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REACTOR PHYSICS QUARTERLY REPORT

APRIL, MAY, JUNE 1969

August 1969

AEC RESEARCH &
DEVELOPMENT REPORT

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Reactor Technology

REACTOR PHYSICS QUARTERLY REPORT
APRIL, MAY, JUNE 1969

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REACTOR PHYSICS QUARTERLY REPORT

April, May, June 1969

FORWARD

The objective of the Reactor Physics Quarterly Report is to inform the scientific community in a timely manner of the technical progress made on the many phases of reactor physics work within the laboratory. The report contains brief technical discussions of accomplishments in all areas where significant progress has been made during the quarter. The results presented herein should be considered preliminary, and do not constitute final publication of the work. A list of publications and papers issued during the last quarter is included in the report. Anyone desiring additional information concerning the work reported herein is encouraged to contact the author directly.

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1.0 SUMMARY

REACTOR THEORY AND CODE DEVELOPMENT

The computer code FLARE, which was imported from Argonne's Code Center, has undergone considerable revision in order to make it operational on the UNIVAC 1108 computer and to expand its applicability to commercial nuclear power plants.

Revisions are being incorporated into the HRG code which relax the restrictions on the theory of neutrons colliding with absorber and/or moderator nuclei, and allow for a more accurate calculation of resonance absorption events.

A decrease in running time for the KENO Monte Carlo Code was obtained by programming the random function generating routines in fixed point and machine language.

Program Alchemy, which solves the point reactor burnup equations or any set of first order, linear differential equations with constant coefficients, has been converted to the Univac 1108.

Calculations made from Feshbach's "generalized optical potential" formalism with an appropriate nucleon-nucleon potential fit the elastic scattering cross section of O^{16} quite closely. Several papers in nuclear theory have been submitted for publication.

THERMAL REACTORS

The ratio of effective delayed-neutron fraction to mean neutron lifetime ($\hat{\beta}/\ell$) has been calculated for several H_2O -moderated and reflected lattices using the computer code KINPAR. The calculated ratios compare well with measured values.

Analytical correlations were made of measured worths of peripheral and internal fuel elements in PRTR. The spectral-spatial effects of nonuniform core boundary for the PRCF Eatch Core fueled experiments were also studied.

The endogenous pulsed source method of determining the reactivity status of a critical (or subcritical) assembly has been used in an experiment to measure the temperature coefficient of reactivity for a small core of 1/2 in. diameter rods of UO_2 -2 wt% PuO_2 (8% ^{240}Pu) in a 0.75 in. square pitch lattice, moderated with H_2O . Over a temperature range from 30 to 75 °C, the measured result was $-(1.02 \pm 0.05) \times 10^{-4} (\Delta k/k)/^\circ\text{C}$.

An empirical study has been started of the shape of "full-energy" peaks in gamma-ray spectra obtained from Ge(Li) detectors. The method of attack uses a recently developed general scheme for holding fixed any subset of parameters in the nonlinear least-squares program LEARN.

Calculations of power distributions have been performed using various models to represent the reflector region of a critical assembly. Improvements between measured and calculated power profiles resulted from use of a "spectral hardened" reflector region.

FAST REACTORS

A fuel management scheme is proposed which significantly reduces the amount of critical excess reactivity that must be compensated by the control system. The scheme utilizes three driver positions that are critically filled with follower but that are subsequently charged with fuel during the first two refueling operations. For previous schemes, the reactivity loss to equilibrium was between 0.019 and 0.026 δk , whereas, for the present case it is only 0.006 δk .

Flux calculations at several radial positions have been made for four subcritical states in FTR.

Relaxing the maximum allowed PuO_2 particle size specification to 100 microns in the as-fabricated fuel pellets is recommended.

Pressure drop calculations were made for driver fuel subassemblies considering fuel pin bundles with 0.050 and 0.056 in. diameter wire wrap. Calculations were made comparing pressure drops in fuel pin bundles with grid spacers to fuel pin bundles with wire wrap.

The individual hot channel contributors, along with the method of calculation and the maximum expected FTR fuel and clad temperatures, are presented.

CRITICAL MASS PHYSICS

Computed Critical dimensions and masses for unreflected and water reflected spheres of $^{239}\text{PuO}_2\text{-U}_{(\text{Nat})}\text{O}_2$ aqueous mixtures and for water reflected mixtures containing PuO_2 with 15 isotopic % ^{240}Pu are presented. Since few experimental data are available for checking the accuracy of the computed critical parameters, Monte Carlo calculations were made for some heterogeneous Pu-U mixtures and subsequently for homogeneous mixtures. These calculations provide a measure of the accuracy of the computed critical parameters of undermoderated aqueous mixtures of PuO_2 and UO_2 .

Critical experiments were performed in support of the Gas Cooled Fast Breeder Reactor (GCFR) Program to evaluate the reactivity worth of such resonance and thermal neutron poisons as dysprosium, gadolinium, and hafnium. The experiments were designed to check basic neutronic data and computational techniques used by Gulf General Atomic in their design analysis of the GCFR. Interpretation of the experimental data is being performed at Gulf General Atomic.

Criticality experiments were completed at the Critical Mass Laboratory with unmoderated PuO_2 . The experiments are a part of the basic research program to provide data for evaluating the effects of neutron moderation and ^{240}Pu on

intermediate to fast neutron spectra plutonium systems. Theoretical analysis and interpretation of these experiments, which provide the only critical experimental data on unmoderated PuO_2 , are under way.

Neutron surface multiplication measurements have been made on small cubes of plutonium-polystyrene material having an H/Pu ratio of 15 for samples constituting fractions of a critical mass ranging from only 0.44 to 28% of criticality. These initial experiments, although of limited accuracy, have established the feasibility of experimental measurements of surface multiplication: an index of criticality.

A study has been made on the impact of criticality safety on the fabrication of LMFBR fuel. It is concluded from this study that:

- 1) LMFBR fuel fabrication will be most limited by criticality safety in operations involving solutions.
- 2) Criticality safety should not seriously limit dry operations involving mixed-oxide powders and pellets.
- 3) Special controls will be necessary when handling subassemblies because fewer than ten will support a neutron chain reaction dry.
- 4) From the standpoint of criticality safety, reasonably high, competitive throughput rates should be possible for FBR fuel fabrication.

2.0 REACTOR THEORY AND CODE DEVELOPMENT

USE OF FIXED POINT ARITHMETIC IN MONTE CARLO CODES

C. W. Lindenmeier and J. L. Baker

Computer programs for generating the statistical distributions required by Monte Carlo codes are frequently programmed in assembly language to obtain maximum speed. Since many of the algorithms in current use were developed in the era before hardware floating point arithmetic was available, and since the fixed point fractional arithmetic hardware operations are faster than the floating point hardware operations (with the exception of division) on the Univac 1108, fixed point Sleuth versions of the following have been developed:

- (1) Von Neuman Uniform Angle. $\sin \phi$ and $\cos \phi$ are generated, where the angle ϕ is uniformly distributed over the range $(0, 2\pi)$. This method, developed by Von Neumann, selects two uniformly distributed random numbers where one (x) of the random numbers lies between $(0, 1)$, and the other (y) between $(-1, 1)$. The rejection test requires that $x^2 + y^2 \leq 1$. If the test fails, two new random numbers are generated and retested. If the test succeeds, $\sin \phi = 2xy/(x^2 + y^2)$ and $\cos \phi = (x^2 - y^2)/(x^2 + y^2)$.
- (2) Von Neumann Uniform Isotropic Unit Vector. Uniformly distributed, random direction cosines are generated as

$$\begin{aligned} a &= \sin \theta \cos \phi \\ \beta &= \sin \theta \sin \phi \\ \gamma &= \cos \theta \end{aligned}$$

where $\cos \theta$ is a uniformly distributed random variable and ϕ is a uniformly distributed azimuthial angle.

$\cos \phi$ and $\sin \phi$ are generated as described in (1) above. $\cos \theta$ and $\sin \theta$ are generated from uniformly distributed random numbers and a rejection test. Two random numbers (x,y) are selected from the range $(0,1)$. The rejection test requires

$$x^2 + y^2 \leq \frac{8xy}{3\sqrt{3}}$$

If the test fails, two new random numbers are selected and retested.

If the test succeeds

$$\cos \theta = \frac{x^2 - y^2}{x^2 + y^2}$$

$$\sin \theta = \frac{2xy}{x^2 + y^2}$$

- (3) Coveyou Uniform Isotropic Unit Vector. This method, developed by Coveyou selects three uniformly distributed random numbers where the first one z lies between $(0,1)$ and the other two ξ, ζ lie between $(-1,1)$. The random numbers ξ, ζ are rescaled to form $x = \xi * 3^{1/2}/4^{2/3}$, $y = \zeta * 3^{1/2}/4^{2/3}$. The rejection test requires $D = (x^2 + y^2 + z^2) \leq z$. If this test is satisfied then the three direction cosines are $a = 2xz/D$, $\beta = 2yz/D$, and $\gamma = (z^2 - y^2 - x^2)/D$.

The above algorithms are discussed in more detail in Alder, Fernbach, and Rotenberg.(1)

Table 2.1 shows a comparison of the time required to generate 20,000 random functions from routines of the same algorithm but differing in programming methods. The fixed point Sleuth routine AZIRN shows a 25% decrease in running time over the floating point Sleuth routine AZIR. The fixed point Sleuth routine ZGTISO shows a 33% decrease in running time over the FORTRAN routine FGTISO.

The random routines AZIRN and ZGTISO are currently being used in the computer code KENO, a multigroup Monte Carlo Criticality program that was recently imported from Oak Ridge. When the FORTRAN routines supplied with KENO were replaced by the above routines, the running time for the test case supplied with Keno decreased by 18%.

References

1. *Methods in Computational Physics, edited by Berni Alder, Signey Fernbach, and Manuel Rotenberg. Accademic Press, New York, 1963.*

TABLE 2.1.

<u>Subroutine Name</u>	<u>Algorithm</u>	<u>Programming Method</u>	<u>Time/20,000 Random Functions</u>
AZIR	Von Neumann Uniform angle	Floating Point Sleuth	1.504 sec
AZIRN	Von Neumann Uniform angle	Fixed Point Sleuth	1.120 sec
EGTISO	Von Neumann Isotropic unit vector	Floating Point Sleuth	3.115 sec
GTISO	Von Neumann Isotropic unit vector	Fixed Point Sleuth	3.003 sec
FGTISO	Coveyou Isotropic unit vector	Floating Point Fortran	3.497 sec
FRNCOS	Coveyou Isotropic unit vector	Floating Point Sleuth	2.798 sec
ZGTISO	Coveyou Isotropic unit vector	Fixed Point Sleuth	2.347 sec

ETGX, A CODE TO CALCULATE GROUP CONSTANTS FOR NUCLEAR REACTOR CALCULATIONS

R. E. Schenter. J. L. Baker and R. B. Kidman

1. Name of Code: ETOX⁽¹⁾

2. Computer for Which Code is Designed: Univac 1108.

Programming Language: "Standard" FORTRAN-IV.

3. Nature of Code: ETOX (ENDF/B⁽²⁾ TO 1DX⁽³⁾) calculates multigroup constants for nuclear reactor calculations using data from the Evaluated Nuclear Data File (ENDF/E⁽²⁾). The code is designed to compute and punch:
 - a) Infinite dilute cross-sections
 - b) Temperature dependent self-shielding factors for arbitrary values of σ_0 (total cross-section per atom) in the "Russian" (Bondarenko⁽⁴⁾) format
 - c) Inelastic scattering probability matrices.
4. Method of Solution: Microscopic cross section values are constructed as specified by the ENDF/B.⁽²⁾ Group constants are obtained by integrating the microscopic data over group intervals using the flux weighting scheme

$$\phi(u) \propto 1/\sum_t(u), \quad \sum_t(u) = \sum_j N_j \sigma_{tj} + N_0 \sigma_0.$$
 Integration methods used include Romberg,⁽⁵⁾ Gaussian Quadrature,⁽⁶⁾ N-point,⁽⁷⁾ Simpson,⁽⁶⁾ and Trapezoidal.⁽⁶⁾ The code allows as input, arbitrary sets of values of group energies, temperatures, and a_0 's.
5. Restriction on Complexity: ETOX uses random access drum storage to make maximum use of core.
6. Running Time: Twenty-six group data for a representative set of 14 isotopes requires about 28 min on the Univac 1108.
7. Related and Auxiliary Programs: The format of the input data is compatible with the ENDF/B data tape. The format of the output is compatible for input to the computer codes 1DX⁽³⁾ and FCC-IV,⁽⁸⁾ which are multipurpose diffusion codes for generating cross-sections to be used in fast-reactor analysis.
8. Status: In use at Battelle-Northwest.
9. Machine Requirements: A 65K memory, 3 tape units, and random access drum storage.
10. Material Available: A source deck, sample problem, and operation instructions are available from the authors.

References:

1. R. E. Schenter, J. L. Baker and R. B. Kidman. ETOX, A Code to Calculate Group Constants for Nuclear Reactor Calculations. BNWL-1002. Battelle-Northwest, Richland, Washington, 1969.
2. H. C. Honeck. ENDF/B, Specification for an Evaluated Nuclear Data File for Reactor Applications, BNL-50066 (T-46). Brookhaven National Laboratory, Upton, N. Y.
3. R. W. Hardie and W. W. Little, Jr. 1DX, A One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections, BNWL-954. Battelle-Northwest, Richland, Washington, 1969.
4. Group Constants for Nuclear Reactor Calculations, edited by I. I. Bondarenko. Consultants Bureau, New York, 1964.
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6. Handbook of Mathematical Functions, edited by M. Abramowitz and I. A. Stegun. Dover Publications, Inc., 1965. pp. 916.
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8. W. W. Little, Jr., and R. W. Hardie. FCC-IV, A Revised Version of the FCC Fundamental Mode Fast Reactor Code, BNWL-450, Battelle-Northwest, Richland, Washington, 1967.

