) of each benchmark core was calculated using the Monte Carlo criticality code KENO IV (Ref. 3). The 123-group XSDRN cross-section set was used.⁴ The XSDRN library was generated by combining the GAM II library for epithermal groups and the THERMOS library for thermal groups. The XSDRN library was obtained from Oak Ridge National Laboratory in binary coded decimal (BCD) format, as was an auxiliary program, AIM, that was used to convert the library from BCD to binary format. The NITAWL code⁵ was used to calculate the resonance self-shielding for ^{238}U and to develop the working library for the KENO calculations. Some of the important resonance data for NITAWL input are:

1. Dancoff correction factor, C
2. effective moderator cross section per ^{238}U atom
3. first moderator scattering cross section per ^{238}U atom.

Three different Dancoff coefficients were used. The infinite Dancoff coefficient was calculated analytically,⁶ and the others used data from ANL-5800 (Ref. 7). The formula used was

$$C = C_{\infty} - \Sigma C_i$$

where C is the desired Dancoff coefficient, C_{∞} is the infinite Dancoff coefficient, and the C_i 's are Dancoff coefficients to be subtracted because of symmetry. For the square pitch modules, the eight closest pins (i.e., four contiguous and four diagonal pins) were used in the above formulation. Only the six contiguous pins were used in the triangular pitch modules.

The Dancoff correction factors used in NITAWL for each of the three core configurations were

Core	C_{∞}	C_1	C_2
I, II, III	0.78623	0.54383	0.42263
IV	0.59753	0.40677	0.29877
V	0.32520	0.22310	0.16260

Here, C_{∞} was used for interior pins; C_1 for edge pins in Cores IV and V and for even-row edge pins in Cores I, II, and III; and C_2 for corner pins in Cores IV and V and for staggered-row edge pins in Cores I, II, and III.

The cross sections for Cores I, II, and III had to be homogenized because the pins were loaded in a triangular geometry. The XSDRNPM code⁸ was used to homogenize the cross sections for these cases.

Quarter-core symmetry was used throughout these calculations. For the square pitch lattices, the computer representation explicitly modeled each fuel rod, aluminum rod, intermediate band (Core IV), center grid plate (Core V), side sheet, moderator, base plate, and core tank in the X-Y direction. The fuel support structure above the water was not represented in the computer model.⁹ The only difference in the modeling of the triangular pitch lattices (Cores I, II, and III) was that the fuel regions were homogenized except for the external pins in each fuel assembly. The outward half of each of these pins was explicitly modeled (see Table I).

Two conclusions can be drawn from the calculated results presented in the table:

1. When the spacing between fuel modules is 1.8 cm, the AMPX-KENO model accurately predicts k_{eff} for each of the three packing densities tested.
2. When the intermodular spacing is increased to 2.5 and 3.8 cm, the AMPX-KENO model significantly underpredicts k_{eff} , and this underprediction seems worse at the wider spacing.

The second finding was made only for the T-type modules, since no cores were built with the S- or SO-type modules that used spacings wider than 1.8 cm. Nevertheless, this finding is consistent with the results reported in Ref. 1, in which the AMPX-KENO model was found to underpredict k_{eff} at intermodular spacings >1.64 cm. It should be noted that this discrepancy was found both for cores using the highest fuel packing density possible (T-type modules) and for cores using fuel modules simulating the packing density of conventional light water reactor assemblies.¹⁰ This could be due to the cross-section set used in the calculations.¹¹

Two things we cannot ascertain from these results are (a) the exact size of the discrepancy (data indicate ~2%, but there is ~0.6% statistical uncertainty in this number), and (b) whether or not the discrepancy will worsen at even wider spacings or, if so, by how much. This discrepancy is significant, however, and should be considered when confidence limits are established for criticality calculations of storage systems using these wider spacings.

TABLE I

Benchmark Loading	Intermodular Spacing	Module Water Volume Fraction	Calculated $k_{eff} \pm 1\sigma$
Core I	1.778/1.945 ^a	.10	1.002 $\pm$ 0.007
Core II	2.539/2.709 ^a	.10	0.984 $\pm$ 0.007
Core III	3.807/3.976 ^a	.10	0.979 $\pm$ 0.006
Core IV	1.778	.22	0.996 $\pm$ 0.005
Core V	1.792	.43	1.003 $\pm$ 0.004

^aBecause of the triangular geometry, the pins on two edges of the T-type modules were evenly aligned and those on the other two edges were staggered. The first spacing quoted is the separation between staggered row edges of adjacent modules, and the second is the distance between even-row edges of adjacent modules.

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Validation of Criticality Calculation Methods for LWR Storage Pools, D. Doutriaux, M. Roshd, S. Zero (Framatome-France)

To increase the capacity of fuel storage pools, Framatome is developing high-density fuel storage. To qualify Framatome's criticality calculation methods, in accordance with the American National Standards Institute's N 16.9-1975, several critical experiments were selected for comparison with corresponding Monte Carlo calculations. Table I gives details of these experiments, which were performed with enrichments varying from 2.35 to 93.2%, and with a large range of absorber slabs between fuel assemblies: water, borated water, stainless steel, borated stainless steel, boral, and cadmium.

These critical experiments were undertaken in several American (Oak Ridge National Laboratory,¹ Pacific Northwest Laboratory,² and Babcock and Wilcox³) and French laboratories [Commissariat à l'Energie Atomique⁴ (CEA)]. The French results were obtained during an important experimental program intended to study the criticality safety of pressurized water reactor (PWR) fuel elements.

This paper is essentially concerned with the assessment of Framatome calculational methods by comparison with results from these critical experiments. However, Framatome is also engaged with the CEA in a joint experimental program, CRISTO 2, which should give further results on PWR fuel assemblies, in experimental conditions equivalent to those encountered in a high-density fuel storage pool.

For each experiment, the effective multiplication factor k_{eff} is computed by running the three-dimensional Monte Carlo code KENO IV (Ref. 5), with a 16-group Hansen and and Roach cross-section library,⁶ taking into account resonances, self-shielding, and the Dancoff correction.

The geometry description used in KENO is as accurate as possible. When approximations were made, e.g., in the reflector, sensitivity studies were performed to demonstrate the validity of these approximations.

The fuel cells are homogenized such that they conserve equivalence with heterogeneous cells. All structures (absorbers between assemblies, lower and upper reflectors) are represented in a discrete geometry. The dry fuel above critical water level is also represented in the calculational model.

TABLE I
List of Experiments and Monte Carlo Results

Experiment	Enrichment	Inter-Assembly Gap (cm)	Absorber Material	Monte Carlo k_{eff}
ORNL - 11,1	93.2	13.74	—	1.006
ORNL - 11,2	93.2	15.17	—	.994
ORNL - 11.5	93.2	23.48	—	.997
BNWL - 14	2.35	8.41	Water	1.009
BNWL - 20	2.35	6.34	Boral	1.015
BNWL - 28	2.35	6.88	Stainless steel	1.008
B&W - 2245	2.46	0	Water	.995
B&W - 2255	2.46	0	Borated water	.997
B&W - 2268	2.46	1.636	Borated water	1.017
B&W - 2355	2.46	1.636	Borated water	1.017
V - 1762	4.75	4.0	Water	1.001
V - 1913	4.75	4.7	Stainless steel	1.000
V - 1899	4.75	1.7	Boral	0.994
V - 1900	4.75	2.7	Boral	0.991
V - 1902	4.75	4.7	Boral	0.988
V - 1859	4.75	4.8	Cadmium	0.990

Fig. 1. Monte Carlo results.

In the Monte Carlo calculation, the initial guess flux distribution is flat and ~100 batches of 300 neutrons are run, the first three batches being skipped to eliminate the effect of the initial flux distribution.

Table I and Fig. 1 give the KENO values of k_{eff} for the selected experimental configurations. The average k_{eff} of the calculated experiments is 1.001, which demonstrates that there is virtually no bias associated with the method.

Figure 1 shows that the computed values of k_{eff} are uniformly scattered around 1.00, and that they are not affected by spacing or the type of absorbing material between assemblies. Hence, it is clear that there is no systematic error in the method.

Taking into account statistical uncertainties and experimental errors, there is a 95% probability with a 95% confidence level that the uncertainty in k_{eff} due to the method is not greater than ± 1600 pcm. This wide range analysis shows the reasonably accurate results in fuel storage criticality predictions obtained by the KENO-IV Monte Carlo code and the appropriate choice of cross-section data set, which takes into account the heterogeneity of the cells.

Criticality of High-Density, Boral-Poisoned Fuel Racks for Maine Yankee, D. G. Napolitano, R. J. Cacciapouti (YAEC)

Maine Yankee, an 830-MW(e) pressurized water reactor, is seeking to increase the number of spent fuel locations from the 953 now existing to 1476 by decreasing the canister center-to-center spacing. Maine Yankee now (1982) has 577 assemblies in its spent fuels racks and will lose full-core reserve capability in 1984 and reload discharge capability in 1987. Therefore, Maine Yankee is seeking to increase storage capacity by reracking and pin compaction, such that the equivalent of 2038 spent fuel assemblies can be stored in the spent fuel racks.

Historically, Maine Yankee has reracked once from the original 20.5-in. center-to-center spaced unpoisoned racks to a 12.0-in. center-to-center spaced, boral plate, poisoned rack design. This increased the capacity from 318 assemblies to the present 953-assembly capacity. The criticality analysis of the present racks was studied by Denver et al.¹ Their analysis indicated that the k_{eff} for the nominal configuration at the most reactive pool temperature (68°F) and fuel enrichment of 3.2 wt% was 0.7558. This multiplication factor is highly subcritical and allows further reracking.

The proposed rack design differs from the present in two major aspects: a decrease in canister center-to-center spacing from 12.0 to 10.25 in. and the use of double-walled stainless canisters instead of U-shaped aluminum channels. Boral plates are used in both designs as poison, but the boral plates in the present design are sealed in between the inner and outer canister rack walls on all four sides of the canister (see Fig. 1). Both the present and proposed design utilized a flux trap principle to control criticality. A water gap between canisters thermalizes fast neutrons leaking from stored assemblies, and the thermal neutrons are captured by the boral poison. The nominal mechanical design assumes fresh Maine Yankee fuel assemblies of the Exxon type² are placed in the canisters.

Criticality analysis of the proposed Maine Yankee spent fuel racks was done with two methodologies: NITAWL-KENO with the 123-group XSDRN library³⁻⁵ and LEOPARD-ROD WORTH-PDQ (Refs. 6, 7, and 8), with four-group collapsed cross sections. In both methods, a two-dimensional (x-y) model of the rack was constructed from a basic unit of the rack. The basic unit of the rack is one-quarter of a fuel assembly and rack unit. Calculations are done assuming no leakage in either axial or radial directions, i.e., reflecting boundary conditions. In addition, the models for Maine Yankee's spent fuel racks employed "worst-case" mechanical uncertainties with fresh fuel at 68°F to bound reactivity effects.

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TABLE I

Enrichment (w/o U^{235})	Center-to-Center Spacing (In.)	Boral Boron Loading (atm/b-cm)	Flux Trap Void (%)	k_{eff}		
				KENO-IV ^a	PDQ-7	ΔK
3.0	10.1875	0.039737	0	0.9150 $\pm$.0046	0.8794	0.0356
3.5	"	"	"	0.9329 $\pm$.0047	0.9117	0.0212
4.0	"	"	"	0.9598 $\pm$.0054	0.9380	0.0218
3.5	10.00	"	"	0.9678 $\pm$.0050	0.9408	0.0270
"	10.25	"	"	0.9283 $\pm$.0051	0.9028	0.0255
"	10.50	"	"	0.8850 $\pm$.0057	0.8706	0.0144
"	10.1875	0.029022	"	0.9392 $\pm$.0047	0.9226	0.0166
"	"	0.051442	"	0.9293 $\pm$.0046	0.9029	0.0264
"	10.25	0.039737	25	0.9504 $\pm$.0047	0.9277	0.0227
"	"	"	50	0.9856 $\pm$.0052	0.9560	0.0296

^aEach run was approximately 18 800 histories.

Fig. 1. Maine Yankee spent fuel rack—two-dimensional layout.

