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REACTOR PHYSICS QUARTERLY REPORT

OCTOBER, NOVEMBER, DECEMBER 1969

February 1970

AEC RESEARCH &
DEVELOPMENT REPORT

BNWL-1304

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Reactor Technology

REACTOR PHYSICS QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER 1969

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REACTOR PHYSICS QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER 1969

FOREWORD

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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REACTOR PHYSICS QUARTERLY REPORT

October, November, December 1969

1.0 SUMMARYREACTOR THEORY AND CODE DEVELOPMENT

A one-dimensional multigroup diffusion-theory code HYFN has been written in FORTRAN IV for the UNIVAC 1108. A preliminary version of HYFN has been completed and is being thoroughly tested. It is as flexible, is more accurate, and is expected to be faster than HFN.

A new version of the nonlinear least squares fitting program LEARN has been written. Both reliability and speed for difficult problems will be increased using the new version.

THERMAL REACTORS

The analyses of experimental data collected in the AEC-CNEN cooperative program has been completed and a document describing the results has been drafted for publication.

Analytical Monte Carlo studies were made to determine the error expected in the analyses of the Batch Core resulting from assuming a one-dimensional model for a 19-rod cluster fuel element. The results of these calculations show that major errors in neutronic characteristics do not occur using a one-dimensional cell model.

The relative reaction rates of plutonium and uranium have been measured for rods containing a solid solution of $\text{UO}_2\text{-PuO}_2$ and a mixture of 350 μ PuO_2 particles in UO_2 in the PCTR test cavity with a 4 in. graphite lattice. The measured data are in good agreement with the relative reaction rates calculated for the same fuels with the THERMOS-GRANIT Code which uses a special geometry routine to treat finite resonance shielding of particles embedded in cylindrical media by the method of Lane, Nordheim, and Sampson.

A computer program, EDIS, developed and incorporated into the software package for the Programmed Measurement and Control System (PMACS) at HTLTR, provides a significant cost savings by eliminating the need to replace a character generator for the data display system.

An HTLTR subcritical reactivity monitor is to be utilized on an on-line digital computer system to calculate the subcritical reactivity and predict a reactor excess reactivity.

A computational study has been conducted to determine whether the addition of a neutron absorbing material to the graphite inner sleeves in HTLTR horizontal control rods would increase the total control strength of the rods. The results show that the addition of an absorber reduces rather than increases the rod strength.

A technique has been developed to safely add and/or remove reactivity to the HTLTR at any temperature. The development permits the operator to compensate for a large negative temperature coefficient of the core. The larger than expected compensation range makes possible significant improvements in the experimental efficiency of the facility.

The low-energy cross sections of ^{232}Th , ^{233}U , ^{235}U , ^{238}U , and ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu were re-evaluated as part of an effort to provide revised data files for ENDF/B. For the fissile nuclei, the re-evaluated data are consistent with the 2200 m/s values and cross-section shapes recommended in the recently completed 1969 IAEA study.

FAST REACTORS

A large fast-neutron cavity (FNC) assembly in the center of the Physical Constants Testing Reactor (PCTR) has been developed to permit alteration of the neutron spectrum from a typical thermal reactor spectrum to a spectrum quite similar to that in large dilute fast breeder reactors. Critical experiments performed in this fast-thermal coupled reactor core have demonstrated the feasibility of zoned fast reactor experiments in the FNC. Experimental results with these core have proved most accurate and far less expensive than the conduct of full-sized critical experiments. It is concluded that reactivity and neutronics experiments for large dilute fast breeder reactor media can be conducted in the FNC at large cost savings.

A description is given of critical approach and exponential experiments to measure material bucklings, extrapolation distances, and critical masses as a function of lattice spacing for 25 wt% PuO_2 - UO_2 fuel pins in water.

The results of a study of metal swelling effects on FTR core components are given. The purpose of the study was to develop an understanding of the predicted major interaction between FTR components caused by fast neutron induced metal swelling, identify the most severe problem areas, and provide guidance to the RPD for core design.

The results of a neutronics study of the adequacy of peripheral control rod worth measurements in ZPR-III Assembly 56B are discussed. The initial measurements were shown to be in error since they did not properly account for spatial flux distortion.