TIME-VARIANT ISOTOPIC TRANSMUTATION(Battelle-Northwest Program Alchemy-II)

B. H. Duane

Machine Analysis Abstract

Program Alchemy-II solves the vector differential equation representation of time-variant isotopic transmutation

$$d\mathbf{N}(t)/dt = \mathbf{N}(t) \cdot \mathbf{H} + \sum_L \mathbf{Q}(L) e^{t\omega(L)}, \quad (1)$$

where $\mathbf{N}(t)$ is a row vector whose elements are the isotope concentrations (nuclei/volume) at exposure time t , the transmutation probability rates (/time) are the elements of the

asymmetric square matrix H , and $Q(L)$ is a row vector whose elements are discrete Laplace transform source-sink rates (nuclei/volume/time) having inverse period $\omega(L)$. The numeric solution is constructed by matrix power series evaluation of the matrix exponential analytic solution

$$\underline{N}(t) = \underline{N}(0) \cdot e^{tH} + \sum_L \underline{Q}(L) \cdot [e^{tH} - e^{t\omega(L)}] / [H - \omega(L)], \quad (2)$$

making exhaustive use of both exponent scaling to control series convergence rate and numeric rounding to minimize accrual of truncation error.

Input consists of all the quantities appearing on the right side of Equation (2), fed in via teletype, cards, bulk peripheral storage, remote transmitter, or fast memory input linkage. Output consists of the left side of Equation (2), fed out via plotter, teletype, line printer, remote transmitter, or fast memory output linkage.

Recent Refinements

During the past month, in response to a funded request from Douglas United Nuclear, the original IBM-7090 Fortran-II version of Program Alchemy was converted to the present Univac-1108 Fortran-V version, Program Alchemy-11. The analysis logic remains unaltered, to enhance retention of its accrual of six years of trouble-free performance in intensive chain-linked usage. New features include optional numeric rounding for best reliability or numeric truncation for best speed, plotter output, bulk peripheral storage input, and teletype input and output.

References

The physical theory and analysis logic of Program Alchemy-11 are developed in detail in the following sequence of documents available both at USAEC depository libraries and from the USAEC Technical Information Service Extension, P.O. Box 62, Oak Ridge, Tennessee 37830.

B. H. Duane. "Time-Variant Isotopic Transmutation, GE-HL Program *Alchemy*," *Physics Research Quarterly Report*, October, November, December 1963, HW-80020, pp. 4-7.

B. H. Duane. "Transmutation Logic for *S* Transport Analysis," *Physics Research Quarterly Report*,ⁿ July, August, September 1964, HW-84369, pp. 11-19.

E. H. Duane. "Transmutation Logic for *S* Transport Analysis," *Proceedings of the Conference on the Application of Computing Methods to Reactor Problems*, May 17-19 1965, ANL-7050, pp. 291-298.

THEORETICAL CALCULATIONS OF OPTICAL-MODEL POTENTIALS FOR USE IN NUCLEAR REACTION STUDIES ON LIGHT NUCLEI

R. E. Schenter

Introduction

This program is concerned with the determination of the nucleon-nucleus interaction from a knowledge of the fundamental nucleon-nucleon interaction and the appropriate many-body Schroedinger equation. The theoretical frame-work within which this study is being conducted is provided by Feshbach's "generalized optical potential" formalism.⁽¹⁾ It is believed that this formalism, when employed in conjunction with the "no polarization" approximation,⁽²⁾ provides a sufficiently flexible basis for meaningful nuclear reaction studies without involving prohibitively complex calculations.

The basic input "data" to this theoretical model are the ground state wave function of the target nucleus and the basic form of the nucleon-nucleon potential. Consequently, a comparison between the theoretical predictions of the model and the existing experimental data can provide insight into the proper form of the two-body potential as well as the wave functions which characterize the target nucleus.

Output from these calculations also include single-particle wave functions which describe the motion of the scattered nucleon. Recent theoretical calculations by other

groups⁽³⁾ have verified our previous "no-polarization" results⁽²⁾ that the single-particle wave functions differ considerably within the nuclear interior from those generated by phenomenological optical-model potentials. This confirmation could be quite significant since essentially all current nuclear reaction calculations employ single-particle wave functions obtained from phenomenological potentials.

Results

Calculations were made for the elastic scattering of neutrons by O^{16} targets. The transition matrix, τ , was divided into two components, one of which described the potential scattering and one which characterized resonance scattering. The potential scattering contribution was calculated in terms of the nucleon-nucleon interaction and the ground state wave function of the target. The resonance scattering was treated phenomenologically in terms of the usual Breit-Wigner expression. Several current nucleon-nucleon potentials which exhibit different spin-isospin dependences and contain both soft and hard cores were employed in these calculations. It was found that the results varied significantly with the form of the nucleon-nucleon potential. Energy dependent s and p wave phase shifts were calculated which fit the experimental results quite closely for one of the hard core forms. The p wave single-particle wave function for this case was found to be extremely damped within the nuclear interior.

References

1. H. Feshbach. *Ann. Phys.*, vol. 5, p. 357. 1959; *Ann. Phys.*, vol. 19, p. 287. 1962.
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HARD CORE EFFECTS WITH A SINGLE, SEPARABLE T MATRIX*

F. Ford,** E. Schenter, and L. Cassolat

The two-body interaction is divided into two parts, one of which gives rise to hard core effects; a δ' interaction is used to represent this part. The resulting equations for the T Matrix are solved exactly, and it is shown that the wave function vanishes everywhere within the hard core. The significant feature of this T matrix is that its hard core effects are isolated in a single term which is separable in momentum space.

EQUIVALENCE OF THE δ' AND THE HARD CORE POTENTIAL ■ ALS~~

R. E. Schenter, H. S. Zwibel and R. L. Cassola†

The introduction of the hard core potential into the two-body problem forces the system wave function to vanish if the interacting particles approach each other more closely than the hard core radius. In this paper it is shown that this "hard core effect" can also be obtained in the two-body problem by replacing the hard core potential with a δ' interaction of suitable strength.

BATTELLE REVISED FLARE (BRF-1) †††

N. E. Carter, W. L. Purcell, R. H. Holeman, W. C. Wolkenhauer

The computer code, FLARE⁽¹⁾ which was imported from Argonne's Code Center,⁽²⁾ has undergone considerable revision in order to make it operational on the UNIVAC 1108 computer and to expand its range of applicability to commercial nuclear power plants.

* *Abstract of paper submitted for publication in Phys. Rev. Letters.*

** *NASA Lewis Research Center, Cleveland, Ohio.*

† *AEC Faculty Appointee, Richland, Washington.*

†† *Abstract of paper accepted for publication in Am. Jour. of Phys.*

††† *Partially funded by Division of Production*

Modifications Required to Make Operational on UNIVAC 1108

- The input data to FLARE is read by means of a generalized input routine which uses free field format and repeat and skip counters. In order to unpack this free field data, a subroutine BITA⁽³⁾ was added in which the Field function is used for partial word manipulation.
- In the original version all variables used in FLARE appear in Equivalence statements. This was a source of confusion to the optimizer portion of the UNIVAC 1108 compiler; hence, all equivalence statements were removed and replaced by named common blocks.

Modifications and Corrections to Increase Flexibility

- The code was expanded to $17 \times 17 \times 16$ nodes. The INCLUDE and PARAMETER options on the 1108 are used to facilitate changing the dimensions of the problem to any desired by the user within the limits of $I \times J \times K \leq 4624$.
- Subroutines EDT1 and PCH1 were modified to handle problems as large as $34 \times 34 \times 34$. (This was in preparation for a larger version of BRF.)
- Subroutine CASE was revised to output more alphanumeric information concerning the problem being run. Also, an independent case is indicated by I and a dependent case by D on the pertinent input cards.
- Subroutine SETUP was changed so that alternating directions can be used in the source iteration. The direction of iteration is controlled by the variable NALT, which is an input parameter which follows CHI on card type 02.

NALT = 0, forward direction only
 = 1, alternate direction
- Subroutine CLEAR was added to clear common.

- Two of the four required scratch tapes were removed. The arrays E, V are written to drum and the refuel information remains on Scratch Tape D. Tape A is not needed.
- The function EX was replaced by the library function EXP.
- The array FLARC was removed since it was not needed.
- Subroutine EDT1 was modified for the 48-line pages used on the 1108. An error in EDT1 relative to calculation of average quantities in the 1/4 core option was corrected. In addition several mechanical errors in formats, etc. were corrected.
- All tape operations were modified to correspond to 1108 hardware.
- The transport kernels were modified to increase their flexibility. The kernels now in BRF are

$$WV_{ijk} = \left[(1-g_V) \sqrt{\frac{M^2}{2r}} + g_V \frac{M^2}{r^2} \right] \cdot GVN$$

$$WH_{ijk} = \left[(1-g_H) \sqrt{\frac{M^2}{2r}} + g_H \frac{M^2}{r^2} \right] \cdot GHN$$

or

$$WV_{ijk} = \frac{e^{-g_V a/M^2}}{GVN}$$

$$WH_{ijk} = \frac{e^{-g_H a/M^2}}{GHN}$$

where g_V , g_H , GVN , and GHN follow the variable CRNR on card type 01, respectively.

- The refuel option was modified so that I*J fuel elements can be described; the previous limit was 84. In addition, various errors in subroutine REFUEL were corrected.
- The following subroutines were deleted:
 - WTBT
 - RDBT

- SBITF
- SBITB
- RWD
- RWDLO
- BLKSIZ
- FSUP
- ISUP
- EX

Current Developments (BRF-2)

A basic limitation on BRF is that it can accept a maximum of $17 \times 17 \times 16 = 4624$ nodes, which is not sufficient to describe each fuel element in the large power reactors currently being built. By using the drum storage capability of the UNIVAC 1108, the BRF code is being extended to handle $34 \times 34 \times 34 = 39304$ nodes.

A preliminary version is operational and indicates that problem running time is not excessive (less than double the non-drum running time). Work is progressing on optimization of this version.

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THE INTERMEDIATE RESONANCE APPROXIMATION IN HRG

J. L. Carter

According to the procedure^(1,2) previously used in HRG for calculating the resonance integral of a resolved

resonance of an absorber, the resonance is classified as either wide or narrow relative to the energy loss of a neutron in a scattering collision with the absorber. All resonances are considered narrow relative to the energy loss in collisions with either admixed or external moderator atoms. These restrictions lead to simplified basic expressions involving functions which are evaluated from tabulated values. Revisions are currently being made in HRG which relax the restrictions on the absorber and admixed moderator collisions, yet give the same basic expressions with modified parameters. The revisions are particularly important for fuels with an admixed moderator because of the effect of the improved moderator treatment on the wide strong resonances at 6.7 eV in ^{238}U and at 1.06 eV in ^{240}Pu . To illustrate, resonance absorption decreases of about 3% in ^{238}Pu and 4.5% in ^{240}Pu and a reactivity increase of about 18 mk were calculated for a representative light water lattice with mixed oxide fuel containing 2 wt% PuO_2 with 8% ^{240}Pu . For comparison, a previously reported revision⁽³⁾ to HRG which allocates the contribution of a resonance to several fine groups gave, for the same lattice and fuel combination, absorption increases of about 5% in ^{238}U and 10% in ^{240}Pu and a reactivity decrease of about 25 mk. The effects of these two revisions provide a good illustration of the cancellation of errors in the former HRG resonance treatment. There is a net change, however, which results in improved theory-experiment correlation for a range of critical experiments.

To obtain the expressions used in the current revisions, we begin by observing that, for nonmetallic fuels, the results of Reference 1 for the narrow resonance (NR) and infinite mass (IM) approximations can be combined in the single expression:

$$I = \frac{c_o \Gamma_Y}{E_o} \left[\beta_\lambda J(\xi, \beta_\lambda) + L(t_\lambda, \xi, \beta_\lambda) \right], \quad (1)$$

where

$$\beta_{\lambda} = \frac{(1 - \lambda)\sigma_m + \lambda\sigma_p}{\sigma_o} \frac{\Gamma}{\Gamma_Y + \lambda\Gamma_n}$$

$$t_{\lambda} = \left\{ (1 - \lambda)\sigma_m + \lambda\sigma_p \right\} N\bar{x} ;$$

and $A = 0, 1$ for the IM and NR approximations respectively. The other parameters are in common use and are further defined in References 1 or 2. The two tabulated functions are

$$J(\xi, \beta_{\lambda}) = \int_0^{\infty} \frac{\psi}{\psi + \beta_{\lambda}} dx , \quad (2)$$

$$L(t_{\lambda}, \xi, \beta_{\lambda}) = \int_0^{\infty} \frac{\psi^2}{\psi + \beta_{\lambda}} P_o \left\{ \frac{t_{\lambda}}{\beta_{\lambda}} (\psi + \beta_{\lambda}) \right\} dx , \quad (3)$$

where $\psi = \psi(\xi, x)$ is the Doppler broadening function and P_o is the escape probability from the fuel. The form of β_{λ} immediately suggests that an improvement would result if a value of A intermediate to 0 and 1 were used. An investigation of this intermediate resonance (IR) problem in a homogeneous system was first described by Goldstein and Cohen⁽⁴⁾ who developed the basic iterative procedure for determining A . Further work by Goldstein^(5,6) and others^(7,8) has established the validity of the method, has introduced heterogeneity through the Wigner rational approximation, has extended the method to include both admixed and external moderators by inserting additional parameters, and has improved the procedure for determining the parameters.