The criticality of the proposed rack design was studied as a function of enrichment, canister center-to-center spacing, boron plate boron loading, voiding of flux traps, pin compaction, fuel burnup, and abnormal configurations. Results from both NITAWL-KENO and LEOPARD-RODWORTH-PDQ analysis are shown in Table I for variations in enrichment, canister center-to-center spacing, boron boron loading, and flux trap void. From enrichment analysis, 3.5-wt% ^{235}U fuel was selected as the highest enrichment allowable in the racks, which produces $k_{\text{eff}} < 0.95$ with uncertainties. Further analysis was done with this enrichment. Pin compaction from the present 14×14 , 0.65-in. pin pitch array with 3.5 wt% ^{235}U fuel to a 17×17 , 0.5-in. pin pitch array provided a 10% $\Delta\rho$ reduction in reactivity. Also, fuel burned to an average discharge exposure of 30 to 36 K MWd/MTU provided a 25 to 26% $\Delta\rho$ reduction in reactivity.

Both NITAWL-KENO and LEOPARD-RODWORTH-PDQ show nearly the same changes in reactivity due to changes in design; however, the PDQ model consistently underpredicts the KENO model k_{eff} by 2 to 3% ΔK . This underprediction of k_{eff} by PDQ is due, in part, to an inadequate representation of the boron environment and, therefore, the collapsing spectrum, which is input to the RODWORTH code and used to generate boron cross sections. If a 3% ΔK uncertainty is applied to the LEOPARD-RODWORTH-PDQ methodology, it can be used with some degree of confidence to study changes in design, but at considerably less expense than NITAWL-KENO.

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Nuclear Criticality Safety of Dissolver Vessels

A Calculation Route for Criticality Assessments of Oxide Fuel Dissolvers, R. T. Keay, P. D. Clemson, M. Dunn (BNFL-U.K.)

A typical dissolver design for thermal reactor oxide fuel dissolution involves shearing of irradiated fuel elements and dissolution of the fragments in nitric acid solution. From a criticality standpoint, this results in a heterogeneous fissile system of fuel particles in fissile liquor. The criticality safety assessment is complicated by the possible variation in the size and distribution of sheared fuel particles and, hence, the large variation in the degree of thermalization of the neutron spectrum that can arise. The paper details a calculation route developed using the U.K. code suite WIMSE and its associated nuclear data, which enables libraries of resonance self-shielded multigroup data to be generated for the different regions in the dissolver. The data can be spatially smeared and condensed to few-group form for use in rapid diffusion theory solutions, thus affording an economical approach to survey work. Selected geometries can then be reassessed using the original multigroup data in an explicit geometry Monte Carlo calculation.

The resonance treatment for such selected isotopes as ^{238}U is via a subgroup method, which subdivides the resonance energy groups and performs independent collision probability solutions for each subgroup to obtain a set of spatial flux distributions. The effective group cross sections are then obtained by recombining the subgroup results using lethargy widths as weighting functions. The collision probability solution can be applied to regular packed spherical fuel particles or to arrays of fuel pins.

The resulting self-shielded multigroup cross-section data can be used directly in the Monte Carlo solution or collapsed and smeared for the few-group diffusion solution. In the latter case, the collision probability solutions can be used to obtain the condensing spectrum. After smearing using flux-volume weighting, the data are condensed to few groups and can be used in a finite difference diffusion solution.

For calculations relating to post-dissolution, the self-shielded data can be condensed and used in a reduced group Sn transport solution. The total calculation scheme uses data interfaces, enabling easy manipulation of data and construction of the calculation route.

The novel feature of this work is the extension and validation of cheap and efficient codes developed for reactor physics applications to cover design aspects of nuclear criticality safety for chemical plants. Illustrative results have been produced for application of the calculation route to a conceptual design of batch fuel dissolver using gadolinium as a soluble neutron poison. An assessment has been made of the sensitivity of reactivity to fuel particle size and packing fraction and also the relative proportions of dissolved and undissolved fuel (see Fig. 1). The results indicate that when poison is present in the system, peak reactivity occurs at a significantly higher packing fraction than for an unpoisoned core. In addition, the presence of dissolved fissile material reduces reactivity relative to a system of fuel particles in water.

The calculation route has also been applied in a validation program including correlations made with Hanford critical uranium/plutonium solution systems as well as arrays of fuel pins in uranyl nitrate solution (Table I). The results

TABLE I
WIMSE Validation Results for Critical Homogeneous and Heterogeneous Systems

A. Critical Cylindrical Solution Systems of Uranium/Plutonium Nitrate									
Reference	Uranium Data		Plutonium Data		Specific Gravity (kg/m ³)	Gadolinia Concentration (kg/m ³)	Critical Height (m)	Calculation Details	
	Concentration (kg/m ³)	Enrichment (^{235}U wt%)	Concentration (kg/m ³)	Enrichment (^{239}Pu wt%)				Axial Extrapolation Length (m) Assumed	WIMSE K _{eff}
1	---	---	116	90.68	1.2552E+3	0	15.44E-2	8.0E-2 ^a	1.0332
1	---	---	116	90.68	1.2552E+3	0	15.44E-2	6.3E-2 ^b	0.9787
2	176.8	0.67	77.3	93.78	1.462 E+3	1.06	75.32E-2	6.3E-2	1.0072
1	---	---	116	93.78	1.2552E+2	2.38	80.37E-2	6.3E-2	1.0010
B. Critical Array of 4.3 wt% UO ₂ Rods in Uranyl Nitrate Solution									
Reference	Number of Fuel Rods	Triangular Lattice Pitch (m)	Gadolinium Concentration (kg/m ³)		Critical Solution Height (m)		Calculation Results WIMSE 69-Group Monte Carlo Predictions		
4	433	0.028	0		0.309		0.9841 ± 0.0135		

^aClassical value for water.

^bValue from Ref. 3.

Fig. 1. WIMSE calculations of sensitivity of K_{∞} to packing fraction and liquor composition for 40 wt% UO_2 steel-clad pellets.

indicate good agreement on reactivity predictions for both homogeneous and heterogeneous systems.

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Criticality Safety of the Ten-Well Insert for the Pot Dissolver, James L. Forstner (SRP)

One of the tasks of the Savannah River Plant (SRP) is to recover uranium and certain transuranium isotopes from government-owned reactor fuels and from some foreign research reactor fuels. These fuels come in many shapes, sizes, materials, and enrichments. Nuclear safety for most fuels dissolved at SRP is assured by the use of some form of favorable geometry insert in a pot dissolver. The purpose of the work discussed here was to design an insert that would permit an adequate charge of highly enriched U-Al alloy fuels, especially the numerous material testing reactor (MTR) types, and yet require a minimum of administrative control.

Prior to the development of the Ten-Well Insert, several available inserts were used for dissolving nonplant fuels. A

special High-Flux Isotopes Reactor (HFIR) insert is still used for HFIR and RIIF cores, and an electrolytic insert is still used for steel- or zirconium-clad oxides and cermet. U-Al fuels were generally dissolved in HM-type inserts (multiple slabs). The latter had disadvantages for offsite fuels. Some fuels were received in containers too large to fit into the insert and required repackaging. The individual slab sections of the insert could accommodate four or more standard square tube bundles used for packaging MTRs and some tubular assemblies, while safe mass limits would only allow one or two per slab. Overcharging was prevented by administrative control. Although overcharging did not occur, there was much concern about the possibility.

It was decided that the new insert¹ should consist of a group of individual wells sized to receive one square tube bundle or one cylindrical container up to 12.7 cm in diameter. The wells must fit into a rectangular opening 126 X 42 cm, the opening in the top of the existing dissolver pots. Scoping calculations indicated that a 2 X 5 array of cylindrical wells constructed from 6-in., Schedule 120, Type 304L stainless steel pipe would provide adequate mass limits. A 2 X 4 array would provide larger limits, but other considerations would prevent their full utilization, while a 2 X 6 array resulted in inconveniently low well limits. Final limits were calculated on the basis of wells on a rectangular pitch 24.48 X 26.67 cm, 0.635 cm less in both directions than design nominal. Allowances were made for corrosion and pipe tolerances. Fifteen percent of the well wall was removed, with holes 2.54 cm in diameter for solution circulation.

The MGBS and TGAN codes² were used for the basic calculations. MGBS computes bucklings and two-group parameters for homogeneous media and for cylindrical cells. It contains a built-in 12-group cross-section library. TGAN performs two-group diffusion theory computations for slabs, cylinders, and spheres. The combination, MGBS-TGAN, has been extensively correlated with uranium exponential and critical experiments.² TGAN, a one-dimensional code, cannot be used for a finite array of cylinders, but can be used for a finite array of slabs. Cylindrical wells are treated as square wells with the same cross section and wall thickness. To calculate the average extrapolation distance in one dimension, the lines of wells perpendicular to that dimension were treated as infinite slabs. Extrapolation distance calculated separately for all three dimensions were used to calculate k_{eff} . Nitric acid was conservatively ignored in all cases. Check calculations were made with HRXN-II (Ref. 3), which contains a slightly modified Hansen-Roach 16-group cross-section set, and the Monte Carlo criticality code, KENO (Ref. 4).

Safe quantities of uranium in each well were calculated for U-Al alloys and UO₂-Al cermet as a function of weight percent uranium, uranium concentration in the pot solution, and well configuration. Several different well combinations are possible—all wells, isolated pairs, and isolated wells. Wells not to be charged are physically plugged. Table I illustrates the effect of these variables on the mass limits for U-Al alloys. Limits were also calculated for Pu-Al and for mixed U-Pu-Al alloys. From these masses, safe operating well limits were derived that recognize possible reduced safety margins during dissolving and that permit successive well charging for lightly loaded fuels.

Several different limits affect the charging of the insert. For the most common fuel, the MTRs, the square tube bundle is only long enough to hold six assemblies. In our storage facility, RBOF, we do not permit handling of more fissile material in a bundle than would be safe if the bundle broke and the assemblies assumed the most reactive configuration. For some heavily loaded assemblies, fewer than six can be loaded into a bundle. For a large variety of MTR assemblies, the safe mass of ²³⁵U is ~2 kg. The bundle

TABLE I
Safe Masses for U-Al Alloys in the Ten-Well Insert

Concentration ²³⁵ U in Pot Solution, g/l	wt% U ^a in Alloy =		
	50%	40%	30%
	Safe Mass ²³⁵ U, kg/Well		
All Wells, 2 X 5 Array			
0	2.09	3.25	∞
1	1.77	2.42	∞
2	1.53	1.95	4.37
3	1.33	1.60	2.67
4	1.17	1.37	1.98
Isolated Pairs, 1 X 2 Array			
0	3.85	∞	∞
1	3.00	13.0	∞
2	2.47	4.80	∞
3	2.08	3.30	∞
4	1.80	2.50	∞
Isolated Well			
0	∞	∞	∞
1	14.0	∞	∞
2	5.90	∞	∞
3	4.10	∞	∞
4	3.22	∞	∞

^aU is 93.5 wt% ²³⁵U.

loading must also meet the well limits for the well configuration that will be used. In addition, there is a pot limit to protect against inadvertent overconcentration of the pot solution through boildown.

The Ten-Well Insert is used to dissolve many fuels. It has a reduced dependence on administrative control, it provides a means to handle cylindrical fuel bundles up to 13 cm in diameter, and allows a certain amount of flexibility in charging because of the various well configuration options.

1. W. C. PERKINS and J. L. FORSTNER, "Nuclear Safety of the Ten-Well Insert for the SRP Fuel Element Dissolver," DP-1429, E. I. du Pont de Nemours and Company (June 1977).
2. H. K. CLARK, "Computer Codes for Nuclear Criticality Safety Calculations," DP-1121, E. I. du Pont de Nemours and Company (Nov. 1967).
3. H. K. CLARK, *Nucl. Sci. Eng.*, **79**, 65 (1981).
4. L. M. PETRIE and N. P. CROSS, "KENO-IV—An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Lab. (Nov. 1975).

Criticality Analysis of the ICPP Zirconium Dissolver Using MONK, I. E. Fergus (Exxon Nuclear Idaho), B. J. Layman (Rockwell Hanford)

This paper discusses how the safe ²³⁵U mass and boron limits were determined for the zirconium fuel dissolver at the Idaho Chemical Processing Plant (ICPP). This vessel, which has a 508-mm (20-in.)-o.d. cylinder with a complicated internal structure (Fig. 1), uses boric hydrofluoric acid to dissolve highly enriched uranium-zirconium alloy fuels.

Fig. 1. ICPP zirconium fuel dissolver.