CRITICAL MASS PHYSICS

In an effort to establish an improved basis for safe handling, and to act as a guide in planning possible future criticality experiments, k_{∞} values, critical masses, and critical radii have been calculated for ^{237}Np , ^{238}Pu , and ^{240}Pu , ^{241}Am , ^{242}Am , and ^{244}Cm . These survey calculations, using the best currently available nuclear data, were made for both bare and reflected and for both metal and slightly moderated spheres. The calculations show that, in the unmoderated state, the critical mass for some of these nuclides is less than that of the common fissile nuclides ^{233}U , ^{235}U , and ^{239}Pu . However, for ^{238}Pu , ^{240}Pu , ^{241}Am , and ^{244}Cm , k_{∞} rapidly becomes less than one upon addition of water to the metal system, and criticality then is no longer possible. This effect is due to the fission thresholds characteristic of these nuclides. Scattering degrades the neutron spectrum and essentially no fissioning occurs when the energy of the neutron is below the threshold. The effect of neutron energy degradation also appears in the reflector savings. It is shown that (for metal cores) steel, which has a relatively small effect on the energy of the returning neutrons, is a comparable or better reflector than water, which returns a higher percentage of thermal or near-thermal neutrons.

2.0 REACTOR THEORY AND CODE DEVELOPMENT

PROGRAM HYFN: A NEW MULTIGROUP DIFFUSION CODE

R. D. Johnson

A one-dimensional multigroup diffusion-theory code HYFN has been written in FORTRAN IV for the UNIVAC 1108. Design of this code is similiar to the HFN code,⁽¹⁾ which has been available for several years.

The derivation⁽²⁾ of computational forms for HFN begins by formulating a finite difference analog to the second-order differential equation of one-dimensional, multigroup diffusion at an internal point of solution. A three-point recurrence relation for the flux is obtained, but the geometric coefficients of this expression require modification to obtain the correct results in specific limiting cases. In final form, these coefficients may assume physically unrealistic values under realistic conditions. A two-term Taylor series expansion is used to relate the flux at and near an external boundary of the system to the boundary condition.

In contrast, the derivation used for HYFN proceeds from the macroscopic neutron balance equation for one-dimensional multigroup diffusion in a spatial cell extending from one point of solution halfway to the next, where a central difference is used to approximate the flux gradient. In this way, the external boundary condition reduces immediately to a two-point relation for the fluxes at the boundary and the first internal point. Application of the flux and current continuity relations across each interface between two internal cells yields a set of three-point recurrence relation for the fluxes. The geometric coefficients appearing in these expressions are uniquely determined and always properly valued.

The procedure used to evaluate the group fluxes from the recursion formulae is the same in both codes and, whenever possible, the logical flow of the calculations have been simplified.

A preliminary version of HYFN has been completed and is being thoroughly tested. It is as flexible, is more accurate, and is expected to be faster than HFN.

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NONLINEAR LEAST-SQUARES DATA ANALYSES

D. A. Kottwitz

A new version of the nonlinear-least-squares fitting program LEARN,⁽¹⁾ called LEARN-I-B has been written and will eventually supercede LEARN-I-A.⁽²⁾ Two important changes have been made in the iterative portion of the program. The overall effect is to increase both reliability and speed for difficult problems. Reliability has been improved by requiring that each iterative step satisfy an accuracy criterion, i.e. a specified level of agreement between the predicted and actual consequences of a step. Speed has been increased by reversing the direction of a matrix step when it rises toward a "saddle" point. In addition, two new options involving the iterative procedure have been added. At execution time, the user may now elect (1) to allow a saddle solution as well as a minimum solution, and/or (2) to omit the calculation of the

second partial derivatives in his THEORY subroutine.
Debugging of the new version is almost complete.

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3.0 THERMAL REACTORS

RE-EVALUATED THERMAL CROSS SECTIONS FOR FISSILE AND FERTILE NUCLEI

B. R. Leonard, Jr.

The low-energy ($\sim < 1$ eV) cross sections of the isotopes ^{232}Th , ^{233}U , ^{235}U , ^{238}U , and ^{239}Pu , ^{240}Pu , ^{241}Pu , ^{242}Pu were re-evaluated as part of the effort of a task force organized by the Cross Section Evaluation Group of the National Neutron Cross Section Center (NNCSC), Brookhaven National Laboratory. The re-evaluated data files were prepared in ENDF/B format and, along with summary documentations, recommended to the NNCSC for inclusion in Version 2 of ENDF/B. For the fissile nuclei, the re-evaluated data are consistent with the 2200 m/s values and cross-section shapes recommended in the recently completed 1969 IAEA study.⁽¹⁾ The cross-section shapes were derived from the shapes given in previous evaluations and considered more recent experimental results where appropriate. The shapes were then modified, subject to the criterion of mathematically smooth first and second derivatives of $\sigma\sqrt{E}$ versus E , until they produced the shape-factor indices (the Westcott "g" factors) essentially identical to the 1969 IAEA recommended values.

For the fertile nuclei, the cross sections were re-evaluated to include more recent experimental results and attempts were made to improve the estimate of the shape by detailed considerations of the assignments of negative-energy levels.

For ^{232}Th , the evaluation considered the results of Lundgren⁽²⁾ who measured the relative capture cross section from 0.1 to 3.4 eV and assigned parameters for a negative energy level. The shape of the cross sections resulted from

a resonance calculation retaining the 2200 m/s capture cross section value⁽³⁾ of 7.4 barns.