These results can be readily adapted to HRG, but only selected ones have been included at this time. To include the effect of the admixed moderator in Equation (1) it is

only necessary to make a slight change in the definitions of β_λ and t_λ . These parameters become

$$\beta_{\kappa\lambda} = \frac{(\kappa - \lambda)\sigma_m + \lambda\sigma_p}{\sigma_o} \frac{\Gamma}{\Gamma_\gamma + \lambda\Gamma_n} \quad (4)$$

$$t_{\kappa\lambda} = \{(\kappa - \lambda)\sigma_m + \lambda\sigma_p\} N\bar{\ell} \quad (5)$$

In present notation, and neglecting the interference between resonance and potential scattering, the transcendental equations which are solved to find κ and λ are

$$\lambda = 1 - \frac{\tan^{-1} X_{\kappa\lambda}}{X_{\kappa\lambda}} \quad (6)$$

$$\kappa = 1 - \frac{\tan^{-1} Y_{\kappa\lambda}}{Y_{\kappa\lambda}}, \quad (7)$$

where

$$X_{\kappa\lambda} = \frac{4A}{(A+1)^2} \frac{E_o}{\Gamma} \left(\frac{\beta'_{\kappa\lambda}}{1 + \beta'_{\kappa\lambda}} \right)^{1/2}$$

$$Y_{\kappa\lambda} = \frac{4A_m}{(A_m+1)^2} \frac{E_o}{\Gamma} \left(\frac{\beta'_{\kappa\lambda}}{1 + \beta'_{\kappa\lambda}} \right)^{1/2}$$

$$\beta'_{\kappa\lambda} = \frac{(\kappa - \lambda)\sigma_m + \lambda\sigma_p + \sigma_e}{\sigma_o} \frac{\Gamma}{\Gamma_\gamma + \lambda\Gamma_n}$$

$$\sigma_e = 1/\bar{\ell}N \quad ,$$

and A and A_m are, respectively, the masses of the absorber and admixed moderator in units of the neutron mass. The steps in the HRG procedure are first to find κ and λ from Equations (6) and (7), then use these values in Equation (4) and (5) to

obtain the parameters to evaluate Equation (1). The one additional input number required is A_m .

The preceding discussion was confined to nonmetallic fuels; that is, $\sigma_m \neq 0$. For metallic fuels, $\sigma_m = 0$. In the NR approximation ($A = 1$), both $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ are greater than zero and Equation (1) is still used by Adler et al. (1). In the M approximation ($A = 0$), however, both $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ equal zero. Under these conditions,

$$\lim_{\beta \rightarrow 0} \beta J(\xi, \beta) = 0$$

$$\lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} L(t, \xi, \beta) = L'(t'', \xi) = \int_0^\infty \psi P_O(t'', \psi) d\psi, \quad (8)$$

where

$$t'' = \sigma_O N \bar{\ell} \frac{\gamma}{\Gamma},$$

and the subscripts on $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ have been suppressed for notational convenience. Adler et al., argue that in this limit the "volume" term, $\sigma_O \Gamma_Y / \xi_O \beta J(\xi, \beta)$, should approach the slowing down density in the fuel $20_{\text{pot}}/A$, rather than zero. For metallic fuels in the IM approximation, they replace the form of Equation (1) by

$$I = \frac{2\sigma_{\text{pot}}}{A} + \frac{\sigma_O \Gamma_Y}{E_O} L'(t'', \xi) \quad (9)$$

and tabulate the L' -function separately. In the current revisions the $20_{\text{pot}}/A$ term is omitted; its small effect is already included to some extent in the IR formulation. Since the L' term is a limiting form of the L -function, Equation (1) can thus be used in all cases. Numerical problems occur, however, whenever the values of β and t from Equations (4) and (5) lie sufficiently far below the

ranges covered by the L-tables that the extrapolation procedures normally used in HRG are inadequate. To cover this situation, two additional extrapolation formulas have been introduced, one for the L-tables and one for the L'-tables. Both are based on the Wigner rational approximation. It can be shown that, using this approximation, Equation (3) becomes

$$L(t, \xi, \beta) = (\beta + \beta/t) J(\xi, \beta + \beta/t) - \beta J(\xi, \beta) . \quad (10)$$

The identity

$$L(t, \xi, \beta) = L(t_m, \xi, \beta) \times \frac{L(t, \xi, \beta)}{L(t_m, \xi, \beta)}$$

can be converted into an extrapolation formula for L by using Equation (10) to evaluate the ratio on the right hand side.

The result is

$$L(t, \xi, \beta) = L(t_m, \xi, \beta) \times \frac{(\beta + \beta/t) J(\xi, \beta + \beta/t) - \beta J(\xi, \beta)}{(\beta + \beta/t_m) J(\xi, \beta + \beta/t_m) - \beta J(\xi, \beta)} , \quad (11)$$

where t_m is the smallest tabulated value of t . Similarly, the identity

$$\begin{aligned} \beta J(\xi, \beta) + L(t, \xi, \beta) &= \lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} \{ \beta J(\xi, \beta) + L(t, \xi, \beta) \} \\ &\times \left(\frac{\beta J(\xi, \beta) + L(t, \xi, \beta)}{\lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} \{ \beta J(\xi, \beta) + L(t, \xi, \beta) \}} \right) \end{aligned}$$

can be converted into an extrapolation formula for L'; the ratio is evaluated from Equation (10) and its limiting form, and the first factor is given by Equation (8). The result is

$$\beta J(\xi, \beta) + L(t, \xi, \beta) = L'(t'', \xi) \frac{(\beta + \beta/t) J(\xi, \beta + \beta/t)}{1/t'' J(\xi, 1/t'')} . \quad (12)$$

Equation (12) is used whenever both β and t are well below the ranges covered by the L-tables; this situation naturally arises for those strong, broad resonances in metallic fuels previously evaluated by Equation (9). If t is too small, Equation (11) is used.

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ANALYTICAL CORRELATIONS OF MEASURED POWER DISTRIBUTIONS

V. O. Uotinen, G. L. Gelhaus and S. R. Dwivedi

In the analysis of H₂O-moderated and reflected, clean, critical experiments, one generally assumes the reactor to be composed of two regions, core and reflector. Few group

cross sections for core and reflector are calculated, assuming that an infinite-medium spectrum applies in each region.

This two-region, infinite-medium model is generally adequate for the calculation of k_{eff} , but does not satisfactorily predict the power peaking near the core-reflector boundary. The reason for this is that in reality the spectrum near the boundary is modified, i.e., the spectrum in the core adjacent to the reflector is softer than the spectrum in an infinite medium of core material, and the spectrum in the reflector adjacent to the core is harder than the spectrum in an infinite medium of water. The two-region, infinite-medium model does not take into account this modification of the spectrum near the core-reflector boundary.

Calculations of power distributions have been performed using a modified model. The modification involves use of two reflector regions. The inner reflector region (one lattice unit thick, adjacent to the core) is represented by cross sections that are averaged over a spectrum characteristic of the core; the outer reflector region is represented by cross sections characteristic of an infinite H_2O medium.

This modified model, which partially accounts for the spectrum change near the boundary, has been applied to several lattices. Theory-experiment correlations of power distributions are improved in all cases by the use of this modified model. The type of improvement obtained is illustrated in Figure 2.1. The results shown in Figure 2.1 were obtained with the one-dimensional, diffusion-theory code HFN using four-group cross sections from calculations with the codes THERMOS and HRG. Similar improvement was observed when a modified reflector region was used in two-dimensional (x-y) calculations, with the code EXTERMINATOR-2, of power distributions measured in the CNEN-BNW experimental program.

FIGURE 2.1. Effects of Core-Reflector Constants on Radial Power Ratio

Further improvement would be expected if one were to also modify the cross sections of the outermost core region. It appears that appropriate cross sections for both modified regions (i.e., outer core and inner reflector) may be obtained from a cell calculation that contains an extra thickness (e.g., one mean free path) of moderator. This approach is being investigated and will be tested against clean lattice experiments and against experiments involving water holes, slots and crosses. The latter experiments more closely resemble actual power reactor configurations and thus provide more severe and more meaningful tests of the adequacy of the calculational model.

The present preliminary investigation has also pointed out differences between calculations with several standard diffusion theory codes, namely HFN, EXTERMINATOR-2 and 2DB. Power distributions calculated with these three codes are significantly different (Figure 2.2). Calculated values of k_{eff} also differ slightly for the three codes, and it is interesting to note that the discrepancy in power distribution is correlated with the discrepancy in k_{eff} .

MEASUREMENT OF THE TEMPERATURE COEFFICIENT OF REACTIVITY USING THE ENDOGENOUS PULSED SOURCE METHOD

J. B. Edgar

Introduction

The reactivity status of a critical assembly can be determined by observing the decay of spontaneous bursts of prompt neutrons. The measured quantity is the decay constant α , in the equation $n(t) = n_0 e^{-\alpha t} + \langle n \rangle$, which describes the time decay $n(t)$ of a burst of amplitude n_0 above the long-time averaged counting rate $\langle n \rangle$. This technique, called the endogenous pulsed source method,^(1,2) has been applied to a variety of experiments which required measurement of the reactivity status of a critical assembly.⁽³⁾

FIGURE 2.2. Comparison of Radial Power Ratios

This method has been applied recently to the measurement of the temperature coefficient of reactivity for a small core of 1/2-in. diameter rods of UO_2 -2 wt% PuO_2 (8% ^{240}Pu), arranged in a square pitch lattice with a center-to-center spacing of 0.75 in. and a moderator-to-fuel volume ratio of 1.55. A comprehensive series of measurements had previously been made in this core under the joint USAEC-CNEN program⁽⁴⁾ in the Plutonium Recycle Critical Facility (PRCF). This work represents an extension of that effort.

Experimental Details

The value of the prompt neutron decay constant at delayed critical, α_c , was determined for the core fueled with UO_2 -2 wt% PuO_2 rods by measuring values of α at several known levels of subcriticality and extrapolating to the value at delayed critical. This procedure was necessitated by the relatively large spontaneous fission rate in the plutonium-enriched fuel. The reactor was made critical with 319 fuel rods at an ambient temperature of 32 °C. With all controls withdrawn and fully reflected with H_2O on all boundaries, the loading had a measured excess reactivity of 15.64. The reactor was made subcritical by insertion of a calibrated control sheet located in the radial reflector. A plot of the measured inverse multiplication versus the measured value of a at each step was extrapolated to $1/M = 0$ to obtain the value of a at delayed critical. The results of these measurements are presented in Table 2.1.

At the beginning of the measurements to determine the variation of a with temperature, four additional fuel rods were removed from the periphery of the core, leaving a total of 315 rods. Based on prior determinations of peripheral fuel rod worth, the core was approximately 40¢ subcritical with all control mechanisms withdrawn and infinitely reflected. Measurements of the reactivity status were made at intervals of

approximately 5 °C, over the range from 30 to 75 °C, by determining the values of a at each temperature step. The results of these measurements are presented in Table 2.2 and in Figures 2.3 and 2.4.

TABLE 2.1. *Measurements Used in Extrapolation to Critical a*

Control Sheet S-2 Position, % Withdrawn	a (seconds) ⁻¹	Multipli- cation	Reactivity, dollars
46%	204.2 ± 1.5 (0.7%)	0.216	-0.517
42.25%	222.2 ± 1.8 (0.8%)	0.272	-0.651
39%	240.9 ± 1.9 (0.8%)	0.326	-0.780
36%	253.7 ± 2.0 (0.8%)	0.380	-0.909
33%	274.7 ± 2.8 (1.0%)	0.430	-1.029
30%	295.7 ± 5.9 (2.0%)	0.482	-1.154

Extrapolated Critical $a = 135.1 \pm 1.0$ (seconds)¹

Reactivity Calibration: Delayed critical to prompt critical
interval = 1\$ = 0.418 inverse
multiplication units

TABLE 2.2. *Measured Values of Alpha and Reactivity
as a Function of Temperature*

Temperature, °C	a (seconds) ⁻¹	Reactivity, dollars
30.48	202.7 ± 1.7 (0.8%)	-0.500 ± 0.015 (3.0%)
37.42	217.8 ± 2.2 (1.0%)	-0.612 ± 0.018 (2.9%)
41.37	229.6 ± 1.8 (0.8%)	-0.700 ± 0.016 (2.3%)
47.18	247.2 ± 2.3 (0.9%)	-0.830 ± 0.020 (2.4%)
50.09	279.3 ± 2.4 (0.9%)	-1.067 ± 0.021 (2.0%)
52.74	275.0 ± 2.3 (0.8%)	-1.035 ± 0.022 (2.1%)
55.56	285.1 ± 2.6 (0.9%)	-1.100 ± 0.022 (2.0%)
58.32	296.2 ± 3.3 (1.1%)	-1.192 ± 0.026 (2.2%)
63.81	317.0 ± 3.8 (1.2%)	-1.346 ± 0.031 (2.3%)
66.57	343.6 ± 3.8 (1.6%)	-1.558 ± 0.031 (2.0%)
69.48	338.6 ± 4.6 (1.4%)	-1.506 ± 0.036 (2.4%)
71.82	358.4 ± 5.3 (1.5%)	-1.653 ± 0.042 (2.5%)
74.03	350.0 ± 5.3 (1.5%)	-1.591 ± 0.041 (2.6%)

FIGURE 2.3. Variation of Alpha with Moderator Temperature UO_2 -2 wt% (8% ^{240}Pu) Rods in 0.75 in. Square Pitch Lattice Moderator/Fuel Volume Ratio = 1.55

FIGURE 2.4. Variation of Reactivity with Moderator Temperature UO_2 -2 wt% 18% ^{240}Pu) Rods in 0.75 in. Square Pitch Lattice Moderator/Fuel Volume Ratio = 1.55

Due to the extended period of time (up to three hours) required to accumulate experimental data for alpha measurements, thermal stratification of the moderator was anticipated as a problem. Also, at the higher temperatures, the moderator temperature loss was expected to be significant over the period of the measurement. Modifications were made to the reactor hardware to permit continuous mixing and heating of the moderator to assure that the moderator temperature remained constant and uniform over the time span of measurement.