An almost exact model of the dissolver was described using the MONK code.¹ For example, this dissolver has a slotted fuel charging chute with three different patterns—12, 23, and 34 equally spaced vertical slots at different elevations. Each slot was represented in its exact location and with its correct dimensions. The only difference is that the actual slots have rounded corners to minimize corrosion.

Fig. 2. Section of model near bottom of dissolver.

Figure 2 shows a section of the model near the crash plate at the bottom. The fissile material was then modeled as a sphere, the most reactive shape. This sphere is not homogeneous because it can be formed from two sources: undissolved fuel from the current batch and dissolver heel precipitates. The fissile material density within the part of the sphere in the chute is thus higher than without, and mass balance calculations were made to determine the sphere radius for each case of several parametric studies for different heel masses.

Four parameters were then studied: (a) the solids mass (SM) of the slurry sphere, (b) the radius (R_0) of the slurry sphere, (c) the metal volume fraction (MVF), and (d) the weight percent of uranium in the alloy, uranium enriched to 93 wt% in the ^{235}U isotope [wt% of U(93)]. The SM is the mass of ^{235}U in the form of $\text{U}(93)\text{F}_4 \cdot 2.5 \text{ H}_2\text{O}$ contained in the slurry sphere. This parameter was set equal to 3, 5, 6, and 9 kg in the analysis, and k_{eff} increased slightly over 10% from 3 to 9 kg ^{235}U . R_0 was varied over the range from 102 to 238 mm (4.0 to 9.38 in.) (the largest sphere that fits in the E-101 dissolver). The effects of decreasing R_0 were to decrease the H/U ratio, to decrease the B/U ratio, and to increase neutron leakage from the sphere. No matter what value was chosen for SM, the highest k_{eff} values resulted from spheres having radii in the range from 152 to 178 mm (6.0 to 7.0 in.). Spheres having radii <152 mm (6.0 in.) resulted in lower k_{eff} values due to increased neutron leakage. Spheres having radii >178 mm (7.0 in.) resulted in lower k_{eff} values due to overmoderation and increased poisoning. In this analysis, the MVF parameter was varied over the range from 0.0 to 0.8. The effects of increasing the MVF were to decrease H/U and to decrease B/U. The relationship between k_{eff} and MVF was found to be very complicated. Nevertheless, one definite trend should be mentioned. The k_{eff} values for the most reactive

spheres [having radii in the range from 152 to 178 mm (6.0 to 7.0 in.)] increased as the MVF decreased. In fact, MVF = 0.0 gave the highest k_{eff} values. Calculations were performed for fuels containing 3.2 and 1.6 wt% U(93). As expected, lower weight percent U(93) fuels resulted in lower k_{eff} values, with a difference of ~2%.

The safe mass of ^{235}U heel was found to be 5 kg ^{235}U for a charge mass of 16 kg ^{235}U , 1.6 wt% ^{235}U in the fuel, and 3.6 g B/l. Using the MONK code made this evaluation much easier than it would have been with other available calculational tools, and the code is currently being modified to be even more versatile.² Therefore, it is recommended that the MONK code be investigated by criticality safety personnel for applicability.

1. V. S. W. SHERRIFFS, "MONK—A General Purpose Monte Carlo Neutronics Program," SRD-R-86, Safety and Reliability Directorate, U.K. Atomic Energy Authority (1978).
2. G. WALKER, Private Communication (1981).

Criticality Safety in the ICPP Rover Fuel Dissolution System, E. Fast (Exxon Nuclear Idaho)

The Idaho Chemical Processing Plant (ICPP) Rover fuel dissolution system consists primarily of a dissolver vessel and a complexer vessel, along with a centrifuge and a number of smaller tanks used in product treatment. An annular tank formerly used as a combined dissolver and complexer¹ has been relegated to the role of complexer only, and a new vessel was designed as a dissolver. Designing this new dissolver vessel and rearranging other process vessels involved concurrent criticality safety evaluations.

Four designs of the dissolver vessel were proposed:

1. an "organ pipe" model of four 8-in. nominal diameter tubes connected at the bottom by a diagonal tube
2. a single 8-in. nominal diameter tube
3. two 6-in. nominal diameter tubes connected by a nominal 4-in. diagonal pipe
4. four 6-in. nominal diameter tubes in rectangular array connected by nominal 4-in. diagonal pipes.

In all cases, the tubes of the dissolver vessels were lined with 5.5-mm-thick Kynar (polyvinylidene fluoride resin) for protection against the action of nitric and hydrofluoric acids used for dissolution. The vessels were ~5 m high. Spacing between centers of the tubes in model 1 was 395.2 mm, in model 3, 533.4 mm, and in model 4, 549.3 and 457.2 mm (rectangular). The inner (solution) diameter in models 3 and 4 was 141.3 mm.

Rover fuel is in the form of graphite rods having dispersed UC_2 or UO_2 and some NbC and ZrC. The rods are burned, resulting in an ash of U_3O_8 , Nb_2O_5 , small amounts of carbon, and minor impurities. The ash is transferred batch-wise into the dissolver vessel. It is dissolved with 8 M HNO_3 , dilution water, and 22 M HF, all borated at 3.85 g B/l. Excess fluoride is removed by complexing with 2.2 M $\text{Al}(\text{NO}_3)_3$.

Calculations were directed toward determining criticality limits of various parameters. The KENO-IV Monte Carlo criticality code² was used with 16-group Hansen-Roach cross sections³ for all calculations. Dissolver models 3 and 4 proved to be geometrically critically safe for any concentration of uranium, with or without boron, and any degree of water reflection. The latter vessel, design 4, was chosen because of its larger capacity. Design studies also included the effect of tube spacing on k_{eff} . Conservative assumptions for calculations included using water as the

solvent to form a homogeneous mixture with ash that was assumed to be $\text{U}(93.5)_3\text{O}_8$. KENO calculational models represented the diagonal pipe sections with a series of cuboid approximations. Maximum k_{eff} was ~0.8 when the environment was dry or 0.92 when flooded. Partial reflection was also considered.

The annular complexer was modeled exactly except that the conical (bottom) section was represented by a series of cylindrical increments. This vessel was not critically safe by geometry. Boron soluble poison was required in the fuel solution to maintain $k_{eff} < 0.95$ for concentrations of ^{235}U above 80 g/l. Satisfactory experience with soluble poisons has been demonstrated.⁴

A KENO model of the entire Rover dissolution cell was used to calculate nuclear interaction between the various vessels. The interaction was ~5% of neutrons causing fissions in vessels other than where they originated.

Criticality safety limits of ^{235}U concentrations with normal boron present was shown to be an order of magnitude or more greater than was expected to be the case in normal operation. Expected flowsheet conditions yielded k_{eff} of ~0.21. Uncertainties in the process such as in transfer of the ash and its composition make the liberal tolerances desirable, however.

Criticality safety calculations made concurrently proved to be a desirable factor in the design of the revised Rover dissolution system and the determination of safe limits.

1. J. A. EGGERT, "Soluble Poisons and Chemical Processing Equipment at ICPP," *Trans. Am. Nucl. Soc.*, **19**, 184 (1974).
2. L. M. PETRIE and N. F. CROSS, "KENO-IV—An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Lab. (Nov. 1975).
3. G. E. HANSEN and W. H. ROACH, "Six and Sixteen-Group Cross Sections for Fast and Intermediate Critical Assemblies," LAMS-2543, Los Alamos National Lab. (1961).
4. R. E. WILSON and S. R. MORTIMER, "Experience with Soluble Neutron Poisons for Criticality Control at the Idaho Chemical Processing Plant," *Trans. Am. Nucl. Soc.*, **30**, 264 (1978).

Evolution of Safety Philosophy for the ICPP Electrolytic Dissolver Process, I. E. Fergus, J. E. Johnson (Exxon Nuclear Idaho)

At the Idaho Chemical Processing Plant (ICPP), highly enriched ^{235}U fuels clad and/or alloyed with stainless steel are dissolved in an electrolytic process.^{1,2} Nitric acid containing soluble gadolinium poison is recirculated from a cylindrical surge tank, through the trough-shaped dissolver, and back to the surge tank, as shown in Fig. 1. Fuel is dissolved by passing direct current through the nitric acid electrolyte; fresh acid addition and product take-off may be either batch-wise or continuous. Examples of the fuels dissolved in this system are Experimental Breeder Reactor II (EBR-II) fuel elements (uranium alloy rods), which are clad with stainless steel and are in stainless steel cans.

Recently, a new fuel can design was adopted that mandated a new criticality safety evaluation. The safety philosophy used was to assume that the cans had filled with water while in pool storage. The original worst-case accident scenario³ involved dilution of the gadolinium soluble poison from 4.3 to 2.6 g/l by water in the cans during fuel dissolution in a stagnant dissolver, along with dissolution of uranium to the saturation level of 610 g U/l. A quick look at

Fig. 1. Schematic diagram of electrolytic process.

the new can design indicated that using a similar philosophy would result in a threefold increase in the required gadolinium concentration, a very expensive proposition! So, a new approach was tried—controlling the process parameters such as can and acid addition rates, minimum volume of liquid in the system, and electric current on or off, depending on adequate recirculation. Calculations based on new worst-case scenarios resulted in a nearly threefold reduction to 1.5 g gadolinium/l.

The geometry and loading of the fuel cans limit the dissolver loading such that the worst-case ^{235}U concentration under stagnant dissolution conditions is ~ 230 g/l for fuel enriched to no more than 67 wt% in ^{235}U . Several conservative features were used in the calculational models to cover such concerns as localized poison dilution and interaction among the various nuclear units in the cell. These included assuming full water reflection and having the fissile material extend the full length of the elements. Other conservatisms arose while simplifying the model to a manageable size for input to the KENO-IV code.⁴

The following conditions were studied: dissolver full of internally water-filled cans, but no dissolvent; the same condition, but with the dissolver filled with poisoned acid; all elements from the same cans suspended in a lattice in the dissolver; and various stages of dissolution in the stagnant mode, that is, without recirculation. The most reactive

Several Rows Omitted

Fig. 2. Dissolver model showing optimum array.

condition was for partial dissolution, which is illustrated in Fig. 2. Since dissolution of individual elements proceeds rapidly after the cladding is breached, this model contains only undissolved elements and uranium-bearing solution. The cans are totally dissolved, but the elements are assumed to be suspended in optimum spacing in the upper section of the dissolver. These two conditions are mutually exclusive, but this model envelops all combinations that may occur during actual dissolution. Then, the optimum array was maintained while other parameters were varied. The failure point ($k_{\text{eff}} = 1.0$) for gadolinium is ~ 0.4 g/l. The safety limit is 1.5 g/l, which allows for 20% dilution of the poison in addition to the safety factor. This concentration of gadolinium was found to be adequate for the surge tank and the other vessels in the process, even allowing for evaporation to 150% of the previous concentrations of both uranium and poison. The safety is assured by the control system for this process, which requires adding cans and acid in a fixed ratio so that in the stagnant case, dissolution would cease before an unsafe mass could accumulate.

1. J. A. EGGERT, "Soluble Poisons and Chemical Processing Equipment at ICPP," *Trans. Am. Nucl. Soc.*, 19, 184 (1974).
2. R. E. WILSON and S. R. MORTIMER, "Experience with Soluble Neutron Poisons for Criticality Control at ICPP," *Trans. Am. Nucl. Soc.*, 30, 264 (1978).
3. G. F. OFFUTT, Ed., "Safety Analysis Report for the ICPP Electrolytic Dissolution Process," ICP-1009, Idaho Chemical Programs (1972).
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Nuclear Criticality Safety Practices in Digestion Systems of the Large-Scale Production Facility of the DOE at Fernald, Linda Capano-Dolan (National Lead Co of Ohio)

INTRODUCTION

Nuclear criticality safety in massive low-enrichment systems is a blend of common sense and practicality, as well as nuclear reactor physics. For almost 30 years, the classic principles of criticality safety, i.e., mass, geometry, and concentration, have been applied in the digestion areas of the U.S. Department of Energy (DOE) Feed Materials Processing Center, located in Fernald, Ohio. Several different dissolver vessels are used at this site, including a

Fig. 1. Metal dissolver.

Safe by ²³⁵U enrichment

Fig. 2. Continuous digestion.

Fig. 3. Drum digestion.

Fig. 4. Safe geometry.

batch metal dissolver, drum digesters, a safe geometry digestion system that can handle enrichments $\leq 10\%$ ^{235}U , and a continuous digestion system. Modifications in the processes, loading, and configurations of these systems have been made from time to time as production requirements have changed. These modifications have been made based on a variety of criteria including published reports, solid angle hand calculations, as well as computer simulation models using KENO and ANISN.