For ^{238}U , the recent result of Bigham, et. al.⁽⁴⁾ who derived a 2200 m/s capture cross section value of 2.721 ± 0.016 barns was the primary factor in the chosen value of 2.72 barns. The assignment of negative energy resonance parameters was made by re-evaluating the contribution of the first few positive energy resonances and also by fitting the total cross section data of Bollinger, et. al.⁽⁵⁾ in the eV range.

For ^{240}Pu , the 2200 m/s capture cross section was evaluated to be 290 barns based primarily on the precision results of Lounsbury, et. al.⁽⁶⁾ A negative energy level assignment resulted from a re-evaluation of the parameters of the 1 eV resonance and the average values of the positive energy resonances.

For ^{242}Pu , the 2200 m/s capture cross section was chosen to be 20 barns based primarily on the integral result of Durham and Lounsbury⁽⁷⁾ and the total cross section measurements of Young and Reeder.⁽⁸⁾ The assignment of a negative energy level considering average values of resonance parameters was made.

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SOFTWARE CHARACTER GENERATOR FOR HTLTR

W. C. Roberts, Jr.

A computer program, EDIS, developed and incorporated into the software package for the Programmed Measurement and Control System (PMACS) at HTLTR, provides a significant cost savings by eliminating the need to replace a character generator for the data display system.

A 19-in. cathode ray tube with a capacity of over 400 characters and 3 colors displays information from essentially all of the signal transducers on the reactor systems at HTLTR. In conjunction with the general purpose computer, a PDP-7/9, the display functions as a single point readout by converting the digital information into engineering units and component identification for the operator. The display system is essential to operation of the reactor and the process systems. A nearby standard 5-in. panel-mounted oscilloscope has been interfaced to serve as a backup display unit in event the color display section should fail. However, both displays depend upon a hardware (electronic) character generator of a stroke-type currently capable of generating 38 different kinds of characters or symbols. The existing generator, the first unit of its kind produced by a vendor at a time when this technology was new, functions well. Numerous design deficiencies, however, require considerable maintenance, and a serious failure could adversely affect the schedule of the experimental programs underway. A replacement unit of

current design thus would normally be justified. However, an alternate approach involving software was employed.

The software program eliminates the need for a new character generator at this time and functions by interrogating the computer core locations containing the normal display information and by converting the character code words into a programmed character. The various types of characters, virtually unlimited, are developed by selecting an array of dots within a 5 by 5 lattice. The lattices are displayed in the desired locations on a standard oscilloscope using an existing 34 GW display driver provided with the PDP-7. Each dot generated by the display driver forces the PDP-7 to enter a slow cycle of about 20 μ -sec. This extended time limits the number of characters that can be displayed simultaneously during each program cycle. It has been determined that four lines (up to 84 characters) could be displayed simultaneously on a time-sharing basis without interfering with the other normal functions under program control. It has been further determined that two groups of four lines could be displayed on alternate PMACS cycles and still give good resolution. The operator currently can display any desired eight lines (168 characters) from the normal display of up to 20 lines. This capability is considered adequate for backup.

THE HTLTR SUBCRITICAL REACTIVITY MONITOR

C. E. Haines

A subcritical reactivity monitor is under development at the High Temperature Lattice Reactor (HTLTR). The general purpose digital computer, a part of the instrumentation and control system, has been programmed initially to calculate on-line, the subcritical reactivity by means of the following expression:

$$-\rho = \frac{Sf(P)}{P}$$

where: ρ = subcritical reactivity

P = reactor power level (neutron flux)

S = a source term, namely the output of a positive ion accelerator which the program normalizes.

$f(P)$ = a function of the reactor power and other factors including a temperature correction.

The reactor excess reactivity is calculated by summing the foregoing computed value with the reactivity available from the control and vertical safety rod systems. The rod worths are determined from their positions and previous calibrations. Hardware provides the computer with control rod positions whose uncertainties are less than 0.04 of a cent in reactivity worth. In order to determine vertical safety rod positions, a subroutine was written to record the safety rod travel time and direction. This position indication is accurate to about 1/2 cent in reactivity worth. Initial subcritical values indicate that a correction factor will be required for non-symmetrical rod configurations, and additional experimental data are required to determine the correction.

The accuracy of the excess reactivity calculation is dependent not only on the subcritical reactivity value but on the precise knowledge of vertical and horizontal control rod worths. The latter are known quite accurately since they are measured by asymptotic positive periods. The worth of the vertical safety rods can be measured only while the reactor is subcritical. Heretofore, this worth has been determined by extrapolation of an inverse multiplication curve using the calibrated control rods as a base. Procedural and software changes which have been completed to avoid the extrapolation of the curve will reduce uncertainty in the vertical safety rod worth determination.