Discussion of Analysis

The value of a was determined for each measurement by fitting the observed prompt neutron die-away to the equation $n(t) = n_0 e^{-at} + \langle n \rangle$ using Program LEARN. (5)

Alpha critical (α_c) had been determined to be 135.1 ± 1.0 seconds⁻¹. It was used to convert the measured a values obtained during the temperature experiment to levels of subcriticality in dollars (\$) by the following relationship:

$$\text{Level of subcriticality (\$)} = \frac{\alpha_c - a}{\alpha_c}$$

This alpha-reactivity relationship has been previously demonstrated by Pacilio⁽⁶⁾ to be linear up to about 7 dollars subcritical. This conversion relationship is not strictly valid for these measurements since the neutron lifetime in these cores increases slightly with increasing moderator temperature. This effect is estimated to reduce the measured values of reactivity by about 2%. No corrections have been made to the measured data for the change in neutron lifetime in the results reported here.

The a and reactivity versus temperature data listed in Table 2.3 have been fitted with a linear function, as shown in Figures 2.3 and 2.4, respectively. The coefficients of the fitted function and their one standard deviation uncertainties are listed in Table 3.5.

TABLE 2.3. *Coefficients of the Linear Relationship*

<u>Equation</u>	<u>A</u>	<u>B</u>
$\alpha(T) = A + BT$	82.5 ± 9.0	3.70 ± 0.18
$\rho(T) = A + BT$	$(38.5 \pm 6.5) \times 10^2$	$-(2.73 \pm 0.13) \times 10^{-2}$

Differentiation of the fitted relationships yield the temperature coefficients of α and reactivity, as shown below.

$$\frac{d\alpha}{dT} = 3.70 \pm 0.18 \text{ alpha units}/^{\circ}\text{C}$$

$$\frac{d\rho}{dT} = -(2.73 \pm 0.13) \times 10^{-2} \text{ } \$/^{\circ}\text{C}$$

Using a value of 3.74 mk per dollar, the temperature coefficient of reactivity is $-1.02 \times 10^{-4} \frac{\Delta k}{k}/^{\circ}\text{C}$.

The ability of the endogenous pulsed source method to determine the reactivity status of a subcritical assembly has a distinct advantage over other techniques for measuring changes in reactivity status with changes in temperature. The advantage is that the core loading (number of rods) and geometry remains unchanged throughout the full temperature range of the experiments. Thus, these measurements are much more amenable to direct calculation than are measurements which require a series of stepwise changes in the core loading.

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THE SHAPE OF PEAKS IN GAMMA-RAY SPECTRA FROM Ge(Li) DETECTORS

D. A. Kottwitz

The accurate, quantitative analysis of complex gamma-ray spectra obtained from Ge(Li) detectors depends on a quantitative understanding of the basic shape of a "full-energy" peak and its immediate vicinity. This information is developed best from an empirical study of the intense, isolated peaks provided by standard sources. During this quarter, we have started work on this problem, using the nonlinear least-squares computer program LEARN-I-A.

The shape of a full-energy peak depends generally on processes occurring in the detector crystal, amplifier circuitry, and the analogue-to-digital converter. Since it also depends on the operating conditions, e.g., voltages and time constants, great variety is possible. A typical example obtained at BNW (with features exaggerated for clarity) is indicated below by the solid line. There are three notable characteristics: (1) the high-energy side is well represented by a Gaussian function, (2) the low-energy side has a broad tail, and (3) there is long, flat shelf at energies below the tail. The asymmetry generally cannot be removed by subtracting a resolution-broadened shelf (shown by a dashed line). The purpose of the present effort is to find a simple but accurate representation of this asymmetry, which will be suitable for

incorporation into a least-squares spectrum-analysis computer program. The results obtained so far are too preliminary to be reported.

The methods being used to study the gamma-peak shape problem are based on a more complete and correct treatment of the problem of holding fixed any subset of parameters in the non-linear least-squares program LEARN. The original version of LEARN⁽¹⁾ provided only limited capability of doing this directly. The present scheme is based on a systematic expansion and contraction of the set of free parameters (and associated quantities). It is implemented by adding four more-or-less standard batches of FORTRAN cards to subroutines PUTIN, THEORY and PUTOUT. The scheme has been designed for adoption in existing LEARN packages with a minimum of reprogramming. A write-up of the method together with sample batches of cards is being prepared for distribution to any interested users of program LEARN.

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3.0 THERMAL REACTORS

THEORETICAL CORRELATION OF THE PRTR BATCH CORE EXPERIMENT

S. Matsuura* and D. L. Prezbindowski

Introduction

A detailed theoretical analysis is being made of the Batch Core fuel experiments conducted in the Plutonium Recycle Critical Facility (PRCF)⁽¹⁾ and the Plutonium Recycle Test Reactor (PRTR).⁽²⁾ The purpose of the analysis is to determine the limitations of certain calculational methods in their application to predicting the behavior of the PRTR Batch Core experiment and more fundamentally in their application to predicting the behavior of plutonium-fueled thermal reactor systems. The calculational methods selected for this analysis are HTH⁽³⁾ and HAMMER⁽⁴⁾ multigroup cross section codes and the HFN,⁽⁵⁾ FLOG,⁽⁴⁾ EXTERMINATOR⁽⁶⁾ 2DB^(7,8) PERT4⁽⁹⁾ and KINPAR⁽¹⁰⁾ reactor codes. The results of the analysis will provide a basis for selecting methods appropriate to the calculation of the PRTR Batch Core experiments and identify those areas in which further theoretical development is required.

Results of the analysis of the PRCF and PRTR critical experiments were reported earlier.⁽¹¹⁾ This report contains results of a theory-experiment correlation of the worths of peripheral and internal fuel elements in the PRTR. It further contains an analysis of the effects of an inaccurate representation of PRCF core leakage and nonasymptotic spectra on k_{eff} and β/λ .

Correlation of PRCF Critical Experiments

A series of analytical correlations of the PRCF critical experiments to determine the feasibility of the present analytical methods have been reported.⁽¹¹⁾ The results showed

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that the calculated effective multiplication factors were definitely larger than the measured values by amounts of 0.3 to 1.9% $\Delta k/k$. A study has been conducted to find the main causes of this trend, as well as to define the size of inevitable error (e.g., the error caused by manufacturing tolerances) in the correlation. The effects on effective multiplication factors due to the approximate prescriptions representing the measured system in the calculation, and the experimental uncertainties have been additionally studied. The most essential and difficult problem among them seemed to be the validity of cylindricalization and homogenization of the reactor cores, since the cores were actually highly heterogeneous assemblies with rod cluster fuels. There were two main problems to be considered for these processes of homogenization and cylindricalization:

- A. The cell calculation as the base of homogenization was made under the assumption of an infinitely repeated lattice and did not take into account the spatially transient spectrum at the core-reflector boundary. The fuel cells which were close to the reflector should have had a softer spectrum than the interior fuel cells because their cell spectrum was affected by the surrounding reflector. Similar considerations could be applied inversely for the peripheral reflector. The problem was to determine the transient spectrum effect on k_{eff}
- B. In order to perform 1D calculations in the horizontal direction, the cores were approximated by cylinders. The equivalent core radii were determined in such a way that the cylindricalized cores had same volumes with the original assemblies of hexagonal cells. This cylindricalization of the core would have underestimated the effective area of the core surface because of the neglect of the unevenness of core surface. This would in turn, have caused an underestimation of the leakage rate.

Determining the size of the leakage correction for k_{eff} was another problem. These transient effects at core-reflector boundaries were considered by two methods. The first method concerned only A) and did not include B). It treated the core as consisting of three regions, e.g., the standard fuel cell region, the boundary region, and the asymptotic reflector region. The boundary region was a mixed region of peripheral fuel cells and a part of reflector. Since it was not obvious how much additional moderator should have been added to the fuel cell to form the boundary region, a set of parametric cell calculations was performed as a function of the moderator-to-fuel volume ratio. The results were used for the cylindricalized 1D-three-region reactor calculation with a varying boundary region thickness. These calculations were made for the 13-element core with HAMMER. The moderator-fuel volume ratios of the boundary region were 18.4, 24.6 and 30.1 in addition to 12.6 (standard cell). From the results of these calculations, the changes of spectrum, microscopic and macroscopic cross section and other cell constants due to the change of moderator-to-fuel volume ratio were observed. The calculated values of k_{eff} were 1.0175, 1.0220, 1.0229 and 1.0206 for the 12.6, 18.4, 24.6, and 30.1 moderator-to-fuel volume ratio cores respectively, while the measured k_{eff} value was 1.066. Although this method did not improve the calculated results, it indicated how the microscopic cross-sections in the fuel cell changed with the addition of moderator, and the results were used in the second method.

The second method considered problems A) and B). For problem A, the cores were considered to consist of four regions: the standard fuel cell region, the boundary fuel cell region, the boundary reflector region, and the asymptotic reflector region. The boundary fuel cell was a peripheral fuel cell adjacent to the reflector. On the other hand, the boundary

reflector region was a part of the reflector adjacent to the fuel cells. The boundary region had some intermediate spectrum and microscopic cross sections between those of standard fuel cell and those of a well-moderated cell. After several trials to determine the proper mixture of the microscopic cross-sections of the boundary regions, the simple average of the macroscopic cross sections of the asymptotic regions on either side of the boundary regions was taken as the microscopic cross sections of the boundary region. Because the maximum change of microscopic cross section between the two extreme spectrums was about 10%, and the trend of changes was monotonic, the difference of final results in k_{eff} due to the difference of microscopic cross-section estimating method was less than 2.5 mk, even in the most sensitive case of the 7-element core. The effect on k_{eff} due to the change in the microscopic cross sections was estimated by perturbation theory and the results are shown in Table 3.1. It shows that the transient spectrum effect on k_{eff} is less than about +5 mk, and it decreases with the increase of core size.

The effects on k_{eff} due to the additional leakage from the uneven surface of the core were estimated by considering the effective surface area for the assembly of hexagonal cells and by equating the effective surface-cylindricalized surface ratio and the ratio of the geometrical bucklings. The effective surface area of a core was obtained from the calculation of the average probability of neutron leakage from each surface of an assembly of hexagonal cells, and the critical equation of the point reactor model was used to obtain the effect on k_{eff} due to the change of geometrical buckling. The final results are shown in Table 3.1. It shows that the leakage effects are much larger than the spectrum effects, and the amount of the effect depends on the degree of unevenness of core surface and the size of the core. The preceding discussion suggests the following conclusions concerning the inherent uncertainties in the correlation.

TABLE 3.1. Spectrum and Leakage Correction to k_{eff}

PRCF Core	Calculation Method(a)	Calculated, k_{cal} (not corrected)	Experimental, k_{exp} (b)	Difference, Δk $= k_{cal} - k_{exp}$	Boundary Spectrum Correction, Δk_s	Surface Leakage Correction, Δk_l	Corrected k_{cal}	Difference, $\Delta k'$ $= k'_{cal} - k_p \times \rho$
7	1	1.0248		+0.014			1.0186	+0.008
	2	1.0226	1.011	+0.012	+0.0049	-0.0111	1.0164	+0.005
	3	1.0297		+0.019			1.0235	+0.013
13	1	1.0208		+0.015			0.9989	-0.007
	2	1.0193	1.006	+0.013	+0.0035	-0.0254	0.9974	-0.009
	3	1.0249		+0.019			1.0030	-0.003
28	1	1.0093		+0.002			1.0045	-0.003
	2	1.0010	1.007	+0.004	+0.0010	-0.0058	1.0062	-0.001
	3	1.0133		+0.006			1.0085	+0.002

(a) Calculation method 1 = HAMMER 1D-4 groups, 2: HTH 1D-6 groups,
3 = EXTERMINATOR 2D-6 groups.

(b) These values were revised from previously reported values by the result of
re-checking the experimental data of control rod worths.

- The present 1D method with measured vertical buckling is suitable to analyze a large core within an accuracy of 5 mk.
- For a small core, agreement within 10 mk between measurement and calculation cannot be obtained without appropriate correction for the core-reflector transient spectrum effect and the leakage effect due to the unevenness of core surface.
- No clear cut superiority is evident between HHH and HAMMER, and between energy group schemes.
- A number of problem areas in the calculational model remain; they are: the joining region of the thermal spectrum, the feasibility of the cylindrical approximation of the 19-rod cluster fuel element, and the relation between thermalization and the low lying plutonium resonance absorption.

The following results were found regarding error contributions due to unavoidable errors in the correlation.

- The effect of manufacturing tolerance (especially with the tolerance of fuel density $9.45 \pm 0.09 \text{ g/cm}^3$) is ± 0.53 mk.
- The effect of uncertainty of measured vertical buckling (0.4% in B^2) is about ± 1 mk.
- The effects of uncertainties of measured B^{10} concentration (50.05 wppm) are ± 1.6 , ± 1.0 and ± 0.6 mk for 7, 13, and 28 element cores respectively.
- The effect of uncertainties of measured moderator level: negligible.

The uncertainties in the residual control rods worth are ± 0.4 , ± 0.2 , and ± 0.2 mk for 7, 13, and 28 element cores respectively.

From these results, it could be concluded that the final limitation of the accuracy of the analytical correlation with measured values are ± 1.3 to ± 2.0 mk for the PRCF cores.

The Effect in β_{eff}/ℓ Due to the Transient Spectrum and the Change of Leakage Rate

The theoretically predicted values of β_{eff}/ℓ which were previously reported differed significantly with the measured values. The theoretical values of β/ℓ were obtained by KINPAR.⁽¹⁰⁾ The error in the calculated value of β/ℓ due to the transient spectrum effect, was estimated by applying the changes in Σ_f to the expression for calculating β/ℓ used in KINPAR. The result for the 13-element core showed that the relative change of β/ℓ due to the transient spectrum was about 0.34%.