Several feed metal enrichments (0.86, 0.95, 1.25, and 2.1% ^{235}U) are processed routinely through the NLO digestion systems. To the casual observer, the differences in enrichment might seem to be of little concern, yet the amount of material that may be processed at the lowest enrichment can be several orders of magnitude more than that which may be processed at the highest enrichment. Thus, special consideration has to be taken of the process equipment involved, particularly since much of it was originally designed for natural uranium.

The four types of dissolver systems and methods for ensuring criticality will be discussed. A brief description of the system is outlined, and modifications made to ensure the nuclear safety of a specific campaign will be presented. The various secondary operations, i.e., UNH storage, transfer piping, etc., have not been included in this paper.

METAL DISSOLVER

Figure 1 shows the process flow through the metal dissolver. Uranium metal that is unsuitable as feed for direct remelting is dissolved in nitric acid. The dissolver is a stainless steel cylindrical tank with a raised grating near the bottom of the vessel. Metal is placed on the grating, and the dissolver is charged with nitric acid (HNO_3) until the uranium is immersed. To sustain an efficient process, a pump recirculates the acid from the bottom of the tank through a heat exchanger that cools it, and then circulates it back to the main tank through spray nozzles. In addition to the recirculated acid, fresh HNO_3 is introduced to maintain the required concentration (g U/l). The uranyl nitrate (UNH) product gradually attains the specified concentration and is drawn off to storage tanks until needed for further processings.

Specifications permit several alternative methods for

nuclear criticality safety control when charging enriched material. These all have to do with loading on the grating.

CONTINUOUS DIGESTION

The continuous digestion facility's process stream, shown in Fig. 2, begins when uranium concentrates and residues are fed continuously into a digester through which acidic uranyl nitrate is being introduced at a metered flow rate. The metered flow is recycled from one of the blend tanks through three digester tanks. Partial digestion occurs in the first digester tank and is continued through the system as the slurry cascades into the next vessel. Any digestion not completed in this process will be finished in the blend tank. Slurry recycling is continued until the dumping of uranium-bearing feeds has been completed. This process is safe by enrichment for enrichments $\leq 1.0\%$. This fact was verified by computer modeling performed at the Oak Ridge National Laboratory.

DRUM DIGESTION

The drum digesters are 55-gal stainless steel drums. They are used for digesting materials not suitable for the other digestion processes, i.e., metal spills, extrusion ends, and ash residues. Premeasured materials are placed in the drum, to which acid and water are added, and the charge is thoroughly mixed (Fig. 3). Heating coils in the vessels are used to facilitate the process. In the case shown, the resulting slurry is transferred to the center tank, where it is combined with several other drum digestion batches before being filtered and transferred to storage.

SAFE GEOMETRY DIGESTION

In 1970, a safe geometry digestion system was installed. It provides the capability for safely processing enriched uranium materials in enrichments $\leq 10\%$ ^{235}U at a rate of 100 kg/8-h shift. Employing safe cylinder diameter in all the process vessels (see Fig. 4), the system is inherently safe by design.

I. J. T. THOMAS, "Nuclear Safety Guide," TID-7016 (ORNL/NUREG/CSD-6), Rev. 2, Oak Ridge National Lab. (June 1978).

Mixed-Oxide Criticality Programs and Experiments

Plans and Equipment for Criticality Measurements on Plutonium-Uranium Nitrate Solutions, R. C. Lloyd, E. D. Clayton, B. M. Durst (PNL), invited

Data from critical experiments are required on the criticality of plutonium-uranium nitrate solutions to accurately establish criticality control limits for use in processing and handling of breeder-type fuels. Since the fuel must be processed both safely and economically, it is necessary that criticality considerations be based on accurate experimental data. Previous experiments have been reported on plutonium-uranium solutions with plutonium weight ratios extending up to some 38 wt% [plutonium weight ratio = $\left(\frac{\text{Pu}}{\text{Pu}+\text{U}}\right) 100\%$] (Refs. 1 through 4). No data have been presented, however, for plutonium-uranium nitrate solutions beyond this plutonium weight ratio. The current research emphasis is on the procurement of criticality data for plutonium-uranium mixtures up to 60 wt% plutonium that will serve as the basis for handling criticality problems subsequently encountered in the development of technology for the breeder community. Such data also will provide useful benchmarks for data testing and analysis on integral criticality experiments for verification of the analytical techniques used in support of criticality control.

Experiments are currently being performed with plutonium-uranium nitrate solutions in stainless steel cylindrical vessels and an expandable slab tank system. A schematic diagram of the experimental systems is shown in Fig. 1, including the Mix Dump (M.D.) tank.

These three separate vessel systems are tied together via the valve system to the solution addition system and into a common solution storage reservoir. This equipment design allows for a more rapid changeover than the remounting of separate vessels. The three vessel system

creates improved efficiency in experimental operation and makes it possible to perform measurements with identical solutions in each of the three vessels separately without reprocessing.

The small cylinder has an internal diameter of 35.39 cm and an internal height of 106.6 cm. The large cylinder has an internal diameter of 68.68 cm and a height of 106.6 cm. The wall thickness of the stainless steel cylinders is 0.079 cm; the bottom and top thicknesses are 0.95 cm.

The slab assembly shown in Fig. 2 for these experiments is of rather unique design in that its thickness can be adjusted over a range of 7.6 to 23 cm. This range of adjustments is made possible by stainless steel bellows fabricated around the periphery of the tank. The slab thickness changes can be accomplished by means of adjustment screws, located at the corners of the vessel, between the opposite sides of the slab; slab thickness is measured by means of a dial caliper. The height and width of the tank is 106.6 cm, based on the average of the bellows variation. The stainless steel sides (0.157 cm) are reinforced with an egg-crate-type structure that is effective in maintaining the side position to within ~ 0.025 cm on filling the tank. The assembly and its parts are of Type 304L stainless steel. A reflected assembly is achieved by attaching gasketed side plates and filling with water. The assembly is positioned in a large hood for contamination control.

The plutonium of the Pu+U in solution will range up to ~ 60 wt%. The plutonium contains 7.9 wt% ^{240}Pu . The uranium contains 0.724 wt% ^{235}U . During the course of the experiments, the Pu+U concentration in the nitrate solution will be varied throughout the range from ~ 100 to 500 g (Pu+U)/l. The critical height (and volume) of solution will be determined as a function of the Pu+U concentration. Criticality is being determined for several different conditions of neutron reflection. In the case of the 35.39-cm-diam cylinder, measurements are being made with a water reflector and also with a concrete shell weighing ~ 2 tons that is 25.4 cm thick. Experiments are also being performed with a reflector of 1.27 cm of steel backed by water.

Data will exist for mixtures up to 60 wt% plutonium in $(\text{U}+\text{Pu})\text{NO}_3$ for a variety of shapes and reflector conditions. The experiments also show the feasibility and flexibility of a multiple tank setup that reduces the time between experiments of different geometric design, and permits operations with a minimum quantity of nuclear materials and fuel reprocessing handling. The use of this system decreases handling of the fissile material and changeout of experimental vessels, which reduces possibility of contamination and keeps the personnel radiation as-low-as-reasonably-achievable (ALARA). On completion of the planned experiments, more than 50 data points will be available on the criticality of Pu-U nitrate solutions in the three vessels described.

Fig. 1. Schematic diagram of experimental setup.

1. R. C. LLOYD, S. R. BIERMAN, and E. D. CLAYTON, "Criticality of Plutonium-Uranium Mixtures Containing 5-8 wt% Pu," *Nucl. Sci. Eng.*, 55, 51 (1974).
2. R. C. LLOYD and E. D. CLAYTON, "Criticality Safety Data Applicable to LMFBF Fuel Processing," *Nucl. Sci. Eng.*, 59, 21 (1976).

Fig. 2. Critical experiments adjustable slab assembly.

3. R. C. LLOYD and E. D. CLAYTON, "Criticality of Plutonium-Uranium Nitrate Solutions," *Nucl. Sci. Eng.*, **60**, 143 (1976).
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Nuclear Criticality Safety Experiments at JAERI, I. Kobayashi, S. Katsuragi (JAERI-Japan), invited

Criticality safety data have been accumulated from around the world, with the help of such unclassified reports as the three-volume publication¹ by Koponen et al., which contains indexes, concordances, and abstracts of reports published up to 1978. We have analyzed the publication and found, as shown in Fig. 1, that the amount of experimental data for criticality safety is less than 50% of the whole criticality data base and that 13, 16, and 28% of the data are on mixtures of plutonium and uranium, solutions of plutonium, and solutions of uranium, respectively. In those data for mixtures and solutions, we could not find data sufficient for reasonable and effective design and operation of out-of-reactor nuclear facilities for light water reactor (LWR) and fast breeder reactor (FBR) fuel cycles.

The criticality data that should be obtained experimentally are for low-enriched uranium, plutonium, and plutonium-uranium mixtures² having chemical and physical forms found in nuclear fuel reprocessing and fabrication facilities. Criticality data of dry and damp powders of

mixed oxides of (0 to 60 wt%)PuO₂-(depleted and natural)UO₂ and nitrate solution of the mixture are especially useful and valuable, because safeguards in Japan require that the products of reprocessing facilities be stored in the form of 50 wt%PuO₂-UO₂.

To supply the data above, Japan Atomic Energy Research Institute has a project to construct an experimental facility consisting of three experimental cells. The following fuel materials will be handled by using split-table, vertical lift, and tank-type critical assemblies:

1. rod-type fuels (UO₂ and PuO₂-UO₂) such as that of LWRs, ATRs (Advanced Thermal Reactors such as the CANDU), and FBRs
2. solutions such as UO₂(NO₃)₂ and Pu(NO₃)₄
3. powder or particles such as UO₂ and PuO₂-UO₂.

TABLE I
Construction Schedule of the Facility

	1981	1982	1983	1984	1985	1986	1987
Pre-Conceptual Design							
Conceptual Design							
Detail Design							
Licensing							
Safety Analysis Report							
Construction							
Experiment							→

Percentage of Metal & Solution Data
In Criticality Safety

Published nuclear criticality reports up to 1978

Fig. 1. Published nuclear criticality reports up to 1978 (Ref. 1).

In the criticality experiments, transient behavior would be measured for solution fuels on mockups simulating criticality accidents due to equipment failure, trouble, misoperation, and so on. Adjacent to the criticality facility, a fuel processing facility of small capacity would be constructed to supply fuels to the critical assemblies. The processing facility would have the function of dissolution of the oxides, separation, purification, and conversion of nitrates to oxides of plutonium, uranium, and their mixture, but will not have processes for canning the oxide fuels. The time schedule of the project is shown in Table I. The experiment would start in 1987. A great deal of effort will go into the licensing of the experimental facilities and assuring public acceptance for carrying out the experiment without trouble.

I. B. L. KOPONEN, T. P. WILCOX, and V. E. HAMPEL,
"Nuclear Criticality Experiments from 1943 to 1978,

An Annotated Bibliography," Vols. 1, 2, and 3, UCRL-52769, Lawrence Livermore National Lab. (1979).

2. R. KIYOSE, I. KOBAYASHI, and Y. NOMURA, "Experimental Benchmark Needs for Criticality Safety," *Trans. Am. Nucl. Soc.*, **39**, 513 (1981).

Definition of Experimental Procedures to Assure Quality in a Program of Mixed-Oxide Critical Experiments, R. E. Malenfant, H. M. Forehand, Jr., A. E. Evans, Jr. (LANL), invited

INTRODUCTION

Current designs for reprocessing facilities require storage of considerable quantities of mixed uranium and plutonium oxides for eventual shipment to a fuel fabrication facility.

The water content, due to absorption of room moisture, incomplete drying, etc., is expected to be low, <3 wt% water. Current cross-section data and codes overestimate the neutron multiplication factor for thermalized mixed-oxide systems and underestimate k_{eff} as the neutron energy spectra harden and approach spectra typical for these systems.

A program to validate calculations has been proposed. The critical experiments involve up to 350 kg of material. It is desirable to evaluate a range of water content, generally expressed as H/Pu ratios, for several Pu/U ratios as bare and reflected configurations. There are several complications that prevent the experiments from being as simple in execution as in concept:

1. The mixed oxides cannot be handled as loose bulk material because the water content and density could not be assured.

2. Safety considerations restrict milling and blending operations to 10-kg batches. Preparation and certification of the quality and homogeneity of the 350-kg batch is therefore difficult and time consuming.