The program computes the values and updates once per second. The computed values are displayed for the reactor operator and recorded by the typewritten log. The validity of the calculations are determined by comparison of the predicted reactor excess reactivity with the value measured by an asymptotic positive period. The results of the comparison is recorded by a typewritten log.

Several of the constants stored in the computer memory, such as the rod worths, require updating when changes to reactor fuel loading are made.

The subcritical reactivity monitor is of value during reactor fuel loading, fuel loading modifications, and during heatup and cooldown of the reactor. As experience is gained with use of the subcritical monitor, the remaining software will be completed and used to make initial measurements of the worth of the control and vertical safety rod systems and of the experimental equipment.

HTLTR CONTROL ROD STUDY

W. M. Harris

Introduction

Operating experience at elevated temperatures with the High Temperature Lattice Reactor (HTLTR) has indicated a need for increased control rod strength to compensate for the reactivity losses incurred as the reactor is heated. It has been suggested that a sufficient increase in control rod strength might be obtained by replacing the inner graphite sleeve in the present horizontal control rods (HCR) with a graphite sleeve impregnated with neutron absorbers. This study was undertaken to determine the control rod strength obtainable in the HTLTR using either $^{238}\text{UO}_2$ or different amounts of a $1/v$ absorber added in the inner graphite sleeve.

Analytical Techniques

The initial analytical technique used in the solution of this problem was the development of a two-dimensional, two-energy group diffusion theory calculational model of HTLTR. The model was devised for two cases. The first case studied was that for a fully opened rod position and the second was for a fully closed rod position. The HCR is in the most reactive position when fully opened.

Basic cross section data were taken from the Battelle Master Library. THERMOS/BATTELLE⁽¹⁾ was used to calculate region-dependent, thermal group constants for one-dimensional cells. The DOTSN code was employed for flux weighting of the region-dependent thermal constants to obtain cell-averaged thermal constants for the complicated r,z cell geometry.

The fast group constants were generated in the EGGNIT⁽²⁾ slowing down code. Resonance shielding was treated by the Nordheim numerical integration technique,⁽²⁾ a space-dependent computation using flat-source first-flight collision probabilities. Collision probabilities were determined by a ray-tracing technique utilizing the BATTELLE MONTE CARLO⁽³⁾ geometry routine. First-flight collision probabilities as a function of total cross section were fitted to a polynomial by PROGRAM LEARN.⁽⁴⁾ These polynomials were incorporated into a geometry routine for the resonance shielding calculation.

The cell constants generated via THERMOS-DOTSN and EGGNIT were used in a two-dimensional (r,z) two-energy group calculational model of the HTLTR, utilizing the 2DB⁽⁵⁾ diffusion theory code. This code was used to compute k_{eff} for the HTLTR with the HCR's either open or closed. The calculations were performed only for the case of the HTLTR at room temperature.

Description of HCR Operation

HTLTR contains nine horizontal control rods. One rod is used for making small precise changes in reactivity only. The remaining eight HCR's are most frequently used for reactor operation. The HCR's are shutter-type and produce their full change in reactivity with a total travel of 6 inches. A cross section view of an HCR is shown in Figure 3.1.

The total rod length is 7-1/2 feet, consisting of an assemblage of graphite and UO_2 -graphite cylinders on a 1/2-inch diameter thoria-dispersed nickel (TD-nickel) rod. The UO_2 -graphite cylinders are made of UO_2 -16 at.% ^{235}U dispersed in graphite. These 6-in. long cylinders of graphite and UO_2 -graphite make up the active section of the HCR. Anchored to the reactor shell is a graphite sleeve in which is housed an outer row of graphite and UO_2 -graphite cylinders alternately placed.

Description of Computational Model

Eight isotopes (^{235}U , ^{238}U , N, ^{232}Th , Ni, O, ^{10}B , and C) were selected from the THERMOS LIBRARY to be used in the calculations. Since boron is a good $1/v$ absorber, it was used. However, boron could not be considered as a possible material HTLTR control rods because of its unstable properties at elevated temperatures. The nuclear density of each material in each of the regions, shown in Figure 3.2 is listed in Table 3.1. Cell dimensions are also indicated in Figure 3.2. The entire 30-space point capacity of THERMOS was used for the transport kernel calculation in cylindrical geometry with reflecting boundaries.