The neutron lifetime ℓ is essentially determined by the average neutron speed, the average cross-section and the nonleakage probability. The effect on ℓ due to the change of leakage rate of neutrons was estimated by the simple point reactor model. The resultant effect for the difference of leakage between the cylindricalized core and the assembly of the hexagonal cells was estimated to be about 0.3% to 1.0%. Therefore, the discrepancies between the measured and calculated values of β/ℓ cannot be explained in terms of the spectrum and leakage effects alone. These studies also indicate that the results of measurements of β/ℓ in small critical assemblies are much more applicable for evaluating calculational methods used for large reactors than are results of measurements of k_{eff} . This occurs because the spectrum and leakage effects in small critical loadings are large compared to the measurement inaccuracies in k_{eff} but small compared to the measurement inaccuracies in β/ℓ .

Correlation of the PRTR Critical Experiment

Worth of a Peripheral Fuel Element

The worth of a peripheral fuel element has been calculated using the 2DB code with the hexagonal geometry option. This

option performs an estimate of the peripheral fuel element worth for the 55-element core by representing the added fuel element as a homogenized, hexagonal fuel cell, rather than an added annulus as is done by HAMMER. The calculated worth is 1.45 mk. The experimental worth is 1.75 mk. The difference is -0.30 mk. (-17%)

Worth of an Interior Fuel Element

The calculated worth of fuel element 6069 in PC-1748 is 8.75 mk; the experimental worth is 7.4 mk. The difference is +1.35 mk (+15%).

Worth of the Creep Capsule

The calculated creep capsule worth is 0.3015 mk. The creep capsule worth has not been measured, however, such measurements are planned. The error in the calculation of fuel element worth falls in the range of 10-20% for most correlations so far performed: HAMMER (PRCF 8 el, 14 el, PRTR 31 el) and 2DB hex. (PRTR 55 el). The hexagonal description seems to have no defined superiority over the annular description for this correlation.

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EXPERIMENT-THEORY CORRELATIONS OF $\hat{\beta}/\ell$

V. O. Uotinen and S. R. Dwivedi

The ratio of effective delayed-neutron fraction to mean neutron lifetime ($\hat{\beta}/\ell$) has been calculated for several H₂O-moderated and reflected lattices using the computer code KINPAR.⁽¹⁾ The calculated ratios compare well with measured values.

Over the last few years the ratio $\hat{\beta}/\ell$ has been measured, using reactor noise techniques, in several light water

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lattices⁽²⁻⁵⁾ in the PRCF. The experimental technique consisted of obtaining a magnetic tape recording of the reactor noise, frequency-analyzing the tape with a multichannel frequency-analyzer, and least squares fitting an equation of the form:

$$y = A_1 + \frac{A_2}{\omega^2} \frac{\omega^2 + A_3}{\omega^2 + A_4}$$

to the data points. In the above equation y is the power per unit bandwidth, and ω is 2π times the frequency; A_1 is the white noise background, A_2 characterizes the amplitude of the plateau, $(A_3)^{1/2}$ is 2π times the low frequency breakpoint, and $(A_4)^{1/2}$ is $\hat{\beta}/\ell$.

Another method⁽⁶⁾ of deducing $\hat{\beta}/\ell$ has been to extrapolate to critical from subcritical measurements of the prompt neutron decay constant, a .

The computer code KINPAR⁽¹⁾ has been used to calculate values of $\hat{\beta}/\ell$ for these lattices. The code uses standard perturbation theory expressions for β and a . KINPAR is linked to the one-dimensional, multigroup, diffusion theory code HFN, from which it obtains direct and adjoint fluxes for the perturbation theory expressions. The isotopic delayed neutron fractions used in KINPAR are derived from absolute yields (delayed neutrons/fission) reported by Keepin et al.⁽⁷⁾ and Cox⁽⁸⁾ and are consistent with values reported in ANL-5800. Velocities, prompt neutron spectra, and values of $v\sigma_f$ were obtained from the computer codes HRG and THERMOS for six neutron energy groups, with group boundaries at 10 MeV, 0.498 MeV, 0.235 MeV, 11.7 keV, 2.38 eV, 0.683 eV and 0.0 eV. All of the delayed neutrons were assumed to be in the energy group between 0.498 MeV and 0.235 MeV. For two of the lattices a four-group calculation was also done in which there was only one group between 10 MeV and 11.7 keV. This four-group scheme made no differentiation between the spectra of prompt and delayed neutrons.

Calculated and measured values of $\hat{\beta}/\lambda$ are compared in Table 3.2. Calculated values ($6 \frac{\text{GROUPS}}{\text{sec}}$) are consistently larger than measured values, but agree well within experimental uncertainties in all cases but one. (In this isolated case the measured value was deduced from an extrapolation to critical of subcritical measurements of α .) Calculated values of $\hat{\beta}/\lambda$, obtained with four energy groups, are approximately 10% smaller than values obtained with six groups. This indicates that the difference in age, τ , (and thus also the leakage, τB^2) between the prompt and delayed neutrons is significant.

Some improvement in the calculated $\hat{\beta}$ could be expected with a more exact representation of the delayed neutron spectrum, and some improvement in the calculation of λ could be expected with the use of multiple thermal groups. Nevertheless, for the uniform light water lattices investigated here, the six-group calculational scheme that has been used gives good agreement between measured and calculated values of $\hat{\beta}/\lambda$.

TABLE 3.2. *Measured and Calculated Values of $\hat{\beta}/\lambda$, sec^{-1}*

	<u>Lattice Description</u>	<u>Measured</u>	<u>Calculated</u> 6 GROUPS 4 GROUPS	
1.	2.35% enriched UO_2 (Ref. 5) 0.75 in. sq. lattice	159 $\pm$ 10	162.0	---
2.	1.5 wt% PuO_2 - UO_2 (depl.) (Ref. 3) 0.71 in. hex. lattice	71 $\pm$ 6 (75 $\pm$ 8) ^b	75.6	---
3.	UO_2 -2 wt% PuO_2 (8% ^{240}Pu) 0.85 in. hex. lattice (Ref. 2)	128 $\pm$ 8	128.0	115.0
4.	UO_2 -2 wt% PuO_2 (^{240}Pu) 0.85 in. hex. lattice (Ref. 2)	133 $\pm$ 8	139.7	129.6
5.	UO_2 -2 wt% PuO_2 (8% ^{240}Pu) 0.75 in. sq. lattice (Ref. 6)	135 $\pm$ 1 ^a	144.7	---
6.	A1-1.8 wt% Pu (Ref. 4) 0.80 in. hex lattice	33 $\pm$ 3 (26 $\pm$ 3) ^b	34.8	--

a. *Deduced from an extrapolation of subcritical measurements of α .*

b. *Deduced from measured flux decay data obtained after insertion of negative reactivity.*

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4.0 FAST REACTORS

NUCLEAR AND PROCESS TECHNOLOGY

R. E. Heineman

1. Nuclear Analysis

W. W. Little

a. Fuel Burnup

R. B. Kidman

This summary gives a brief description of a possible fuel management scheme for the Fast Test Reactor (FTR).

Unless otherwise specified, all 2-D computations were performed using the 2DB⁽¹⁾ diffusion code with 4-group cross sections. The cross sections were generated by IDX.⁽²⁾ Since the FTR consists of hexagonal subassemblies, essentially all 2DB calculations were performed in hexagonal geometry.

A schematic cross section of the 2-zoned FTR core and reflector is given in Figure 4.1. The height of each cell is 91.4 cm; the cell flat-to-flat distance is 11.98 cm. Table 4.1 gives the subassembly compositions.

In these fuel burnup calculations, control rod followers are placed in all in-core control rod positions. The peripheral control rod positions are filled with control rod material at 20% the poison strength given in Table 4.1 brings the initial k_{eff} close to 1.00, and insures that the subsequent fuel burnup pattern will be more indicative of an actual case. In order not to mask the reactivity loss

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FIGURE 4.1 Core Map and Reloading Plan for Proposed Fuel Management Scheme

TABLE 4.1. Subassembly Compositions

Material	Inner Driver	Outer Driver	Radial Reflector	Control Rod	Control Follower	Open Test Loop	Closed Test Loop
PuO ₂ (a)	5.7	7.6	--	--	--	4.3	1.1
UO ₂ (b)	25.4	23.5	--	--	--	19.1	4.2
Void	3.9	3.9	--	4.2	--	3.6	0.7
316 SS	26.0	26.0	34.0	31.6	37.0	40.0	46.0
Na	39.0	39.0	34.0	33.6	35.0	33.0	48.0
Ni			32.0		28.0		
B ₄ C (c)				30.6			

- a. Pu is 88 at.% ²³⁹Pu and 12 at.% ²⁴⁰Pu.
b. U is 99.29 wt% ²³⁸U and 0.71 wt% ²³⁵U (natural U).
c. The boron is enriched to 35 at.% ¹⁰B.

an actual case. In order not to mask the reactivity loss caused by fuel burnup, the poison atoms are not burned. The fuel in the open and closed test loops is burned, but the loops are recharged with their original composition at the end of each cycle.

In broad outline, the proposed fuel management scheme consists of recharging, with original fuel, 1/3 of each ring of driver subassemblies at the end of each 100-day power run at 400 MW. Disregarding for the moment the three dummy fuel positions, the detailed recharging scheme is shown by the numbered driver subassemblies in Figure 4.1. The driver positions, labeled 1, 2, and 3, are recharged in sequence after each cycle. This pattern is repeated for further cycles.

The three dummy fuel positions are initially control rod followers. After the first cycle, two of these positions are charged with new fuel (the ones that have a number 1). After the second cycle, the third dummy position is charged with new fuel (the one with a number 2). After the third cycle, the one with a number 3 is recharged with new fuel. For further cycles the three dummy positions are treated as driver positions as is done in the preceding paragraph (the dummy position with a 1/3 designation is henceforth assumed to have just a 3 designation).

The purpose of this maneuvering with the three dummy elements is to reduce the reactivity loss in getting to the equilibrium cycle. Figure 4.2 is a plot of k_{eff} versus time, and it bears out this contention. In previous fuel management schemes, which did not include dummy elements, the reactivity loss to equilibrium was between 0.019 and 0.026 Δk . In the present case, it is only 0.0064 Δk . This method then, practically frees the control system from having to compensate for an additional amount of initial excess reactivity.

FIGURE 4.2 Effective Multiplication Factor During Burnup

Figure 4.3 is a plot of the power peak-to-average ($P/\bar{P}$) versus time. The slight decrease during each cycle occurs because ^{239}Pu is consumed faster at the power peaks. The behavior from cycle-to-cycle is rather complicated, but can be explained by keeping track of the following: (1) How the dummy driver positions are rotated into service; (2) Where the peak flux in the core is located; (3) How the driver fuel elements are reloaded with respect to the peak flux.

Table 4.2 shows the breakdown of the burnup experienced by the fuel discharged from an equilibrium cycle.

Table 4.3 shows the heavy atom inventories at the beginning and end of each cycle. Note that the equilibrium cycle is reached by the fourth cycle.

Once the equilibrium cycle is reached, the δk per cycle is approximately -0.022 (see Figure 4.2). This was obtained from a 2-dimensional model, (hexagonal geometry) and therefore does not account for the fact the ^{239}Pu is burned fastest in the axial center where its worth is highest. Previous 2-dimensional R-Z burnup calculations have shown that the 1-dimensional δk should be multiplied by about 1.05 to properly include an axial variation in burnup.

Previous studies have also shown that when Np is included in the nuclide chain an additional 0.001 loss in δk is needed to account for the ^{239}Pu that is held up in Np . With these changes the actual loss in reactivity per cycle is (-0.022) (1.05) -0.001 = -0.0241 δk . Figure 4.2 shows that there is a 0.0064 δk loss in reaching the equilibrium cycle. Thus the minimum initial excess reactivity needed is $0.0241 + 0.0064 = 0.0305 \delta k$.

FIGURE 4.3 Peak-to-Average Power Ratio During Burnup

TABLE 4.2. *Average Burnup of Discharge Fuel^(a)*

<u>Assemblies Discharged</u>		<u>at% Burnup</u>
Central Subassembly	(1)	5.26
1/3 of First Ring	(1)	5.20
1/3 of Second Ring	(4)	4.90
1/3 of Third Ring	(4)	4.39
1/3 of Fourth Ring	(6)	4.78
1/3 of Fifth Ring	(9)	3.22
1/3 of Dummy Positions	(1)	4.33
Average Burnup of Driver Fuel		4.22
Average Burnup of the three Open Test Loops		1.67
Average Burnup of the six Closed Test Loops		1.50

a. Represents discharge fuel from an equilibrium cycle.

TABLE 4.3. *Material Inventories During Burnup (kg)(a)*

Material		1	2	3	4	5	6	7	8
Pu-242	Before	0	0.00695	0.01 70	0.01 79	0.0172	0.0171	0.0179	0.0172
	After	0.0113	0.0312	0.0476	0.0494	0.0479	0.0478	0.0493	
Pu-241	Before	0	1.08	1.64	1.69	1.65	1.65	1.69	1.65
	After	1.73	2.79	3.33	3.38	3.34	3.34	3.38	
Pu-240	Before	71.9	76.9	79.3	79.4	79.3	79.3	79.4	79.3
	After	76.4	81.2	83.5	83.6	83.5	83.5	83.6	
Pu-239	Before	526.0	529.1	530.4	529.8	530.1	530.1	529.7	530.1
	After	506.7	510.0	511.5	510.8	511.1	511.2	510.8	
U-238	Before	2016.2	2051.4	2068.5	2067.9	2068.6	2068.5	2067.8	2068.6
	After	1990.4	2025.6	2041.2	2041.8	2042.8	2042.5	2041.9	
U-235	Before	14.8	14.3	14.1	14.1	14.1	14.1	14.1	14.1
	After	13.4	13.1	12.9	12.8	12.9	12.9	12.8	
Fission Products		0	24.4	37.1	38.4	37.4	37.5	38.4	37.4
		39.0	63.0	75.4	76.7	75.7	75.8	76.7	

a. Includes driver and test Zoop subassemblies.