3. Restrictions on physical handling of the material because of the high specific activity and the complications listed above suggest complete encapsulation. However, provision must still be made to easily accommodate changes in the physical parameters.

4. The mixed oxide had a specific activity of ~6 W/kg. This required provision to cool or vent encapsulated material to prevent catastrophic pressure buildup.

We proposed and evaluated a technique to implement this program and to assure the quality of the components. We built a doubly encapsulated canister of aluminum to hold pressings, called pucks, of blended PuO_2 and depleted UO_2 with a water binder. Twelve pucks, 6.5-cm-square X 3-cm-thick and each weighing ~750 g, were pressed to a density of ~6 g/cm³ and assembled into a double-walled welded aluminum container to form a test assembly 6.5 cm square X 36 cm high.

INSPECTION TECHNIQUES

The internal pressure and temperature of the assembly were monitored for several weeks under various external conditions to provide assurance that the integrity of the system would be maintained during the experimental period. At one time, the assembly inadvertently vented. Although this resulted in an uncertainty in the water content, there was no evidence of radioactivity in the escaping gas. The container was modified to prevent a recurrence.

CANISTER EVALUATION

Several techniques were evaluated to monitor the characteristics of the assembly:

1. x-ray radiography
2. neutron radiography
3. activation of wires attached to the outside of the assembly, followed by counter scanning of the wires
4. gross reactivity effects by a bulk replacement measurement.

Two types of x-ray radiographs were made. For the first, a ^{60}Co unit was used to produce a radiograph that clearly showed the fuel inside its container, but gave no details of the individual pucks. Subsequently, the assembly was exposed to 30-MeV radiation from a betatron. These high-energy radiographs displayed the topography of the interfacial region of the pucks and showed that the faces

of some had started to crumble. The degree of crumbling should correlate with the moisture content, because there was no binder other than water.

To study the moisture distribution in the canister, we made 14-MeV neutron radiographs with a Cockcroft-Walton accelerator. Using the Berger¹ transfer method, neutrons transmitted through the assembly activated a copper sheet through the $^{63}Cu(n,2n)^{62}Cu$ reaction. The reaction has a threshold of 11 MeV, so most of the scattered neutrons will not contribute to the activity. After a 30-min irradiation (^{62}Cu has a half-life of 9.9 min), the copper plate was loaded into a film holder for a 30-min exposure of a sheet of Kodak-type XR-5 high-speed x-ray film.

The neutron fluence that can be given to an object using this technique is limited by the output of the accelerator and also by the activation half-life of the copper. To provide sufficient neutrons for imaging, it was necessary to place the object 30 cm from the neutron source with the copper sheet at 40 cm. As a result, the radiograph suffered from beam divergence at distances greater than ~5 cm from the centerline of the beam. Nevertheless, we have established our ability to make useful fast neutron images of such objects as dense fuel elements. In this case, the similarity of the betatron x-ray image and the neutron image suggests that the moisture distribution is fairly uniform throughout the assembly. That is, the water in the assembly appeared to have the same distribution as the material.

WIRE ACTIVATION

Aluminum wires loaded with uranium, each ~76 cm long, were irradiated while in contact with the UO_2 - PuO_2 test assembly at the center of the PARKA critical assembly machine.² A wire was attached along each of the four vertical faces of the test assembly. These wires extended ~5.7 cm below the unit. A fifth wire was inserted into the PARKA core at a radius of 15 cm with the top of the wire aligned with the top of the PARKA fuel. The distribution of fission product activity in the wires was then determined with a scanning wire-counter system. Scanning data were binned in 1.4-cm-long segments, with 29 bins (or channels) in contact with the test assembly. Data were corrected for fission product decay. The corrected data were least-squares fitted to the function

$$y = P_1 \cos \{[\pi(x - P_2)/P_3]\}$$

where

y = corrected responses

x = channel number

P_1 , P_2 , and P_3 = fit parameters.

If the water and fuel were uniform within the assembly, we expected that the distribution would match the unperturbed axial fission profile in PARKA. Any deviation from uniformity should have been reflected in differences in the wire activation. Although not definitive by itself, we hoped that this diagnostic, along with others, would add confidence to the evaluation of the homogeneity of the assembly.

GROSS REACTIVITY MEASUREMENT

To provide additional information on the water content, we measured the reactivity worth of the assembly in the PARKA central test hole. To make the measurement, the

²PARKA is a 0.9-m-diam X 1.3-m-long cylindrical UC reactor with a beryllium reflector. A longitudinal test hole at the center of the core was provided to accommodate test specimens.

test assembly was attached to a remotely operated precision cathetometer for insertion into the center of the core. The worth of the assembly as a function of its position in the core was determined by the calibrated control element technique. The measured worth of the test assembly centered in the PARKA core was 2.0 dollars. Subsequently, 281 g of water was removed from the assembly by pumping on its vent tube. Remeasurement of the reactivity indicated a loss in worth of 0.42 dollar because of the removal of the water.

A single measurement of reactivity is of limited utility. However, by comparing various assemblies in the experiment, and by evaluating differences introduced as a result of changing moisture content, we hoped to increase confidence in the state and quality of the individual assemblies.

CONCLUSION

We believe that the techniques identified in this paper have not shown any indication that water is heterogeneously distributed in the mixed-oxide pressings.

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Nuclear Criticality of Heterogeneous Mixed-Oxide Systems, R. A. Libby, E. D. Clayton (PNL), invited

An existing American National Standard, ANSI/ANS-8.12-1978, provides guidance for nuclear criticality control and safety of homogeneous plutonium-uranium fuel mixtures outside reactors.¹ Data were included in that standard for PuO₂-UO₂ mixtures made up with plutonium of different isotopic composition. Critical dimensions for heterogeneous systems, however, will differ significantly from those of homogeneous systems, and the effect of higher isotopes of plutonium also will differ due to self-shielding effects in the lattice (heterogeneous) systems. Therefore, data are required for criticality control of heterogeneous systems of mixed-oxide fuels, and Work Group ANS-8.12.1 of Subcommittee 8 of the American Nuclear Society Standards Committee was formed to provide these data. In connection with these activities, some data were previously reported on critical dimensions for arrays of (U,Pu)O₂ rods in water.² Since that paper was presented, calculations have been extended to include the effects of higher isotopes of plutonium, i.e., ²⁴⁰Pu and ²⁴¹Pu, on criticality of heterogeneous mixed-oxide systems, and a number of additional lattice benchmarks have been calculated as validation of the methods used.

The results of these calculations presented and discussed herein complement the data on homogeneous systems already available.^{1,3} Calculations have been made with three different compositions of mixed oxides—3 wt% Pu, 15 wt% Pu, and 30 wt% Pu in (U,Pu)O₂—for three different isotopic compositions of plutonium in the mixed oxide: (a) 100 wt% ²³⁹Pu, (b) 79 wt% ²³⁹Pu, 15 wt% ²⁴⁰Pu, 6 wt% ²⁴¹Pu, and (c) 60 wt% ²³⁹Pu, 25 wt% ²⁴⁰Pu, 15 wt% ²⁴¹Pu.

The EGGNIT-II code⁴ was used to generate 17-group macroscopic cross sections in the epithermal neutron energies for a cell of fuel rod material and water. The cross sections for the single thermal group were calculated either with the BRT-I (THERMOS) code⁵ or using the TEMPEST portion of EGGNIT. These codes used cross-section data from the ENDF/B-III and ENDF/B-IV libraries, which were processed by FLANGE (Ref. 6)-ETOG (Ref. 7). The macroscopic cross sections were used in HFN (Ref. 8), a one-dimensional diffusion theory code. Each HFN calculation was run as a

critical search on the major dimension: the slab width, cylinder diameter, or spherical radius. HFN calculations were made to determine the minimum critical dimension as a function of rod size and pitch. The results of the calculation for minimum critical dimension are summarized in Table I, and for minimum critical mass in Table II. It should be noted that for the case of plutonium (25,15) with 34 wt% PuO₂, the smallest critical dimension is obtained with unmoderated material (see footnote d of Table I). Calculations are provided for $k_{eff} = 1$ and also $k_{eff} = 0.95$ to explore the sensitivity of the dimensions and masses to changes in k_{eff} .

In support of validation of the calculations, some 42 PuO₂-UO₂ lattice benchmarks have been calculated using HFN with ENDF/B-IV cross sections processed by EGGNIT. The results are summarized below:

PuO ₂ -UO ₂ Enrichment wt% PuO ₂	Number of Lattices	Average k_{eff} HFN
1.5	6 (Ref. 9)	1.016 ± 0.012
2.0	17 (Ref. 9)	1.026 ± 0.007
3.98	7 (Ref. 9)	1.045 ± 0.014
6.6	5 (Ref. 10)	1.035 ± 0.022
25.2	7 (Ref. 11)	1.051 ± 0.015

The above validation calculations are being used in part to assess allowances that must be made to account for uncertainties to ensure that calculated subcritical limits are indeed subcritical. These calculations indicate the calculated minimum critical dimensions and masses at k_{eff} of 0.95 are likely to serve as the basis for these limits. Further calculations are being made, however, and the decision will be made by Work Group ANS-8.12.1 as to the recommended subcritical limits in each case for further review by Subcommittee ANS-8 of the American Nuclear Society Standards Committee.

The calculations for heterogeneous systems show significantly smaller minimum critical dimensions than for homogeneous systems throughout the plutonium concentration range studied. The effect of higher isotopes of plutonium also differs substantially from the effects for homogeneous systems.

Comparisons between critical dimensions and masses for heterogeneous systems and homogeneous systems will be discussed and comments provided on the uncertainties associated with these calculations.

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TABLE I
Calculated Minimum Critical Dimensions for Heterogeneous Mixed-Oxide Systems

	Pu(100) ^a O ₂ + UO ₂							
	3.40 wt% PuO ₂ (3 wt% Pu) ²		9.07 wt% PuO ₂ (8 wt% Pu) ²		17.01 wt% PuO ₂ (15 wt% Pu) ²		34.01 wt% PuO ₂ (30 wt% Pu) ²	
	k _{eff}		k _{eff}		k _{eff}		k _{eff}	
	0.95	1.0	0.95	1.0	0.95	1.0	0.95	1.0
Cylinder Dia. (cm)	20.1	21.9	17.1	18.7	16.0	17.4	15.0	16.3
W/F Volume Ratio		4.2		6.5		9		10
Pin Dia. (cm)		0.4		0.3		0.2		0.15
Sphere Dia. (cm)	29.8	32.5	26.1	28.1	24.6	26.4	23.3	25.0
W/F Volume Ratio		4.3		7		9.5		11
Pin Dia. (cm)		0.4		0.3		0.2		0.15
Sphere Volume (L)	13.9	18.0	9.3	11.6	7.8	9.5	6.6	8.2
Slab Thickness (cm)	8.9	10.1	6.9	7.9	6.2	7.1	5.5	6.4
W/F Volume Ratio		3.7		5.5		7.5		8
Pin Dia. (cm)		9.4		0.3		0.2		0.15
	Pu(15,6) ^b O ₂ + UO ₂							
	3.40 wt% PuO ₂ (3 wt% Pu) ²		9.07 wt% PuO ₂ (8 wt% Pu) ²		17.01 wt% PuO ₂ (15 wt% Pu) ²		34.01 wt% PuO ₂ (30 wt% Pu) ²	
	k _{eff}		k _{eff}		k _{eff}		k _{eff}	
	0.95	1.0	0.95	1.0	0.95	1.0	0.95	1.0
Cylinder Dia. (cm)	23.1	24.6	19.7	21.6	18.3	20.1	17.3	18.9
W/F Volume Ratio		4.0		7		10		12
Pin Dia. (cm)		0.7		0.4		0.3		0.25
Sphere Dia. (cm)	34	37.3	29.4	32.0	27.7	30.0	26.3	28.4
W/F Volume Ratio		4.0		7.5		10.5		13
Pin Dia. (cm)		0.7		0.4		0.3		0.25
Sphere Volume (L)	20.6	27.2	13.3	17.1	11.1	14.1	9.5	12.0
Slab Thickness (cm)	10.8	12.4	8.5	9.8	7.7	8.8	7.0	8.0
W/F Volume Ratio		4.0		6.5		9		11
Pin Dia. (cm)		0.7		0.4		0.3		0.25
	Pu(25,15) ^c O ₂ + UO ₂							
	3.40 wt% PuO ₂ (3 wt% Pu) ²		9.07 wt% PuO ₂ (8 wt% Pu) ²		17.01 wt% PuO ₂ (15 wt% Pu) ²		Unmoderated ^d 34.01 wt% PuO ₂ (30 wt% Pu) ²	
	k _{eff}		k _{eff}		k _{eff}		k _{eff}	
	0.95	1.0	0.95	1.0	0.95	1.0	0.95	1.0
Cylinder Dia. (cm)	24.7	27.4	20.7	22.8	19.2	21.2	17.2	18.8
W/F Volume Ratio		4.2		7		9		0
Pin Dia. (cm)		0.75		0.5		0.4		-
Sphere Dia. (cm)	35.9	39.6	30.9	33.6	28.8	31.4	26.5	28.5
W/F Volume Ratio		4.2		6.5		8		0
Pin Dia. (cm)		0.75		0.5		0.4		-
Sphere Volume (L)	24.2	32.5	15.4	19.9	12.5	16.2	9.4	12.1
Slab Thickness (cm)	11.8	13.6	9.1	10.5	8.2	9.5	6.7	7.75
W/F Volume Ratio		4		6		8		0
Pin Dia. (cm)		0.75		0.5		0.4		-

^aPu: 100 wt% ²³⁹Pu.^bPu: 79 wt% ²³⁹Pu, 15 wt% ²⁴⁰Pu, 6 wt% ²⁴¹Pu.^cPu: 60 wt% ²³⁹Pu, 25 wt% ²⁴⁰Pu, 15 wt% ²⁴¹Pu.^dCritical dimensions for unmoderated systems at theoretical density of oxides; values given for 3 to 15 wt% Pu are for moderated systems that have been shown to have the minimum critical dimensions for those Pu contents (Ref. 3).