The control rod regions were divided into five material zones for the DOTSN calculation. Cross sections obtained from THERMOS, together with the appropriate atom densities, were

FIGURE 3.1. HTLTR Control Rod, a Cross Section View

FIGURE 3.2. HTLTR Control Rod Model for Dotsn Calculation in R,Z Geometry

TABLE 3.1. Atom Densities of Materials in Each Region of the Cell Calculation

<u>Material</u>	<u>^{10}B</u>	<u>^{12}C</u>	<u>^{14}N</u>	<u>^{16}O</u>
TD Nickel	---	---	---	9.794-04
Nitrogen	---		1.000-20	---
Graphite	---	8.277-02	1.000-20	---
Fuel Sleeve	---	4.303-02	---	1.3684-02
^{238}U Sleeve	---	4.303-02	---	1.3684-02
Boron Sleeve	3.600-02	4.303-02	---	
	Nickel	^{232}Th	^{238}U	^{235}U
TD Nickel	8.914-02	4.875-04	---	---
Nitrogen	---	---	---	---
Graphite	---	---	---	---
Fuel Sleeve	---	---	5.739-03	1.103-03
^{238}U Sleeve	---	---	5.739-03	---
Boron Sleeves	---	---	---	---

entered by position, group, and material. The cell-averaged constants were then calculated by DOTSN in a two-dimensional r,z geometry model.

Calculated Results and Conclusions

Tabulated in Table 3.2 are the cell-averaged, two group constants utilized in the 2DB calculations to compute the total worth of the HTLTR control rods as a function of neutron absorber admixed in the inner graphite sleeve. The resulting control worths of the HCR are given in Table 3.3 and are illustrated in Figure 3.3.

TABLE 3.2. Group Constants Used in the 2DB (R-Z) Calculation of HTLTR

Constants	CASE 1				CASE 2			
	$^{235}\text{UO}_2\text{-Graphite Sleeve}$				$^{238}\text{UO}_2\text{-Graphite Sleeve}$			
	Open		Closed		Open		Closed	
	G-1	G-2	G-1	G-2	G-1	G-2	G-1	G-2
Σ_f	0.114-02	0.156-01	0.113-02	0.104-01	0.586-03	0.135-01	0.584-03	0.134-01
Σ_a	0.282-02	0.205-01	0.268-02	0.141-01	0.700-02	0.368-01	0.698-02	0.316-01
$\nu\Sigma_f$	0.280-02	0.379-01	0.276-02	0.253-01	0.145-02	0.327-01	0.144-02	0.325-01
Σ_{tr}	0.290	0.387	0.291	0.389	0.268	0.409	0.269	0.403
Σ_{ij}	0.286	0.366	0.286	0.375	0.288	0.323	0.288	0.372
Σ_{out}	0.718-02	0.0	0.180-02	0.0	0.109-06	0.0	0.111-06	0.0

Constants	CASE 3				CASE 4			
	$^{10}\text{B-Graphite}$				$^{10}\text{B-Graphite}$			
	Open		Closed		Open		Closed	
	G-1	G-2	G-1	G-2	G-1	G-2	G-1	G-2
Σ_f	0.586-03	0.117-01	0.584-03	0.145-01	0.114-02	0.160-01	0.113-02	0.104-01
Σ_a	0.701-02	0.252-01	0.698-02	0.252-01	0.301-02	0.212-01	0.296-02	0.144-01
$\nu\Sigma_f$	0.145-02	0.285-01	0.144-02	0.353-01	0.280-02	0.389-01	0.275-02	0.254-01
Σ_{tr}	0.269	0.391	0.269	0.391	0.293	0.385	0.293	0.386
Σ_{ij}	0.262	0.366	0.262	0.366	0.288	0.364	0.288	0.372
Σ_{out}	0.110-06	0.0	0.112-06	0.0	0.165-02	0.0	0.167-02	0.0

FIGURE 3.3. Total Rod Worth Versus Thermal Neutron Absorption Cross Section in Inner Graphite Cylinder

TABLE 3.3. HCR Control Worths

<u>Case</u>	<u>Material in Inner Graphite Sleeves</u>	<u>Worth, \$</u>
1	(Graphite)($^{235}\text{UO}_2$ -Graphite)	3.21
2	($^{238}\text{UO}_2$ -Graphite)($^{235}\text{UO}_2$ -Graphite)	3.01
3	($^{10}\text{B}_4\text{C}$ -Graphite)($^{235}\text{UO}_2$ -Graphite)	2.45
4	($^{10}\text{B}_4\text{C}$ -Graphite)($^{235}\text{UO}_2$ -Graphite)	1.23

It is evident from these results that, as the absorber concentration in the inner graphite sleeve increases, the net HCR worth (closed-to-open) decreases. The primary reason for this decrease in the HCR worth is that the added absorber in the inner graphite sleeve depresses the thermal neutron flux in the outer fuel sleeve and this flux perturbation is greatest when the HCR's are in the open position (see sketch A). Thus, by the addition of the absorber to the inner graphite sleeve, the neutron production in the outer fuel sleeve is reduced by a greater amount in the open position than in the closed position (see sketch B).