Table 4.2 shows that the average burnup of the discharge fuel (excluding the test loops) is 4.22 a/o, which is converted to 42.7 MWd/kg of fuel. If one is striving for a burnup of 45 MWd/kg, the present cycle length will have to be increased from 100 to 106 days.

Some attractive characteristics of the proposed fuel management scheme are: (1) Small reactivity loss to equilibrium; (2) Maintains core properties such as power flattening; (3) Requires only two types of fuel; (4) Reloading one third of each ring lends itself to the core geometry resulting in a simple and uniform reloading scheme; (5) And since there is no reshuffling of the fuel subassemblies, the reloading time is minimal.

Some undesirable traits are: (1) A rather large reactivity loss per cycle; (2) Long cycle time; (3) And a wide range in fuel burnup (3.2 $\rightarrow$ 5.3 a/o).

b. Flux Level in the Subcritical FTR

J. R. Sheff

Flux estimates for four subcritical states have now been calculated and are shown in Figure 4.4 for several radial positions at the axial midplane. Note that the curve for the first shield ring has a different normalization than the others and is lower by a factor of 10 relative to the two other locations shown. The fluxes for the four values of k calculated were connected by straight line segments because they actually represent points on three distinct curves. In particular, the four points are reflector rods partially in and full in, reflector and core zone 2 rods in, and all rods in.

FIGURE 4.4 Subcritical Flux Level Versus Multiplication Constant in FTR

2. Core Analysis Section

P. D. Cohn

a. PuO₂ Particle Size

E. G. Stevens

The previous design requirement for PuO₂ particle sizes <50 microns in any dimension was based upon a physics investigation which compared the negative Doppler feedback characteristics of 50 and 100 micron particles. Fabrication reviews indicate that this requirement places excessive demands on manufacturing.

Based upon a re-evaluation of the data, relaxing the maximum particle size specification to 100 microns is acceptable from the physics safety standpoint.

The recently presented data which indicates that discrete particles tend to disappear through interdiffusion upon firing to 1600 °C supports this approach. Furthermore the effects of local hot spots caused by agglomerates up to 300 μ sizes do not appear to present significant problems during steady-state operation, based upon reported experimental data in thermal reactors.^{1,2} Transient effects have not been reported.

It was recommended that the particle size requirement in the Conceptual Component Design Description for the First Fuel Assembly be revised to read: "The fuel pin shall contain as-fabricated fuel with PuO₂ particle concentration regions less than 100 microns in any dimension." The fuel procurement

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specification on homogeneity will clarify this by placing a confidence limit on the degree to which this specification can be achieved and verified in production runs, with a 300 μ upper limit for the infrequent agglomerates.

b. Pressure Drop Calculations for Driver Fuel Subassemblies

D. L. Koreis

Pressure drop calculations were made for driver fuel subassemblies considering fuel pin bundles with 0.050 and 0.056-in. diameter wire wrap. Calculations were made comparing pressure drops in fuel pin bundles with grid spacers to fuel pin bundles with wire wrap.

These pressure drops are shown in graphic form as a function of the mass flow rate-Figures 4.5 to 4.9 for (1) the fuel pin bundle, (2) the coolant entering and leaving the driver fuel subassembly, (3) the instrument package and (4) other unaccounted for losses in the subassembly.

Figure 4.8 shows pressure drops in the instrument package. These values appear to be low. This is the result of the crude methods of calculating pressure drops across the vortex generator and straightening vains are available.

Pressure drops across a fuel bundle with 0.05-in. diameter wire wrap (Figures 4.5 and 4.6) were 21% higher than pressure drops across a fuel bundle with a 0.056-in. diameter wire wrap. The distance across the flats of the duct for a fuel bundle with a 0.050-in. diameter wire wrap was 4.2398 inches whereas 4.335 inches was used with a 0.056-in. diameter wire wrap in the fuel bundle.

FIGURE 4.5. *Fuel Bundle ΔP , as a Function of the Mass Flow Rate with Wire Wrap Diameter = 0.056 in.*

FIGURE 4.6. *Fuel Bundle ΔP , as a Function of the Mass Flow Rate with Wire Wrap Diameter = 0.05 in.*

FIGURE 4.8. Instrument Package ΔP , as a Function of the Mass Flow Rate

FIGURE 4.9. *Other Losses in the Driver Fuel Subassembly
as a Function of the Mass Flow Rate*

Pressure drops in fuel bundles with grid spacers were calculated and compared to pressure drops in fuel bundles with wire wrap for the same diameter fuel pin. The results are plotted in Figure 4.10.

c. Hot Channel Factors

E. G. Stevens

The magnitudes of the individual contributors to the Hot Channel Factors used in calculating the maximum expected fuel temperatures have been mutually agreed upon by BNW and the RPD for proceeding into preliminary design. They are divided into direct and statistical contributors, as shown in Table 4.4.

Temperatures within the core are calculated, considering the Hot Channel Factors, from an equation of the form:

$$T = T_o + \sum_{i=1}^5 a_i \Delta T_i + \sum_{i=1}^5 a_i \Delta T_i \sum_{j=6}^{13} X_{ij},$$

where T = temperature of interest
 T_o = inlet temperature
 a_i = product of direct contributors in region i
 ΔT_i = nominal temperature in region i
 X_{ij} = statistical contributor j in region i

The X_{ij} values are considered to have a joint normal distribution with parameters:

$$E(X_{ij}) = 0$$

$$V(X_{ij}) = (\sigma_{ij})^2$$

$$\text{Cov}(X_{ij}, X_{ij'}) = \rho_{ij,ij'} \sigma_{ij} \sigma_{ij'} \quad (\text{where } \rho = 1 \text{ for dependence and zero for independence among the statistical contributors})$$

FIGURE 4.10. Pin Bundle ΔP as a Function of the Number of Grid Spacers

TABLE 4.4. *Hot Channel Factors*

Radial Factor = 1.40
 Axial Factor = 1.24
 Overpower Factor = 1.20

CONTRIBUTOR

A.	<u>DIRECT</u>	<u>COOLANT</u>	<u>FILM</u>	<u>CLADDING</u>	<u>GAP</u>	<u>FUEL</u>
j = 1.	INLET FLOW MALDISTRIBUTION	1.05	1.02			
2.	INTRA-SUBASS'Y. FLOW MALDIST.	1.15	1.06			
3.	INTER-CHANNEL COOLANT MIXING	0.99	-			
4.	POWER CONTROL BAND	1.02 *	1.02 *	1.02 *	1.02 *	(1.011) * †
5.	WIRE WRAP TEMP. PEAKING		2.00		-	-
	a _i = Product	1.219	2.206	1.02	1.02	(1.011) †
B.	<u>STATISTICAL</u> (3 a limit)					
j = 6.	FISSILE FUEL MALDISTRIBUTION	±2.3%	±3.3%	±3.3%	±3.3%	(±2.39) †
7.	POWER LEVEL MEASUREMENT	9.0	9.0	9.0	9.0	(6.48) †
8.	BURNED FUEL RELOAD	1.2	1.2	1.2	1.2	(0.84) †
9.	ROD DIAMETER, PITCH AND BOW*	2.0	0.9	-	-	-
10.	FILM		33.0		-	-
11.	GAP COEFFICIENT		-		40	-
12.	FUEL			12	-	-
13.	CLADDING CONDUCTIVITY AND TKNS		-		-	(7.22) †
	STATISTICAL COMBINATION	±9.58%	±34.40%	±15.41%	±41.15%	(±10.03%)

* Value goes to 1.0 at overpower

** Bow not considered yet

t Equivalent AT fuel contributor at 12.2 kW/ft as used in calculations

The temperature T is taken to be a normally distributed random variable with parameters:

$$\begin{aligned}
 E(T) &= T_o + \sum_{i=1}^5 a_i \Delta T_i \\
 V(T) &= \sum_{i=1}^5 a_i^2 \Delta T_i^2 \sum_{j=6}^{13} V(X_{ij}) \\
 &\quad + 2 \sum_{i < i'} a_i a_{i'} \Delta T_i \Delta T_{i'} \text{Cov}(X_{ij}, X_{i'j'}) \quad .
 \end{aligned}$$

The probable temperature is $E(T)$ and the maximum expected temperature is $E(T) + V(T)^{1/2}$. Figure 4.11 shows the maximum expected fuel and cladding temperatures as a function of pin power for the FTR core, Concept 5A.

Details on the Eot Channel Factor analysis method are presented in the Conceptual Component Design Description for the First Core Fuel Assembly (CCDD-35) and in a preliminary report entitled, Fuel Fin and Subassembly Conceptual Design Methods and Data.

FIGURE 4.11 Peak Hot Channel Fuel (at overpower) and Clad (steady-state) Temperatures Versus Peak Pin Linear Power

5.0 CRITICAL MASS PHYSICS

CRITICALITY OF POISONED PuO_2 FUELED ASSEMBLIES

S. R. Bierman

A series of critical experiments were performed with unmoderated PuO_2 fuel⁽¹⁾ in support of the Gas Cooled Fast Breeder Reactor (GCFR) Program. These experiments are being performed for Gulf General Atomic, and are designed to check basic neutronic data and methods of calculation employed in the design analysis of the GCFR. Measurements, similar to those previously made^(2,3) in 5 H/Pu moderated PuO_2 fuel, were made to determine the material worth of the thermal and resonance poison control materials, dysprosium, gadolinium, and hafnium. These materials are in the form of oxide powders which, for the purpose of obtaining simple geometric experimental data, have been compacted in stainless steel containers whose dimensions are $6 \times 6 \times 1/4$ in. The experimental data obtained to date are summarized in the Table 5.1.

Additional experiments are planned following preliminary calculational analyses by Gulf General Atomic.

References

1. S. R. Bierman. **See** "Criticality of Unmoderated Plutonium" (next section of this quarterly).
2. L. E. Hansen, R. C. Lloyd, and S. R. Bierman. "Criticality of Poisoned PuO_2 Fueled Assemblies," Reactor Physics Quarterly Report January, February, March, 1967, BNWL-472. Battelle-Northwest, Richland, Washington. May 1969.
3. A. M. Baxter, E. D. Clayton, and L. E. Hansen. "Studies of Critical Assemblies of Homogeneous Mixtures of PuO_2 and Polystyrene," Trans. Am. Nucl. Soc., vol. 20, p. 536. 1967.

TABLE 5.1. *Critical Data of Poisoned PuO₂ Fueled Assemblies*

Description of Critical Assembly	Driver Region ^(a)			PuO ₂ Region ^{(a)(b)}		
	Base, cm	Height, cm	Pu Mass kg	Base, cm	Height, cm	Pu Mass kg
7-2/3 layers driver fuel followed by 1 layer PuO ₂ 1-1/32 in. Al plate and 1.28 layers PuO ₂	31.26 × 30.96	29.96	63.6	33.27 × 33.27	9.44	47.9
8-1/3 layers driver fuel followed by 1 layer PuO ₂ , 1 layer Dy ₂ O ₃ and 1.15 layers PuO ₂	31.26 × 30.96	32.44	69.3	33.27 × 33.27	8.84	45.0
8-1/3 layers driver fuel followed by 1 layer PuO ₂ , 1 layer Gd ₂ O ₃ and 1.15 layers PuO ₂	31.26 × 30.96	32.44	69.3	33.27 × 33.27	8.84	45.0
8-1/3 layers driver fuel followed by 1 layer PuO ₂ , 1 layer HfO and 1.16 layers PuO ₂	31.26 × 30.96	32.44	69.3	33.27 × 33.27	8.87	45.2
8-1/3 layers driver fuel plus 1.92 layers PuO ₂	31.26 × 30.96	32.44	69.3	33.27 × 33.27	7.83	40.2

(a) *Data corrected to a central core temperature of 60 °C.*

(b) *Fuel dimensions without poison plates.*

CRITICALITY OF UNDERMODERATED PLUTONIUM

S. R. Bierman

The continuous flow process in plutonium processing involves, first, the precipitation of plutonium oxalate from the plutonium nitrate solution, second, conversion of the plutonium oxalate to plutonium oxide (PuO_2) through calcination, and third, its conversion to plutonium tetrafluoride (PuF_4) prior to the reduction to metal. Thus, the plutonium goes through all stages of moderation. Experimental critical data are currently available for undermoderated plutonium and plutonium oxide at atomic H/Pu ratios of 5 and 15. Data are also available for unmoderated plutonium metal, but there is no experimental data for unmoderated PuO_2 . Consequently, all criticality parameters for unmoderated PuO_2 have been computed without benefit of direct experimental data for confirmation. Even density corrections to the metal data cannot be used to predict criticality for a reduced density PuO_2 core because of the effect of the oxygen.

To provide this needed data, a series of criticality measurements were recently completed at the Critical Mass Laboratory with unmoderated PuO_2 . These experiments were a part of the research effort at the laboratory to determine the critical parameters of undermoderated plutonium systems. The measurements used the Remote Split-Table Machine and involved determining the critical dimensions and masses for reflected rectangular parallelepipeds, and for bare assemblies within the limits of the fuel available. The density of plutonium in the PuO_2 compacts composing the fuel cores was 5.8 g/cm^3 . The plutonium contained 18.4 wt% ^{240}Pu and 5 wt% ^{241}Pu . The PuO_2 itself was nearly hydrogen-free (0.014 wt% water), but due to the hydrogen contained in the polyolefin plastic cladding material on the PuO_2 fuel cubes, the H/Pu ratio over the stacked

fuel cores was about 0.6. Because of the fuel cladding, the stacked core density was reduced to 4.6 g Pu/cm^3 . Use of the criticality data from the different core sizes permits derivation of the critical cube size and infinite slab thickness. These data, together with theory-experiment correlations and buckling conversions, will also permit derivation of other basic geometries such as the critical diameters for the sphere and infinite cylinder. From previous experiments, the critical mass for a reflected cube is expected to be about equal that of a reflected sphere and may, in fact, be slightly less.