TABLE II
Calculated Minimum Critical Masses for Heterogeneous Mixed-Oxide Systems

	Pu(100) ^a O ₂ + UO ₂							
	3.40 wt% PuO ₂ (3 wt% Pu) ²		9.07 wt% PuO ₂ (8 wt% Pu) ²		17.01 wt% PuO ₂ (15 wt% Pu) ²		34.01 wt% PuO ₂ (30 wt% Pu) ²	
	k _{eff}		k _{eff}		k _{eff}		k _{eff}	
	0.95	1.0	0.95	1.0	0.95	1.0	0.95	1.0
Mass ²³⁹ Pu (kg)	0.54	0.71	0.47	0.60	0.44	0.56	(0.41) ^b	(.52)
Mass Mixed Oxide (kg)	18	23.6	5.9	7.5	2.9	3.7	(1.37)	(1.73)
W/F Volume Ratio		12		28		55		
Pin Dia. (cm)		0.18		0.07		0.03		~0.0
	Pu(15,6) ^c O ₂ + UO ₂							
	0.65		0.56		0.52		0.49	
	0.82		0.70		0.66		0.62	
	27.4		8.8		4.4		2.1	
Mass ²³⁹ Pu (kg)	0.65	0.89	0.56	0.73	0.52	0.68	0.49	0.64
Mass of Pu (kg)	0.82	1.1	0.70	0.93	0.66	0.85	0.62	0.80
Mass Mixed Oxide (kg)	27.4	37.5	8.8	11.5	4.4	5.7	2.1	2.7
W/F Volume Ratio		11		25		50		95
Pin Dia. (cm)		0.25		0.11		0.06		0.04
	Pu(25,15) ^d O ₂ + UO ₂							
	0.64		0.53		0.49		0.46	
	0.11		0.88		0.82		0.77	
	35.5		11.0		5.44		2.55	
Mass ²³⁹ Pu (kg)	0.64	0.87	0.53	0.71	0.49	0.66	0.46	0.61
Mass of Pu (kg)	0.11	1.45	0.88	1.18	0.82	1.09	0.77	1.01
Mass Mixed Oxide (kg)	35.5	48.3	11.0	14.8	5.44	9.28	2.55	3.39
W/F Volume Ratio		9		23		45		85
Pin Dia. (cm)		0.40		0.12		0.07		0.05

^aPu: 100 wt% ²³⁹Pu.

^bValues in parentheses estimated.

^cPu: 79 wt% ²³⁹Pu, 15 wt% ²⁴⁰Pu, 6 wt% ²⁴¹Pu.

^dPu: 60 wt% ²³⁹Pu, 25 wt% ²⁴⁰Pu, 15 wt% ²⁴¹Pu.

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Critical Experiments with Arrays of Three-Litre Bottles Filled with Pu(2.8)(NO₃)₄, B. M. Durst, E. D. Clayton, J. H. Smith (PNL)

The safe storage of nuclear fuel is a prime concern of the nuclear fuel cycle worldwide. In the reprocessing of nuclear fuels, such as in the Purex Facility at Hanford, plutonium nitrate solutions are commonly stored in stainless steel or polyethylene cylindrical bottles, and yet to date, an experiment has never been performed determining the

effect on criticality of spacing or moderation between bottles.

To address this problem, Rockwell Hanford Operations recently initiated a series of experiments at the Battelle Critical Mass Laboratory with arrays of bottles containing plutonium nitrate solution. The experiments were performed with 3-l polyethylene bottles containing plutonium with a ²⁴⁰Pu content of 2.8 wt%. The objective of these experiments was to provide benchmark data to check calculational codes used in the criticality safety program at the Purex Facility. The experiments addressed a number of factors affecting criticality; namely, (a) the critical air gap between bottles in an array of a fixed number of bottles, (b) the number of bottles required for criticality if the bottles are touching, and (c) the effect on critical array spacing and critical bottle number due to the insertion of a hydrogenous substance into the air gap between bottles.

The experiments were performed with sixteen 3-l polyethylene bottles filled with Pu(NO₃)₄. The fissile solution had an average concentration of ~117 g/l plutonium, with the ²⁴⁰Pu content being 2.8 wt% at a free acid molarity (H⁺) of ~6.7. The solution specific gravity was 1.421. The bottles were Plexiglas reflected on all sides and bottom, as shown in Fig. 1. The top was left unreflected. Bottles were positioned with the help of an aluminum framework that could be moved accurately in both the x and y directions. Four tiny aluminum pins securely held each bottle in place to prevent tipping and bottle movement.

Criticality was approached by loading about half of the critical array on each side of a split table device. The fuel loading adjustment was made by hand, and, after each subsequent change to the system, the table was closed

Fig. 1. Isometric view of interacting 3-l bottles.

Fig. 2. The effect of Plexiglas shells on critical number of 3-l bottles.

remotely, completing the array geometry desired. The critical approach method was used to measure the desired critical spacing or critical bottle number, depending on the experiment.

After the initial series of experiments was performed with bottles separated by air gaps, Plexiglas shells of varying thicknesses were placed around each bottle to investigate how moderation between bottles affects both the number of bottles required for criticality and the critical spacing between each bottle.

The experiment results are graphically summarized in Fig. 2. The minimum number of bottles required for criticality was found to be 10.9 bottles, occurring for a square array where bottles were touching one another. As the bottles are spaced apart, more bottles were needed to remain critical. For 16 bottles in a square array, the critical separation between bottle surfaces in both x and y directions is 0.96 cm. The addition of Plexiglas around each bottle decreased the critical bottle number, compared to those separated only by air, but the critical bottle number, even with interstitial plastic in place, was always >10.9 bottles. The most reactive configuration was a tightly packed array of bottles with no intervening material.

As a result of these experiments, calculational benchmarks now exist for criticality analysis of bottle arrays of plutonium nitrate solutions. The experiments illustrate that subcritical mass in an array can be substantially increased if bottles are sufficiently spaced apart. However, if spacing is to be used to ensure criticality safety, the effects of any intervening hydrogenous material must be carefully considered, since this will reduce the margin of safety gained by spacing.

Criticality Accident Experimental Studies with High-Enriched Uranium Solutions, F. Barbry, Jean C. Manaranche (CEA Valduc-France)

The Nuclear Protection and Safety Institute of the Commissariat à l'Energie Atomique (CEA) is conducting a new important experimental program at the Valduc Criticality Laboratory in the field of criticality accident studies, both to provide consistent data for safety analysis and to improve its ability for facing a possible accidental situation.

Fig. 1. Boiling-type experiment: power versus time.

For this purpose, many experiments were performed several years ago in the CRAC facility.¹ The new experimental program² is now being carried out in the pulsed reactor SILENE, and its aim is to obtain better knowledge regarding the phenomenology of power excursion in fissile solutions and the various associated effects. SILENE is a small solution reactor³ using as fuel a solution of high-enriched uranium nitrate. The values of the solution concentration are 71 g/l in a bare configuration and 61 g/l when the core is surrounded by a lead shield.

Many important results have been obtained recently concerning

1. the phenomenology of "boiling"-type experiments
2. the rate of fission gas release during the excursion
3. the pressure wave associated with the first peak of the excursion and the radiolytic gas formation.

They can be summarized as follows:

1. The "boiling"-type experiments (Fig. 1), which were carried out with a solution at the concentration of 71 g/l, have shown that:

- a. boiling is reached for a minimum fission yield of 1.1×10^{16} fissions per litre
- b. the average power level of the "boiling plateau" depends on the inserted potential reactivity and can reach 0.2 kW per litre of solution for a 7-dollar reactivity. It corresponds to a rate of vapor generated of 10% in volume.

Fig. 2. Pressure wave versus maximum fission rate.

2. As to the release of fission gases during severe excursions, it appears that the release of rare gases (xenon, krypton) is high (up to 100%), but that the values for iodine depend on the type of excursion. In any case, these values were lower than 1% during a boiling-type experiment after transfer through a several-metres-long stainless steel pipe.

3. Regarding the dynamic pressure wave associated with the first peak power, several relations have been set up among the pressure Δp , the reciprocal period, and the maximum energy and power per unit of volume $\dot{E}/V$ and $\dot{E}/V$. These relations correctly fit the experimental results. One of them is, for instance (see Fig. 2),

$$\Delta p = 3.9 \times 10^{-15} \left(\frac{\dot{P}}{V} - 6 \times 10^{13} \right),$$

where $\dot{P}/V$ is in fissions $\cdot \text{cm}^{-3} \cdot \text{s}^{-1}$. Moreover, experiments undertaken with capsules designed for radiolytic gas measurements have permitted us to determine a volume of gas ($\text{H}_2 + \frac{1}{2} \text{O}_2$) corresponding to 3.5 l per Megajoule (3.3×10^{16} fissions).

Complementary boiling-type experiments will be per-

formed in SILENE with a solution at a concentration of $220 \text{ g} \cdot \text{l}^{-1}$ to confirm some of these results and to try to get a relation between the power level of the "plateau" and the excess of reactivity ΔK . Besides, the emphasis will be put on the iodine measurements and analysis (forms, activities) at the nearest location from the core, to avoid the iodine deposition on the pipes or stainless steel walls during the transfer. Moreover, intervention tests are planned to improve the ability to face a possible accidental situation.

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Measurement and Calculation of Fast Reactor Parameters

Critical Review: Fast Reactor Benchmarks,
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(ANL), S. G. Carpenter (ANL-Idaho)

Critical experiments are used both for nuclear data testing and for development of calculational models and bias factors for power reactor designs. For data testing, the cores should span a range of material compositions and spectra and have simple geometry and heterogeneity so as to minimize modeling uncertainties. Power reactor mockup cores need to have realistic compositions and geometry to test predictions of the physics parameters required in design. These cores fall into two classes: benchmark cores that reproduce the principal features in a simple design, and the more complex engineering mockup cores. Over the past few years, a number of improvements and additions have been made to the benchmark data at Argonne National Laboratory (ANL). Three areas are discussed: (a) revisions of experimental data and specification of uncertainties and correlations, (b) improvements in calculational modeling, and (c) addition of new cores of interest because of composition, size, or geometry.

Most of the fast reactor benchmarks in the CSEWG compilation¹ date back 10 to 20 years or more. Many of these integral measurements remain of value today, particularly those in the homogeneous cores built at Los Alamos. Data for cores with liquid-metal fast breeder reactor (LMFBR) compositions and spectra are obtained in the plate-type criticals. Some of these data were obtained by methods now considered inadequate and have lost their usefulness. For example, measurements of ^{238}U fission in a counter positioned in a cavity are difficult to correct for environmental effects and often give discrepant results. Measured ^{238}U capture rates are of prime importance, and the foil techniques remain the

subject of detailed investigation.² A large number of material reactivity worths are often given in the benchmark specifications. These were obtained using samples oscillated in a tube through the reactor and are often sensitive to environmental and streaming effects. While the worths are commonly calculated, no serious use has been made of the data. It is anticipated that the current investigations in the ANL diagnostic cores program³ will provide evidence for revision or rejection for much of these data.