It is concluded from this study that the control worth of the HCR can not be improved by the addition of a neutron absorber to the inner graphite sleeves, and some other approach is thus required to increase the control strength of the HCR's in the HTLTR.

Sketch A

Sketch B

G denotes graphite and A denotes absorber

Open Position

Closed Position

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MANUAL REACTIVITY ADJUSTMENTS IN HTLTR AT TEMPERATURES UP TO 1000 °C

J. G. Holt

Initial experiments at elevated temperatures in HTLTR demonstrated the reactor temperature coefficient of reactivity to be more negative than had been predicted. As a result, to obtain an adequate excess reactivity to conduct an experiment at high temperatures required a fuel loading at room temperatures which, while safe, often approached the subcritical margin maintained by the control and safety system. Further, any given fuel loading was operable for experiments only over a rather narrow temperature span. For example, a fuel loading operable at 700 °C provided too much excess at 550 °C to operate within the limits set in the Operating Specifications and Safety Limits for the reactor; and not enough reactivity to conduct experiments above 900 °C. To complete a series of experiments from room temperature to 1000 °C required several

cycles of evacuating the system, charging with nitrogen, heating, conducting the experiment, and cooling to room temperature to adjust the fuel loading.

A modification to HTLTR has been completed which permits addition of reactivity while the assembly is at elevated temperature and removal of that reactivity prior to cooldown, thus avoiding the time-consuming heatup and cooldown cycles required for normal fuel adjustment operations. Eight reactivity adjustment assemblies combining fuel and poison have been constructed. A typical assembly is shown in Figure 3.4. Each column is made up of two 10-in. driver fuel pieces containing 5% enriched UO_2 and a 60-in. portion containing a 28-in. length of Sm_2O_3 balls. The fuel pellets and the Sm_2O_3 balls are contained in three graphite tubes interconnected by double-ended threaded plugs of graphite. One end of the assembly is connected to a 0.25-in. diam rod of T-D nickel which penetrates the reactor shell at the rear. Stops for positioning and locking devices have been included to secure each column in either of the two positions as shown in Figure 3.4. The columns are located in unused control rod channels for which penetrations in the reactor shell already existed. A gas seal was provided for each push rod at the reactor shell.

All movements of these assemblies are made with the reactor at equilibrium temperature and in Core Load Mode with two vertical safety rods up and cocked, as in normal fuel loading changes. Preliminary measurements at room temperature indicate that a change of at least 25¢ per column can be obtained. Safer reactor operation and a significant savings in experimental time have been gained by adding this capability to adjust the reactivity while at temperature.

FIGURE 3.4. Fuel Poison Column Positions

EXPERIMENT AND CALCULATED RESULT FOR UO_2 AND $\text{UO}_2\text{-PuO}_2$
FUEL H_2O MODERATOR LOADINGS

L. D. Williams, J. B. Edgar, P. Loizzo*, R. Martinelli*
 and N. Pacilio*

A cooperative program between USAEC and Italian Comitato Nazionale per L' Energia Nucleare (CNEN) was conducted at Argonne National Laboratory (ANL) and at Pacific Northwest Laboratory (PNL) under the auspices of the Plutonium Utilization Program. The purpose of this cooperative program was to provide experimental information on the neutronic characteristics of plutonium fuel in systems representing a typical Boiling Water Reactor (BWR). As a part of this program, plutonium fueled experiments were conducted at PNL in the Plutonium Recycle Critical Facility (PRCF). Uranium fueled experiments were included in the program for purposes of comparing core characteristics for uranium and plutonium enriched fueled systems. Interim results from the program have been published.⁽¹⁾ The analysis of all experimental data has been completed and a final report is being issued. A brief summary of the contents of the document is given in this report.

The loadings studied included both single zone (i.e., one fuel) and multizoned criticals. The effects of varying degrees of nonuniformity on core parameters were systematically investigated. The core parameters included:

- k_{eff}
- Kinetic Parameters ($\beta_{\text{eff}}/\lambda$ and Rossi α)
- Flux and Power Distributions
- Moderator Temperature Coefficients
- Control Blade Worths

* CNEN Scientist

Power distributions were deduced from gross counting of fission products. Different fuels were intercalibrated by calorimetric measurements. Rossi alpha measurements were made in critical and subcritical loadings using the "endogenous" pulsed source method.⁽²⁾

Analytical correlations of some of these measured data were based primarily on CNEN developed theoretical models.⁽³⁾ The reliability of the various models and the validity of the approximations used were checked by comparison with experiments and with other more sophisticated calculational schemes. The results of these correlations provide confidence to the analyses of studying plutonium recycle in the GARIGLIANO STATION in Italy.

An extensive analytical correlation of these experiments using the PNL developed methods is currently underway.