As a consequence of the high specific heat generation from radioactive decay (about 20 W/liter of core) in the PuO_2 , the temperature of the core varied by as much as 30°C during the course of an experiment. A technique developed by the British at Aldermaston was used to normalize the experimental results to a constant temperature of 50°C .⁽¹⁾ Critical data for the Lucite-reflected cores of different cross sectional dimensions are presented in Table 5.2.

TABLE 5.2. Criticality Data for Lucite Reflected Assemblies of Unmoderated PuO_2

<u>Critical Dimensions, cm</u>			<u>Critical Mass, kg Pu</u>
<u>Length</u>	<u>Width</u>	<u>Height</u>	
27.72	27.72	11.03	38.3
27.72	33.27	9.78	41.1
44.36	44.36	6.60	58.2
44.36	33.27	7.72	50.2
33.27	33.27	8.54	43.7

An insufficient quantity of PuO_2 fuel was available to obtain criticality with an unreflected cube of the above material. To obtain data for analysis of an unreflected assembly, a two region parallelepiped was built (making use of other fuel on hand), of which one region contained a

significant amount of moderation. This region of the fuel core was composed of PuO_2 -polystyrene compacts with an H/Pu ratio of 5 (Pu concentration 2.3 g Pu/cm^3 in which the ^{240}Pu isotopic content was 11.5 wt%). Critical data for the stacked two-region cores of slightly different cross sectional dimensions, both at 60°C , are given in Table 5.3.

TABLE 5.3. Criticality Data for Unreflected Two-Region Assemblies of Unmoderated PuO_2 and 5 H/Pu PuO_2 -Polystyrene

PuO_2 -Polystyrene Fuel Region				PuO_2 Fuel Region			
Critical Dimensions, cm			Critical Mass, kg	Critical Dimensions, cm			Critical Mass, kg
Length	Width	Height		Length	Width	Height	
31.26	30.96	27.16	57.9	33.27	33.27	9.76	49.3
31.26	30.96	32.44	69.3	33.27	33.27	7.83	40.2

Experiments were also performed to evaluate the effect of fuel cladding on the buckling and critical mass. The results indicate the cladding has only a small effect on the measured critical dimensions.

Based on the results of the corrected data, a preliminary value for the critical thickness of a Lucite-reflected infinite slab of PuO_2 at a Pu density of 5.8 g/cm^3 is 4 cm. The critical dimension for a reflected cube is derived as about 19.1 cm, with the critical mass being about 31.7 kg Pu.

Theoretical analysis and interpretation of these experiments, which provide the only critical experiment data on unmoderated cores of PuO_2 , are under way.

References

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CRITICALITY OF MIXED PuO_2 - UO_2 SYSTEMS

L. E. Hansen, S. R. Bierman, and E. D. Clayton

Associated with the development of Liquid Metal Fast Breeder Reactors will be the need to fabricate, process, and recover large quantities of mixed PuO_2 - UO_2 fuels. To provide guidance for setting nuclear criticality safety specifications for fuels typical of those to be encountered in the LMFBR Program, and to provide data necessary for planning critical experiments using homogeneous PuO_2 - UO_2 fuels, survey calculations were performed for systems containing 8, 15, 30, 50, and 100 wt% PuO_2 . The calculations were performed for water-reflected and unreflected spherical systems containing $^{239}\text{PuO}_2$, and for water-reflected spheres containing PuO_2 with 15 isotopic percent ^{240}Pu . The moderation ratios considered extend from dry, unmoderated, oxide powders to very dilute, over-moderated, thermal systems,

Figures 5.1 and 5.2 show the unreflected and water-reflected spherical critical radii for the $^{239}\text{PuO}_2$ - $\text{U}_{(\text{Nat})}\text{O}_2$ - H_2O mixtures examined. The critical dimensions were computed using the HFN multigroup diffusion theory code and the DTF-IV multigroup transport theory code, with multigroup constants obtained from the GAMTEC-II code. As can be seen in these figures, the critical size of $^{239}\text{PuO}_2$ - $\text{U}_{(\text{Nat})}\text{O}_2$ mixtures remains nearly constant for systems containing 30 wt% PuO_2 , for fuel concentrations of more than 300 g (PuO_2 - UO_2) per liter. For PuO_2 contents of less than about 30 wt%, minimum volumes exist for the mixtures. For PuO_2 contents of greater than about 30 wt%, the critical volume always decreases with an increase in the total oxide concentration.

The critical radii of mixed PuO_2 - UO_2 spheres with 15 isotopic percent ^{240}Pu are given in Figure 5.3. The effect of

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FIGURE 5.1. Criticality of Mixed $^{239}\text{PuO}_2\text{-UO}_2$ -Water Unreflected Spheres

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FIGURE 5.2. Criticality of Mixed $\text{PuO}_2 - \text{U}_{\text{Nat}}\text{O}_2$ -Water Reflected Sph

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FIGURE 5.3. Criticality of Mixed $\text{PuO}_2\text{-UO}_2\text{-Water}$ Spheres with 15 Isotopic Percent ^{240}Pu

^{240}Pu that results in equal critical volumes occurring at three different concentrations has been previously observed for Pu systems,⁽¹⁾ and is also clearly evident for mixed $\text{PuO}_2\text{-UO}_2$ systems as well. It should also be noted that this triple point of criticality occurs at approximately equal plutonium concentrations regardless of the amount of uranium present in the oxide mixture.

The mass of PuO_2 required for criticality for systems corresponding to those that were previously mentioned are given in Figures 5.4, 5.5, and 5.6. It is interesting to note that a subcritical vessel containing $^{239}\text{PuO}_2$ and water at $1.0 \text{ g PuO}_2/\text{cm}^3$, for example, can achieve criticality by adding a natural UO_2 -water mixture at $1.0 \text{ g UO}_2/\text{cm}^3$. This occurs because the positive reactivity worth of increasing the H/Pu ratio is greater than the negative reactivity worth of adding natural UO_2 to the system.

Since few experimental criticality data are available for $\text{PuO}_2\text{-UO}_2$ mixtures, a series of experiments using depleted uranium layers interspersed within PuO_2 -polystyrene systems at H/Pu ratios of 5 and 15 were performed to provide an interim check on the accuracy of these calculations until additional experimental data are available. The results of those experiments have been previously reported.⁽²⁾ Multiplication factors for those experiments were computed using the GEM-I11 Monte Carlo code and have also been reported.⁽²⁾ The computed values were quite consistent, and relatively accurate, for all the various heterogeneous arrangements of depleted U and PuO_2 -polystyrene as well as for the various Pu/U + Pu atom ratios examined. Water-reflected spherical critical radii were computed for homogeneous mixtures of $^{239}\text{PuO}_2\text{-U}_{(\text{Nat})}\text{O}_2$ aqueous mixtures at Pu/U + Pu and H/Pu atom ratios that were equal to those obtained experimentally. The GEM-I11 code was then used

to compute multiplication factors for those radii computed to be critical using the DTF-IV transport theory code. The results of those calculations are given in Table 5.4.

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FIGURE 5.4. *Unreflected Spherical Critical Mass of $^{239}\text{PuO}_2$ - $\text{U}_{\text{Nat}}\text{O}_2$ -Water Mixtures*

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FIGURE 5.5. Reflected Spherical Critical Mass of $^{239}\text{PuO}_2\text{-U}_{\text{Nat}}\text{O}_2\text{-Water Mixtures}$

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FIGURE 5.6. Reflected Spherical Critical Mass of PuO_2 - $\text{U}_{\text{Nat}}\text{O}_2$ -Water Mixtures with 15 Isotopic Percent ^{240}Pu

TABLE 5.4. *Criticality Calculations for Mixed
 $^{239}\text{PuO}_2\text{-U}_{(\text{Nat})}\text{O}_2\text{-H}_2\text{O}$ Systems*

<u>Water Reflected Material Examined</u>	<u>DTF Computed Critical Radius, cm</u>	<u>GEM-I11 Computed Multiplication Factor</u>
$^{239}\text{PuO}_2\text{-H}_2\text{O}$, H/Pu = 15	10.74 ^(a)	0.9893 ± 0.0080
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.361	13.29	1.0093 ± 0.0088
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.412	12.91	0.9889 ± 0.0084
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.462	12.59	1.0097 ± 0.0083
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.541	12.23	1.0144 ± 0.0090
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.553	12.19	1.0131 ± 0.0087
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 15 Pu/U + Pu = 0.700	11.68	1.0182 ± 0.0097
$^{239}\text{PuO}_2\text{-H}_2\text{O}$, H/Pu = 5	9.35 ^(b)	0.9881 ± 0.0085
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 5 Pu/U + Pu = 0.465	11.58	0.9578 ± 0.0084
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 5 Pu/U + Pu = 0.554	11.01	0.9794 ± 0.0084
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 5 Pu/U + Pu = 0.631	10.63	0.9627 ± 0.0079
$^{239}\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$, H/Pu = 5 Pu/U + Pu = 0.748	10.18	0.9751 ± 0.0092

(a) *The critical radius derived from experimental data reported by Richey, et al.,⁽³⁾ is 10.84 cm.*

(b) *The critical radius reported by Bierman, et al.,⁽⁴⁾ is 9.34 cm.*

These data indicate that the DTF-IV computed critical radii may be slightly conservative from a criticality safety point of view for systems at an H/Pu ratio of 5, and slightly non-conservative at an H/Pu ratio of 15.

Minimum critical masses for water-reflected spheres of mixtures containing 8, 15, 30, 50, and 100 wt% PuO₂ were also examined, and the results are summarized in Table 5.5. These calculations show that the effect of 15 wt% ²⁴⁰Pu on the minimum critical mass of ²³⁹Pu is greater than the effect caused by mixing the plutonium with natural uranium even with the uranium content of the mixture as high as 92 wt%.

TABLE 5.5.

PuO ₂ Content of Mixture, wt%	Mass of PuO ₂ Required in ²³⁹ PuO ₂ -U(Nat)O ₂ -H ₂ O Mixture, kg	Mass of PuO ₂ Containing 5% ²⁴⁰ Pu Required in PuO ₂ - U(Nat)O ₂ -H ₂ O Mixture
100	0.6	1.05
50	0.64	1.15
30	0.67	1.22
15	0.75	1.40
8	0.84	1.65

Although there is a definite need for homogeneous criticality data to confirm the computed data reported herein, there is sufficient supporting criticality data that will permit these calculations to be used for guidance in establishing nuclear criticality safety specifications.

References

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SURFACE MULTIPLICATION

R. E. Wilson, R. J. Hall and E. D. Clayton

In 1961, E. R. Woodcock⁽¹⁾ reported on a new set of criticality parameters: surface multiplication and reflection. The parameters were chosen from consideration of the balance of neutrons crossing a closed surface, which separates the core from the reflector, in an arbitrary reflected assembly of fissionable material. The surface multiplication (M_s) of the core may be defined as the ratio of the rate at which neutrons cross from the core to the reflector to the rate at which neutrons cross from the reflector to the core. Simply stated, this means that for each neutron that enters the core, M_s neutrons, on the average, leave. Reflection (R) may be defined as the ratio of the rate at which neutrons cross from the reflector to the core to the rate at which neutrons cross from the core to the reflector. Monte Carlo calculations have shown that the value of M_s is primarily a function of the core and depends much less on the reflector, whereas the value of R is predominantly a function of the reflector. The calculations also showed that R and $1/M_s$ vary smoothly and nearly linearly as the system passes through critical. For a reflected system the criticality condition is $M_s R = 1$. For a subcritical

system, $M_s R$ is less than 1; for a supercritical system $M_s R$ is greater than 1. These properties of M_s and R unveil the interesting possibility of experimental determinations of critical size without the necessity of making a close approach to critical. By use of neutron sources, separate measurements can be made of the neutron return from a reflector and of the surface multiplication of a bare core. Combining the results of these individual measurements provides data on the criticality condition of the reflected system. For example, if the neutron return from the reflector is 20%, then the bare core would only have to contain approximately 50% of a critical mass in order for the reflected system, if assembled, to be critical.

The GEM (Monte Carlo) code⁽²⁾ calculates M_s and R as the most basic parameters and defines the more common criticality parameter, k_{eff} , in terms of these. The surface multiplication and reflection parameters are being used routinely, and quite successfully, in a variety of analytical studies of criticality at the Pacific Northwest Laboratory, in England, and elsewhere. However, no experimental measurements of surface multiplication have been reported in the literature. Monte Carlo calculations^(1,3) have shown that (at least for fast-spectrum equilateral systems but possibly more generally) $1/M_s$ varies approximately linearly with linear dimension from zero size through the point of criticality. This property, if experimentally verified, would be useful for initial determination of the unreflected critical mass of isotopes such as ^{238}Pu , ^{237}Np , ^{244}Cm , ^{241}Am , etc. The radiation level and heat generation from radioactive decay, as well as the limited availability for some of these isotopes, may rule out the more conventional techniques as impractical. Recent attempts have been made by the authors to measure the surface multiplication

of bare cubical cores of plutonium-bearing fuel. For the special case of a bare system, the criticality condition is $1/M_s = 0$, where use is made of the generalized definition of M_s as the average number of neutrons that leave the core per neutron entering the core across its boundary.

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 plutonium-polystyrene had an H/Pu ratio of 15 and the isotopic composition of the plutonium was 97.8% ^{239}Pu and 2.2% ^{240}Pu . The beam was taken from an intermediate spectrum critical assembly through a 2 in. square hole in a shadow shield. Measurements were made on 2-, 4-, 6- and 8-in. cubes. Two independent measurements were made on each cube. Values of $1/M_s$, determined from the measurements, are shown in Figure 5.7. The critical cube size (12.2 in) was known from previous experimental measurements. Thus, the measurements included samples constituting fractions of a critical mass ranging from only 0.44 to 28% of criticality.