The diagnostic core experiments have been accompanied by detailed analytical investigations of reactivity worths. It has been shown that a proper treatment of adjoint flux variations in the cells is very important in the calculations.⁴ Fuel worths are reduced by typically 5% in the heterogeneous cells. These corrections indicate that significant improvements can be made to the benchmark homogeneous values.

Uncertainty information provided with the benchmark data is often inadequate. In addition, recent sensitivity studies have highlighted the need for covariance data between the integral parameters. Detailed studies of uncertainties in experimental critical masses and reaction rate ratios have been made for several of the more recent benchmarks. The experimental data were reviewed and the individual sources of uncertainty reassessed. From these data, the covariance matrix for a set of measurements has been developed. Appropriate weighting for several measurements of the same quantity, in some cases, has led to significant revisions of the benchmark parameters. These revised data and covariances have been used in development of the ORACLE adjusted library for LMFBRs (Ref. 5).

For convenience, benchmark models are much simplified versions of the assemblies, frequently homogeneous spherical or cylindrical representations. Correction factors from the as-built assemblies to the idealized versions are provided, but

TABLE I
 Recent ZPR Cores Suggested for Benchmark Consideration

Core	Special Characteristics
ZPR-9/29 (GCFR)	Reference core for a GCFR composition
ZPR-9/34 (U-Fe)	Limited materials, highly enriched uranium and iron
ZPR-9/36 (U-9)	Limited materials, enriched uranium (small amount of steel)
ZPR-9/32 (RSR)	Series of highly distorted cores for investigation of LMFBR accident configurations
ZPPR-7A	Heterogeneous core, no control positions, central blanket and three internal blanket rings
ZPPR-9	Large homogeneous core, no control positions

the details of these procedures are not always clearly defined. The present specifications do not include important corrections to k_{eff} for neutron streaming that are now recognized. Present-day computer codes enable more precise calculation of cell heterogeneity and more detailed reactor calculations using three-dimensional diffusion theory and two-dimensional transport theory. Thus, more precise modeling corrections can be generated, although estimation of their uncertainties (and correlations) may remain little more than inspired guesswork. In addition, increasing use of the VIM Monte Carlo code for whole reactor calculations has been valuable in testing the deterministic calculations.⁶ This has led to several significant advances, but some discrepancies remain and the work is continuing.

Several recent Zero-Power Reactor (ZPR) cores that are suggested for future benchmarks are listed in Table I. These cores have been the subject of detailed experimental and analytical investigations. The last two represent several essential features of a 350-MW(e) heterogeneous core and a 700-MW(e) homogeneous core. In addition to their usefulness for testing design-level methods and codes, the latter two cores have indicated problems in prediction of spatially varying parameters that are related to cross-section data.

Thus, the fast reactor benchmarks require attention to update the current specifications to account for advances in both experimental and calculational methodology. In addition, it may be appropriate to add new data from more complex critical experiments. Much of the data necessary to revise the specifications will be available on completion of the physics assessment program,³ in which the critical experiment data base is being comprehensively reviewed.

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Use of ZEBRA-8 Analysis for Validating the Critical Experiment Cross-Section Processing Technique, P. T. Choong, A. K. Hartman (GE, Sunnyvale)

One of the major objectives of the critical experiment program is to provide analytical support to licensing efforts through confirmation of bias factors and design margins in liquid-metal fast breeder reactor (LMFBR) design methods. The use of critical experiment information to help set those bias factors and design margins is influenced by the heterogeneity and neutron streaming differences between the critical experiment and the simulated LMFBR. Traditionally, the emphasis for a critical experiment program had been placed on near-exact simulation of intended LMFBR design. As a result, all critical experiment analysis at the Zero-Power Plutonium Reactor (ZPPR) was centered around core compositions and spectra very similar to LMFBR designs. The results were useful for setting bias factor and design margin. Because of the very narrow variation of composition and spectra, ZPPR data have limited utilities for testing cross-section libraries and processing techniques.

The ZEBRA-8 series^{1,2} of null-zone measurements featured a different neutron spectrum for each assembly. The series covers a range of spectra both harder and softer than that of the LMFBR and also emphasizes different structural/diluent materials. The experimental data base was ideal for validating the cross-section processing technique because the experimental results had been reduced to simulating the infinite medium, and uncertainties in the extrapolation have been included. The homogeneous compositions for each assembly are shown in Table I.

Heterogeneous calculations for each test zone were done with TDOWN using the LASL-MINX-generated LIB-IV cross-section library in 50 energy groups. A one-dimensional cell model was used to treat resonance shielding for all isotopes. The rigorous integral transport method (the Slicing Technique³) was employed to generate fine structure flux for cross-section homogenization. The spectrum of the homogenized cell was iteratively adjusted until the elastic removal cross section was converged to self-consistency. The adequacy of TDOWN's "f-factor" scheme using 171-group VITAMINE-C was compared with resonance integral scheme of 156-group MC² calculations. The results in Table II showed that except for very soft spectrum cases (8A and 8F), the difference is ~0.45%. Probably one-half of the difference is attributable to the fact that structural and diluent isotopes were not resonance-shielded in MC² results. In the same table, the adequacy of TDOWN's elastic removal scheme is validated. Typical uncertainty attributable to the elastic removal model is on the order of 0.1% in k_{∞} . The heterogeneity effects also included in Table II in general are larger than in ZPPR cells. Extrapolating from the excellent k_{∞} agreements with the Argonne National Laboratory Monte Carlo (VIM) results⁴ shown in Table III, it can be inferred that TDOWN's heterogeneity cell modeling techniques are in very good shape.

The relatively large spread of calculation-to-experiment (C/E) results for ZEBRA-8 k_{∞} ($\pm 1.3\%$) is indicative of the need for a better cross-section library. In particular, the low value of 0.958 for ZEBRA-8C suggested that iron resonance capture was too high. For the important 28C/49F ratio, the C/E results are in excellent agreement with VIM's. However, they are several percent lower than normally observed in ZPR and ZPPR analysis.⁵

The diversity of material composition and resulting spectra coupled with the simplicity of analysis geometry made ZEBRA-8 very useful for rapid validation of cross-section library and processing techniques. It is highly recommended

TABLE I
Homogeneous Atomic Densities for the ZEBRA-8 Test Regions (10^{22} atom/cm³)^a

	8A	8B	8C	8D	8E	8F	8H
²³⁹ Pu	0.07616	0.1302	0.07543	0.08361	0.07112	0.07000	-
²⁴⁰ Pu	0.00394	0.00674	0.00397	0.00433	0.00368	0.00782	-
²⁴¹ Pu	0.00023	0.00038	0.00022	0.00023	0.00021	0.00084	-
²⁴² Pu	-	-	-	-	-	0.00007	-
²³⁵ U	0.00973	0.02079	0.00723	0.01202	0.01022	0.00633	0.2439
²³⁸ U	1.342	2.868	0.9970	1.658	1.410	0.8734	3.797
C	4.402	1.534	0.6367	1.307	0.0215	2.827	0.0433
Na	0.0002	0.0001	-	0.6075	1.033	0.0001	-
Fe	0.3627	0.4054	3.541	0.6298	0.8033	0.7320	0.3123
Cr	0.1016	0.1123	0.9924	0.1755	0.2238	0.2071	0.0864
NI	0.0551	0.0600	0.4778	0.0908	0.1143	0.1075	0.0483
Mn	0.0072	0.0077	0.0584	0.0124	0.016	0.0160	0.0064
Mo	0.0008	0.0008	0.0061	0.0008	0.0008	0.0008	0.0008
Ti	0.0016	0.0016	0.0174	0.0016	0.0016	0.0016	0.0016
Mb	-	-	-	0.0015	0.0025	0.0008	-
Si	0.0125	0.0197	0.0835	0.0166	0.0173	0.0121	0.0216
M	0.01530	0.00527	0.00204	0.00456	0.00012	0.01049	0.0040
O	0.0034	0.0012	0.0004	0.0010	0.0001	1.920	0.0043
Ca	0.0049	0.0085	0.0049	0.0054	0.0046	-	-
Cu	0.0253	0.0430	0.0291	0.0278	0.0237	0.00584	0.0004
Ag	-	-	-	-	-	0.00823	-
Al	0.0021	0.0020	0.0115	0.0020	0.0020	0.0023	0.0024
V	0.0005	0.0005	0.0005	0.0005	0.0005	0.0005	0.0005

^aDensities for trace quantities have been omitted for the following isotopes: ²³⁴U, ²³⁶Np, ²⁴¹Am, ²⁴²P, ⁸Ca, ²⁴Zn, ¹⁰B, ¹¹B, ⁴¹K, ⁴⁰K.

TABLE II
Comparison of TDOWN and MC-2 for ZEBRA-8 Cells

ZEBRA	ΔK_{NET}	ΔK_{EP}^a		ΔK^b	
		50GP LIBIV	171GP VITAMINEC	171GP-50 GP	TDOWN ^c -MC-2
8A	.0547	-.0085	-.0009	-.0002	.0006
8B	.0126	-.0032	-.0004	-.0014	.0041
8C	.0575	-.0011	-.0006	-.0011	.0041
8D	.0266	-.0056	-.0014	-.0010	.0042
8E	.0154	-.0019	-.0004	-.0003	.0043
8F	.0174	-.0057	-.0008	-.0001	-.0022
8H	.0021	+.0001	+.0001	-.0022	.0059

^a ΔK_{EP} = reactivity change associated with elastic removal cross-section adjustment due to deviation of actual spectrum from reference library spectrum.

^b ΔK = change in ΔK_{EP} due to group structure difference.

^cHomogeneous cell calculations: TDOWN in 171 group, MC-2 in 156 group.

TABLE III
Comparison of TDOWN and VIM for ZEBRA-8 Cells

ZEBRA	K_{EXP}	C/E (VIM) ^a			C/E (TDOWN) ^a		
		K=	28C/49F	28F/49F	K=	28C/49F	28F/49F
8A	.992	.971 $\pm$.003	1.041 $\pm$.006	.969 $\pm$.008	.9718	1.036	.956
8B	1.001				.9897	1.019	.998
8C	.986				.9581	1.039	1.003
8D	.973	.981 $\pm$.004	1.031 $\pm$.008	1.008 $\pm$.010	.9844	1.031	1.018
8E	1.006				.9734	1.029	1.030
8F	.971				.9892	1.036	.908
8H	1.030				.9957	1.038	1.093

^aENDF/B-IV based unadjusted cross sections.

that ZEBRA-8 be used further for evaluating ENDF/B-V and the associated adjusted cross-section libraries.

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Measurements and Analysis of Reactor Parameters

Critical Review: Thermal Reactor Benchmarks, O. Ozer (EPRI), D. R. Harris (RPI), R. Sher (Stanford Univ), W. J. Eich (EPRI)

INTRODUCTION

Thermal reactor benchmark experiments can be considered as falling into two major categories:

1. experiments used for integral testing of nuclear data
2. experiments used for validating reactor design and analysis methods.

Although some experiments can be used as benchmarks for both areas of application, in general the requirements of the two categories are quite different. Nuclear data benchmarks consist of simple-geometry, simple-configuration experiments dependent on as few components (cross sections) as possible. Such experiments are often designed to accentuate sensitivities to cross-section features of interest. Their accuracy requirements are very high, since to provide useful information, the experimental uncertainties must be lower than the combined uncertainties of the nuclear data and methods of analysis.

Core physics benchmarks, on the other hand, must be representative of the reactor system of interest. They are designed to test reactor analysis methods in their intended areas of application. Thus, characteristic physical features such as geometric heterogeneities, water gaps, voids, burnable poison rods, etc., must be represented.