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ANALYTICAL STUDIES OF ONE AND TWO-DIMENSIONAL CELL
REPRESENTATIONS FOR CLUSTER GEOMETRY FUEL

D. H. Thomsen

This study was directed towards increasing the understanding of the neutronics of the Batch Core Experiment⁽¹⁻⁴⁾ in the Plutonium Recycle Test Reactor (PRTR). The PRTR fuel is in the form of 19-rod clusters. A two-dimensional calculation of the lattice cell is required to represent all of the rods correctly. However, most reactor physics cell codes are restricted to one dimension. The rods of the cluster are often represented in one dimension as annular fuel rings in the lattice cell calculation. This approximation has potential for introducing some error. The purpose of this study was to determine the magnitude of the errors incurred in using this one-dimensional cell approximation.

The method of solution was to compare the results of two calculations. One calculation represented the exact lattice configuration and the other a one-dimensional cell approximation. The cross sections and calculational methods in both calculations, using the Monte Carlo Code,⁽⁵⁾ were kept constant and only the cell geometry was different. The first calculation used a full two-dimensional representation of the 19-rod cluster and the cell boundary. The second calculation used a one-dimensional representation of the lattice cell.

Calculations were performed for a situation representing the Batch Core and for a natural UO_2 fueled cell. The setup representing Batch Core conditions was a cell containing a nonirradiated Batch Core element, i.e. UO_2 -2 wt% PuO_2 , while the moderator contained 11.25 ppm boron. The second calculational setup was for a natural UO_2 fuel element with unborated moderator. This fuel was chosen because of available experimental data and to allow deducing separately the effects due to uranium and plutonium.

The effects on the lattice cell constants of using the one-dimensional model are shown in Table 3.4. The difference quoted in this table is the one-dimensional value minus the two-dimensional value. These results show a relatively small effect on k_{∞} for both cells. Both values are within one standard deviation, and an upper limit would probably be ± 5 mk. Such a small difference for k_{∞} , however, is due to thermal and nonthermal effects compensating in the mixed oxide cell. The difference in the thermal effects (ηf) are due predominately to the differences in thermal utilization. In addition, the differences in ηf is the only difference significantly outside of statistics.

TABLE 3.4. Comparison of Calculated Cell Parameters for One and Two Dimensional Models of the 19 Rod Cluster

Cell Parameter	Calculation	Difference, %
Multiplication (k_{∞})	$\text{PuO}_2\text{-UO}_2$	-0.002 ± 0.003
	UO_2	0.003 ± 0.004
Thermal Neutron Produced Absorption (ηf)	$\text{PuO}_2\text{-UO}_2$	-0.005 ± 0.002
	UO_2	0.002 ± 0.002
Thermal Utilization (f)	$\text{PuO}_2\text{-UO}_2$	-0.004 ± 0.003
	UO_2	$+0.001 \pm 0.002$
Thermal to Total Absorption (p)	$\text{PuO}_2\text{-UO}_2$	0.001 ± 0.007
	UO_2	0.002 ± 0.004
Total Fission/Thermal Fission	$\text{PuO}_2\text{-UO}_2$	0.003 ± 0.011
	UO_2	0.000 ± 0.010

Parameters which should infer differences in initial burnup reaction rates are compared in Tables 3.5 and 3.6. The ^{238}U and ^{239}Pu capture rates appear to be calculated about 1% low for the UO_2 and $\text{UO}_2\text{-PuO}_2$ cases by the one-dimensional model. This difference would result in underpredicting both the production and destruction of ^{239}Pu so that the concentration during burnup may be fairly close. The error incurred in

the ^{238}U capture rate appears to be due to resonance absorption rates. The ^{235}U reaction rates are calculated very accurately by the one-dimensional model. The ^{240}Pu concentration would probably be underestimated as the ^{239}Pu capture is too low and the ^{240}Pu capture is too high. It is difficult to estimate, from these calculations, the effects on the ^{241}Pu and ^{242}Pu concentrations.

TABLE 3.5. Comparison of Burnup Parameters

<u>Parameter</u>	<u>Calculation</u>	<u>Difference, %</u>
^{238}U Capture	$\text{PuO}_2\text{-UO}_2$	-1.2 ± 0.6
	UO_2	-0.8 ± 0.8
^{238}U Epithermal Capture	$\text{PuO}_2\text{-UO}_2$	-2.6 ± 1.0
	UO_2	-1.6 ± 1.6
^{235}U Capture	$\text{PuO}_2\text{-UO}_2$	-0.5 ± 0.5
	UO_2	-0.2 ± 0.8
^{235}U Fission	$\text{PuO}_2\text{-UO}_2$	-0.16 ± 0.6
	UO_2	-0.2 ± 0.8

TABLE 3.6. Comparison of Burnup Parameters
for the $\text{PuO}_2\text{-UO}_2$ Cell

<u>Parameter</u>	<u>Difference, %</u>
^{239}Pu Capture	-1.1 ± 0.6
^{239}Pu Fission	-0.6 ± 1.0
^{240}Pu Capture	1.9 ± 2.5
^{238}U Capture	-1.2 ± 0.6
^{235}U Capture	-0.5 ± 0.5

The relative effects on spatial reaction rates are shown in Table 3.7. The large difference for the UO_2 fuel case could well be misleading due to poor convergence.