An analytical mock-up of the experimental system was made and the GEM code was used to calculate M_s for a series of plutonium-polystyrene cubes with the composition given above. The results of calculations using two simulated beam sources, whose spectra are assumed to bracket the experimental beam spectrum, are also shown in Figure 5.7.

The reaction rates due to the beam flux and to the total flux leaving all faces of the cubes were measured with indium-covered indium foils. The 54 min half life ^{116}In was counted to determine the indium reaction rate. Indium foils are far from ideal for this measurement; however, available equipment and low fluence restrictions dictated their use. The rate at which neutrons cross the cube boundary was obtained by weighing

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FIGURE 5.7. Inverse Surface Multiplication

the indium reaction rate with the calculated leakage spectrum from each cube face and the effective indium absorption cross sections. The angular distribution of neutrons, cover foil shielding, and self-shielding were considered in obtaining the effective cross sections. The surface multiplication was determined from the ratio of the rate at which neutrons leave the cube to the rate at which neutrons from the beam strike the cube.

To check repeatability, two independent measurements were made for each cube size. The error limits shown for the experimental points represent what we feel are maximum errors. The errors 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 agree with the GEM-calculated curves.

These initial experiments, although of limited accuracy, have established the feasibility of experimental measurements of surface multiplication. Additional experiments are planned in which a ^{252}Cf spontaneous fission neutron source, rather than the beam from a critical assembly, will be used. The indium foils will be replaced by a large flat-plate fission chamber incorporating interchangeable threshold fission foils. These improvements in technique will greatly improve the accuracy of the measurement and will permit a more exact check of calculational techniques.

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IMPACT OF CRITICALITY SAFETY ON THE FABRICATION OF LMFBR FUEL

C. L. Brown

Introduction

Within the next 5 years, the LMFBR program will require fabrication of about 10 tons of plutonium into $\text{PuO}_2\text{-UO}_2$ fuel elements: 6 tons for the Fast Flux Test Facility, and 4 tons for 2 or 3 fast reactor demonstration facilities. Battelle-Northwest has already fabricated over 160 kg of 25 wt% PuO_2 mixed oxide fuel for fast reactor tests. Commercial fuel suppliers are now evaluating FBR fuel fabrication methods. Fabrication technology is rapidly being developed.

An important variable to be determined during the LMFBR Program is the competitive unit cost for fabricating FBR fuel. Of the several factors that will affect fabrication costs, criticality safety is one of the most important. The question is: Will limits imposed by criticality safety on unit mass, equipment size, etc., be the deciding factor in dictating capital equipment costs, manpower requirements, and production rates?

The purpose of this presentation is to provide state-of-the art information to help answer this question. Basic critical parameters for LMFBR fuel have been calculated, and we now have a rather good insight as to the criticality safety limitations that will be imposed at each stage of fuel fabrication.

Comparison of FBR, BWR, and PWR Fuels

The effect of criticality safety on LMFBR fuel fabrication rate can best be shown by comparison to limits currently imposed on light water reactor UO_2 fuel fabrication. As a point of reference, it was decided to compare the fuel requirements for an FBR, BWR, and PWR reactor, each rated at about 400 MW_t . The reactors selected were the FTR (FFTF), Dresden-I, and Yankee Rowe. Basic fuel parameters for these reactors are compared in Table 5.6. The comparison shows a few rather interesting facts:

- (1) The amount of fissionable material in the BWR and PWR cores is nearly the same or higher than in the FTR core to about 700 kg!
- (2) The number of fuel pins in the FTR is higher than in the BWR, but lower than in the PWR.
- (3) The total amount of fuel that must be fabricated for an FTR core is 7 to 14 times lower than for a BWR or PWR core.
- (4) The FTR pin size is much smaller than that of BWR and PWR pins.
- (5) The amount of fuel per FTR pin and FTR subassembly is much less than for BWR and PWR, but fissile content is about the same.

Relating these observations to comparison of criticality limits, one can now say that we will not be too far off to compare criticality safety limits directly. As a matter of fact, it appears that the LMFBR fuel fabrication line will process about 10 times less mixed oxide powder per core than a comparable LWR fuel line.

Criticality Parameters for LMFBR Fuel

Estimated criticality parameters for 25 wt% and 31 wt% PuO_2 - U(NAT)O_2 solutions, powders, and pellets are given in Figures 5.8-5.11. The 31 wt% fuel is included, as it represents

TABLE 5.6. *Comparison of Fuel Requirements
(400 MW_t FTR, BWR, and PWR)*

	<u>FTR</u>	<u>BWR</u>	<u>PWR</u>
<u>Fuel</u>			
Form	PuO ₂ -UO ₂ ⁽¹⁾	UO ₂	UO ₂
Enrichment, wt%	~25	~2.1 ⁽²⁾	3.4
Smear Density, g/cm ³	10.1	9.9	9.9
<u>Pin</u>			
OD, in.	0.230	0.552	0.354
Clad, in.	0.015 (316 SS)	0.032 (Zr)	0.030 (Zr)
Fuel Length, in.	36	144	90
Total Fuel, g	186.9	4,358	989
Total Fissile, g	41.2	80.7	29.6
<u>Pin Bundle</u>			
No. Pins	217	49	304
Array	Hex.	7 x 7	3 6 - 3 x 3
Pitch, in.	0.286	0.738	0.469
Total Fuel, kg	40.6	213.6	300.6
Total Fissile, kg	8.94	3.95	9.01
<u>Reactor Core</u>			
No. Pin Bundles	78	208	76
No. Pins	16,926	10,192	23,104
Total Fuel, kg	3,163	44,420	22,840
Total Fissile, kg	~700	~822	~700

(1) *Depleted Uranium*

(2) *2-Zone Loading*

Neg 06915-6

FIGURE 5.8. Estimated Critical Masses for Homogeneous Water-Moderated Spheres of Pu(METAL) , PuO_2 , and $\text{PuO}_2\text{-UO}_2$

Neg 06915-7

FIGURE 5.9. Estimated Critical Volumes of Homogenous Water-Moderated Spheres of Pu (METAL), PuO_2 , and $\text{PuO}_2\text{-U (NAT)O}_2$

Neg 0691510-8

FIGURE 5.10. Estimated Critical Diameters of Infinite Length Cylinders for Homogeneous Water-Moderated Pu (METAL), PuO_2 , and $\text{PuO}_2\text{-U (NAT)O}_2$

Neg 0691510-6

FIGURE 5.11. Estimated Critical Thicknesses of Infinite Slabs of Homogeneous Water-Moderated Pu (METAL), PuO_2 , and $\text{PuO}_2\text{-U (NAT)O}_2$

the maximum enrichment expected in the FTR driver fuel. These values were calculated using HFN multi-group diffusion theory, DTF transport theory, and Monte Carlo.^(1,2,3) critical parameters for PuO_2 and Pu(metal) systems are included for comparison.^(5,6)

Similar calculations made in the UK⁽⁴⁾ agree exactly at 0.033 g Pu/liter, but are low at 0.8 g Pu/liter and high at theoretical density. The reason for these latter differences is not yet clear, but we have reason to believe that our estimates for unmoderated mixed oxide fuel are slightly conservative.

Not shown in these figures is the effect of ^{240}Pu . Ten percent ^{240}Pu will increase the critical mass about 50% or more, depending on degree of moderation; the D_c and t_c geometry parameters will be about an inch higher.

Fuel Fabrication

The major steps in LMFBR and LWR fuel fabrication are listed in Table 5.7. Coprecipitation and mechanical blending of individual oxides are the two principle methods used to prepare mixed $\text{PuO}_2\text{-UO}_2$.

Fuel elements containing coprecipitated mixed oxide pellet fuel were fabricated by NFS for the SEFOR core. The technology for fabricating fuel elements of either mechanically blended or coprecipitated mixed oxide fuel is being developed further to satisfy the needs of FFTF. This technology will be demonstrated by industry within the next two years as fuel suppliers qualify for FFTF fuel fabrication." Other approaches are thermal denitration and sol-gel methods.

* *BNW is conducting fuel supplier qualification rounds in four phases: Phase 1: analytical round robin; Phase 2: prequalification fabrication of a few hundred pins of mixed oxide; Phase 3: qualification by producing prototype fuel pins using the exact process and equipment that is proposed for FFTF core production; Phase 4: production of the first two FFTF core loadings by November, 1973.*

TABLE 5.7 Fuel Fabrication Operations

Fuel Form	FTR	LWR
Solutions	<ul style="list-style-type: none"> • Co-Precipitation • Sol-Gel 	<ul style="list-style-type: none"> • UNH Sol'n • ADU Precipitation
Slurries	<ul style="list-style-type: none"> • Binder Addition 	
Powders	<ul style="list-style-type: none"> • Blend • Mix • Granulate • Screen 	<ul style="list-style-type: none"> • Granulate • Screen
Pellets	<ul style="list-style-type: none"> • Press • Sinter • Grind 	
Pins	<ul style="list-style-type: none"> • Load Pins • Decontaminate • Weld • Test 	
Assemblies	<ul style="list-style-type: none"> • Assemble Pins • Store 	

Appraisal of Criticality Safety

Critical parameters applicable to LMFBR fuel fabrication are compared to similar parameters for light water reactor fuel in Table 5.8.

TABLE 5.8. *Comparison of Critical Parameters
for FTR and LWR Fuels*

	FTR Fuel [25 wt% PuO ₂ -U(N)O ₂]	LWR Fuel (5% UO ₂)
<u>Solutions</u>		
Mass, kg Fuel	2.8	40
Volume, liters	12	35
%Cylinder Diam, in.	7.2	11
~Slab Thickness, in.	2.7	6.2
<u>Powders (H/X = 0)</u>		
M _C , kg Fuel	340	~(40) ^(a)
V _C , liters	60	~(35)
D _C , in.	12.9	~(11)
t _C , in.	5.5	~(6.2)
<u>Pellets (H/X = 0)</u>		
M _C , kg Fuel	136	~(40)
V _C , liters	12.3	~(35)
D _C , in.	7.5	~(11)
t _C , in.	3.1	~(6.2)
<u>Pins (Water-Moderated)</u>		
Number	122	71
D _C , in.	8.5	9.2
t _C , in.	3.9	4.2
<u>Pin Bundles (H/X = 0)</u>		
Number	9(5)	~(4)
D _C , in.	11 (9)	~(11.6)
t _C , in.	3.5	~(5.5)

a. Numbers in parentheses are for fuel at optimum moderation

Solutions will present the greatest problem. For UO_2 fuel, the critical parameters are high enough to permit rather large solution volumes in safe-by-geometry vessels. Parameters for LMFBR fuel are considerably more restrictive. To achieve the high volumes necessary in PuO_2 - UO_2 solution blending processes, the use of special vessel designs or nuclear poisons appear to be necessary. On the other hand, if credit is taken for fuel burnup and limits can be based on a high ^{240}Pu content, which will be a characteristic of LMFBR Pu, then geometry limits may be high enough to be competitive with limits for LWR fuels without poison.

Because of the potential problems with solutions, there is a strong incentive to use a dry UO_2 - PuO_2 blending process. Surprisingly, powders appear to present no special criticality control problems, provided one can adopt a policy of "no moderation." With UO_2 fuels, moderation control presents a problem. Dry 5 wt% powders are safe in any amount; however, dryness cannot always be guaranteed. In most LWF fabrication plants, UO_2 powders (and pellets) are exposed to potential moderation. So, criticality limits are normally based on the parameters for the water moderated fuels (shown in parentheses in Table 5.8.

Plutonium powders, however, must be perfectly contained for reasons other than criticality. So, dryness can be more readily assured. Of course, one must still account for the leaking or rupture of hydraulic lines and leaking cooling coils. Nonetheless, with proper precautions, criticality should not severely limit mixed oxide powder operations.

Pellet handling is somewhat more restrictive. The philosophies on powder moderation also apply to pellets. Five percent UO_2 pellets are normally processed under limits for moderated fuel. Controls for mixed oxide pellets can be

based on limits for unmoderated fuel. Thus, it appears that controls for mixed oxide pellets will be somewhat more limiting, but should still permit either reasonably large batches or safe-by-geometry operations.

FBR pin handling is not expected to be any more restrictive than LWR pin handling. In both instances, limits must be based on water moderation. The FBR pins are smaller and in a sense more "mobile." However, with proper limits on number and geometry, operations should be as flexible as with LWR pins.

Assembly of fuel pins into the 217-pin bundle will require special control if water moderation is possible. At optimum spacing in water, the MCN (minimum critical number) of pins is estimated to be 122. At the 0.040 in. spacing in the pin bundle, the MCN is about 1500 pins. The technique used to assemble the bundle is to add a row of pins at a time. This technique assures criticality safety under all conditions of accidental moderation.

Once assembled, a 217-pin bundle represents about 0.2 of a critical mass flcoded, or 0.12 of a critical mass dry. Special controls are required in handling FTR fuel bundles, however, because less than 10 dry will support a neutron chain reaction; whereas, dry LWR fuel bundles are safe in any number.

Summary

- (1) LMFBR fuel fabrication will be most limited by criticality safety in operations involving solutions.
- (2) Criticality safety should not seriously limit dry operations involving mixed-oxide powders and pellets.
- (3) Special controls will be necessary when handling sub-assemblies because fewer than ten will support a neutron chain reaction dry.

- (4) From the standpoint of criticality safety, reasonably high, competitive throughput rates should be possible for FER fuel fabrication.

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D. L. Lessor	A. D. Vaughn
W. R. Lewis	E. E. Voiland
R. C. Liikala	L. D. Williams
C. W. Lindenmeier	N. G. Wittenbrock
E. P. Lippincott	W. C. Wolkenhauer
W. W. Little	J. R. Worden
R. C. Lloyd	D. C. Worlton
R. P. Matsen	H. S. Zwibel
D. D. Matsumoto	W. P. Walsh
G. C. Moore	Technical Publications (2)
D. F. Newman	Technical Information
R. E. Nightingale	Files (5)
T. J. Oakes	