NUCLEAR DATA BENCHMARKS

The Cross-Section Evaluation Working Group (CSEWG) Thermal Data Testing Benchmarks set¹ provides a compilation of experiments most frequently used in the United States for integral nuclear data testing activities. This set includes:

1. simple homogeneous critical spheres containing ^{235}U (uranyl nitrate) in H_2O [Oak Ridge National Laboratory (ORNL) spheres]
2. slightly enriched (1.3%) uranium metal and UO_2 lattice criticals in H_2O (Bettis-TRX Series)
3. Massachusetts Institute of Technology (MIT) experiments consisting of uranium metal exponential lattices in D_2O
4. homogeneous critical spheres and cylinders of plutonium nitrate in H_2O [Pacific Northwest Laboratory (PNL) homogeneous criticals]
5. mixed (uranium-plutonium) oxide fueled critical lattices (PNL)
6. ^{233}U -fueled thorium oxide exponential lattices in H_2O (Brookhaven National Laboratory).

Additional experiments consisting of ^{233}U -homogeneous criticals and D_2O -moderated natural uranium lattice criticals are also being considered for inclusion.

The objective of the CSEWG benchmarks is to provide a

comprehensive set of experiments for testing the adequacy of fissile and fertile material cross sections. The ORNL spheres and the TRX lattice criticals have been extensively used to evaluate the adequacy of ^{235}U and ^{238}U cross sections included in the ENDF/B data libraries. Both experiments are well defined and highly accurate. In addition, the TRX lattice experiments include measurements of ^{235}U and ^{238}U reaction rate ratios that have recently been reevaluated² to ensure the correct interpretation of experimental data and to account for flux perturbations introduced by the experimental apparatus. Accounting for such effects in the definition of the measured parameters simplifies the analysis considerably, since only reaction rates in an unperturbed flux need to be calculated, and there are no ambiguities regarding the definition of what is considered to be fast and thermal energy ranges.

Unfortunately, the remaining CSEWG thermal benchmark experiments do not come up to the high level of accuracy and usefulness of the first two sets. Parameters measured in the MIT exponentials have been shown to be internally inconsistent.³ Whether this inconsistency is endemic to exponential systems can be determined from the analysis of the D_2O -moderated critical systems being considered for inclusion in the CSEWG benchmark set.

Monte Carlo analysis of the homogeneous plutonium solution experiments has exhibited a wide scatter in the calculated eigenvalues that could most probably be explained by uncertainties in the isotopic compositions. The mixed-oxide criticals have accurate determinations of eigenvalues (rod worths and critical configurations), but include no measurements of reaction ratios. Moreover, the measured thermal flux disadvantage factors exhibit unacceptably large discrepancies (30%) when compared to calculated values, thus limiting the validity of these experiments for data testing applications.

CORE PHYSICS BENCHMARKS

Many of the benchmark experiments utilized to validate core physics computer program packages are proprietary. The following publicly available experiments are part of the set being used to validate the Electric Power Research Institute core physics programs⁴:

1. Westinghouse critical lattices with 2.75% enriched UO_2 pressurized water reactor (PWR) fuel in H_2O
2. Babcock and Wilcox (B&W) critical lattices including highly borated end points
3. ESADA mixed-oxide critical lattices in H_2O
4. high-conversion-ratio critical lattice measurements carried out at Argonne National Laboratory
5. uniform lattices as well as PWR and boiling water reactor (BWR) assembly mockup experiments carried out with U.S. Atomic Energy Commission UO_2 and mixed-oxide fuel at the KRITZ facility in Sweden⁵
6. measurements of isotopic concentrations in fuel irradiated at the following reactors: Yankee-Rowe, Saxton, H. B. Robinson,⁶ and Calvert Cliffs.⁷

The Westinghouse, B&W, and ESADA criticals provide a measure of the adequacy of methods of analysis in PWR applications, whereas the high-conversion-ratio experiments are representative of a highly voided BWR system. The experiments performed at the KRITZ facility are of particular interest, since they provide temperature-dependent data. The adequacy of burnup calculations can be tested against the Yankee and Saxton experiments as well as the more recent H. B. Robinson data. Measurements that are presently becoming available from cycles 3 and 4 of the Calvert Cliffs plant will provide information in the burnup range from 40 to 50 MWd/ton, of interest for extended fuel cycle studies.

DISCUSSION

Although the CSEWG nuclear data benchmarks are adequate for testing uranium cross sections, more accurate plutonium experiments are needed. The benchmark set should also be expanded to include temperature-dependent experiments, (e.g., Rensselaer Polytechnic Institute and Stanford ^{238}U self-indication measurements), as well as experiments sensitive to moderator properties (e.g., fission neutron age in water) and to control material properties (e.g., criticality of gadolinium loaded lattices). A considerable number of existing experiments performed in Europe and Japan as well as in the United States should be considered as candidates for extending the data and methods validation benchmark sets.

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Calculated Ratios of Spectral Densities for ^{252}Cf -Driven Neutron Noise Subcriticality Measurements with a 5%- ^{235}U -Enriched Uranyl Fluoride Solution, J. T. Mihalcz, W. T. King, J. A. Renier (ORNL)

A method that uses ratios of neutron noise power spectral densities in conjunction with a ^{252}Cf source¹ for determining the reactivity of a highly subcritical system of fissile materials was demonstrated in experiments with an unreflected, ^{235}U -enriched (5%), uranyl fluoride solution

tank.² This method simultaneously measures cross- and auto-power spectral densities (CPSDs and APSDs) between the fluctuating components of the signals from three detectors and calculates a ratio of the spectral densities ($G_{12}^* \cdot G_{13} / G_{11} \cdot G_{23}$) that is directly proportional to $\Delta k/k$ (Ref. 1). This spectral density ratio was formed by combining the measured CPSDs G_{12} and G_{13} between the signals from two ^6Li -loaded glass scintillators (detectors 2 and 3) placed 120 deg apart on the outer periphery of the tank at the solution mid-height, and the signal from a centrally located ionization chamber containing a ^{252}Cf source (detector 1), the CPSD G_{23} between the signals from the two external detectors, and the APSD G_{11} of the source.

The results from these experiments² show the applicability of this method for monitoring the subcriticality of fuel solution tanks in fuel processing or reprocessing facilities. This paper describes calculations of these spectral densities and the spectral density ratio as a function of frequency using the JPRKINETICS code³ and compares the calculated with measured results.

Fig. 1. Comparison of measured and calculated ratios of spectral densities for a 5%- ^{235}U -enriched uranyl fluoride solution experiment.

The kinetics calculation proceeded as follows. A detector adjoint calculation was performed for each detector location to obtain the various frequency-, space-, and energy-dependent transfer functions. A steady-state source calculation provided the fluxes for calculation of the distributed noise equivalent sources⁴ due to fission, capture, scattering, and axial and radial leakages. It was assumed that there were no correlations between types of events. The CPSDs were calculated using the transfer functions and noise equivalent sources. The calculations used two-dimensional diffusion theory, with axial leakage represented by increased absorption and three-group cross sections obtained from ENDF/B-IV. The calculated spectral densities were used to obtain the spectral density ratio. Although the JPRKINETICS code incorporates delayed neutron effects, they were omitted from these calculations because they are not significant for the frequency range of interest.

JPRKINETICS calculations were performed for solution heights varying from 20 to 33 cm; the calculated ratios of spectral densities agreed within 15% with the experimental ratios (Fig. 1). Noise sources other than fission must be included in the calculation of the CPSD between the two detectors (G_{23}) to calculate the experimental ratios. The calculated CPSDs G_{12} and G_{23} as a function of frequency for a solution height of 27 cm were used to obtain a break frequency of 285 Hz, which is within 10% of the experimental value of 310 Hz.

The good agreement between the calculated and measured parameters validates this calculational method for unreflected, low-enriched, fuel-solution systems with a central ^{252}Cf neutron source. Further comparisons between measurement and calculations for a variety of solution systems are required before acceptance of this calculational method for these types of experiments.

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Reactor Theory and Applications

Feasibility of LWR Subcritical Reactivity Monitoring Using the ^{252}Cf -Driven Neutron Noise Method, J. T. Mihalczo, W. T. King, J. A. Renier (ORNL)

Calculations were undertaken to ascertain if the subcriticality of large arrays of light water reactor (LWR) fuel pins can be measured by the ^{252}Cf -driven neutron noise method.¹ Such arrays are typical of those encountered in the initial fuel loading of LWRs and in spent fuel storage pools. To evaluate the applicability of this method, the calculations assessed the feasibility of measuring the reactivity of a large, fully loaded boiling water reactor (BWR) [1200 MW(e)] (Ref. 2) after the initial fuel loading has been completed with all control rods inserted, and thereafter with the central control rod or with the central rod and an adjacent rod withdrawn. Both static and kinetic, two-dimensional diffusion calculations estimated the quantities that would be measured. In addition, the Monte Carlo method determined the spatial distribution of the fission multiplicative processes and thereby qualitatively ascertained how the longer fission chains contribute to the counts registered by the detector.

The ^{252}Cf -driven neutron noise method obtains the reactivity from the ratio of spectral densities, $(G_{12}^2 \cdot G_{13}) / (G_{11} \cdot G_{23})$, where subscript 1 refers to an in-core ionization chamber (the source) containing ^{252}Cf that provides neutrons to initiate the fission chain multiplicative process, and subscripts 2 and 3 refer to two in-core detectors that detect neutrons from the fission chains.¹ This ratio of spectral densities equals the product of a constant C and $\Delta k/k$, where k is the neutron multiplication factor. The constant C does not depend on either the detection efficiency or the absolute source intensity inherent in the fissile fuel. Subcriticality measurements by this method have been successfully performed for a variety of small systems.^{1,3}

JPRKINETICS calculations⁴ (three-energy-group cross sections⁵) were performed for various source and detector locations in a fully loaded BWR core. The californium source was located at the center of the core, and the detectors were positioned symmetrically and colinearly at various distances from this source. The calculated ratio

Fig. 1. Calculated ratios of spectral densities, $G_{12}^2 \cdot G_{13} / G_{11} \cdot G_{23}$, as a function of distance from a central source for the Zimmer core.

of spectral densities is independent of detector position (Fig. 1) beyond 45 cm from the source, and changes with control rod withdrawal.³ For this large BWR, the ratio of spectral densities from the JPRKINETICS code agrees with that from point kinetic calculations for detectors more than 45 cm from the californium source and with all control rods inserted.

The neutron multiplication factor obtained from VENTURE calculations⁶ as a function of fuel loading and with the control rods inserted shows that when ~100 central fuel elements are loaded (out to a 90-cm radius), the neutron multiplication factor is essentially that of an infinite core ($k = 0.910$ for 112 elements, and $k = 0.928$ for 560 elements).

To further assess how large a section of the core the multiplication factor obtained by this measurement represents, Monte Carlo calculations⁷ were used to evaluate the spatial extent of the fission chain multiplicative process as a function of the number of fissions in a given fission chain. Since the longer fission chains extend over large distances from the core center, correlatable events between the californium source and a detector (G_{12}) or between the detectors (G_{23}) must come from fission chains that extend over regions of the core as large as the distance between detectors. For example, fission chains with fewer than 5 fissions are confined to within ~50 cm of the source, whereas fission chains with more than 1000 fissions extend to ~200 cm from the source. Correlatable events in detectors located far from the source are due to particles from long fission chains that extend over large volumes of the core and contain global information. Thus, these measurements can preferentially measure fundamental mode properties of the core.

Detection efficiency (counts per reactor fission) determines the measurement time required to attain the desired statistical precision and therefore limits the distances between detectors. Thus, detection efficiency determines the lengths of fission chains that provide correlatable events for calculation of cross power spectral densities. Measurements have been satisfactorily performed by this method, with detection efficiencies as low as 10^{-5} count per reactor fission.¹ For this large BWR, the calculated detection efficiency is greater than 10^{-5} for a 5-g ^{252}Cf fission detector within 70 cm of the core center. More efficient detectors such as ^6Li glass scintillators (factor of ~100 larger) could be used for unirradiated fuel, and measurements could be made with these detectors more than 100 cm from the core center.

These calculations indicate that subcriticality measurements by the ^{252}Cf -driven neutron noise method during the initial loading of a BWR are feasible. Such measurements would be independent of detector location over a large section of the core in which the detection efficiency would be sufficient for adequate statistical precision. Application to the highly poisoned arrays of LWR fuel pins encountered during the initial loading of pressurized water reactors and in fuel storage pools may also be feasible. For the nearly infinite systems, either multiple ^{252}Cf sources and detector pairs or a single source-detector pair that could be moved around the core may be required to monitor the reactivity of the total system. Further evaluation of the applicability of this method to arrays of LWR fuel pins should be done experimentally in a mockup of a partial core or fuel storage pool.

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