TABLE 3.7. Comparison of Spatial Reaction Rate Distributions

	<u>Calculation</u>	<u>Difference, %</u>
Fission Per Rod (Outer Ring/Inner Ring)	PuO ₂ -UO ₂	2 ± 1
	UO ₂	0 ± 2
²³⁸ U Capture/Rod (Outer Ring/Inner Ring)	PuO ₂ -UO ₂	1 ± 2
(Epithermal)	UO ₂	11 ± 5

A comparison of calculated and experimental values for the lattice cell parameters for the natural UO₂ fuel cell are shown in Table 3.8. The ENDF/B cross sections were used for this calculation. The values in Table 3.8 are not defined in the same manner as are those in Table 3.4 but, rather, are calculated on the basis of the experimental definition. Note the close agreement between the two sets of calculated values. The disagreement in calculated and experimental values of k_{∞} is believed to be real, while the discrepancy in ϵ may still be due to a lack of agreement in the definition. The cause of the discrepancy in k_{∞} is not known. The agreement in all parameters except η^* leads to suspecting the thermal spectrum calculation or the cross sections for ²³⁵U and/or ²³⁸U.

TABLE 3.8. Comparison of Measured and Calculated Lattice Parameters

<u>Parameter</u>	<u>Experimental</u>	<u>Calculated</u>	
		<u>2-Dimensional</u>	<u>1-Dimensional</u>
k_{∞}	1.052 ± 0.003	1.040 ± 0.002	1.043 ± 0.003
ϵ	1.018 ± 0.002	1.024 ± 0.002	1.024 ± 0.002
p	0.868 ± 0.009	0.866 ± 0.002	0.867 ± 0.002
f	0.896 ± 0.003	0.899 ± 0.001	0.900 ± 0.001
η^*	1.329 ± 0.014	1.304 ± 0.006	1.305 ± 0.006

$$\eta^* = \frac{k_{\infty}}{\epsilon p f}$$

On the basis of this study, the errors introduced by using a one-dimensional model are concluded to be relatively small for both the situation representing the Batch Core Experiment and the UO_2 cell, and are probably overshadowed by errors in theory and/or cross sections. The magnitude of the error in the lattice cell constants would probably be less than 5 mk. The errors in burnup reaction rates would probably be in the order of 1%. The errors in local power distribution, i.e., within the cluster may, however, be emphasized during burnup.

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COMPARISON OF CALCULATED VERSUS MEASURED RELATIVE REACTION
RATES WITH RESPECT TO PARTICLE SIZE IN $\text{UO}_2\text{-PuO}_2$ FUEL RODS*

A. D. Vaughn

Introduction

Fuel element development for plutonium recycle has led to fuels composed of the mixed oxides of plutonium and uranium. The fabrication processes presently used in manufacturing of plutonium fuels result in distribution of the plutonium as particles throughout the uranium. In these fuels, power generation occurs mainly in the plutonium particles. Due to self-shielding properties of the particles, the reactivity of the fuel depends upon the particle size. Experimental and theoretical information about the effect of particle size on the characteristics of the reactor system is important for determining its impact on safety and economics.

Special fuel rods containing specific PuO_2 particle sizes were manufactured for a test lattice which could be inserted into the Physical Constants Testing Reactor (PCTR).⁽¹⁾ The experimental data necessary to determine the infinite medium neutron multiplication factor for particle size fuels in a graphite moderated lattice (4-in. pitch) have been obtained as a function of PuO_2 particle size ranging from a solid solution state to 350 μ . These determinations required the measurement of relative reaction rates of the fissionable isotopes with respect to particle size, and the relative reaction rates are presented in this report.

Plutonium and uranium fission foils were irradiated separately in specially constructed fuel rods. One set of foil holding rods was manufactured with solid solution $\text{UO}_2\text{-PuO}_2$ fuel, and another set with 350 μ particles of PuO_2 mixed with

* Portions of this work will be included in A. D. Vaughn's Master of Science Thesis Project in Nuclear Engineering to be submitted to the University of Washington

UO₂. Special foils were designed and fabricated to determine the self shielding of the particles in the 350 μ fuel rods. These foils contain 350 μ PuO₂ particles encased in aluminum.

All measurements were taken in the PCTR at room temperatures and the results are in excellent agreement with THERMOS-GRANIT Code^(2,3) calculations of the same parameters.

Discussion

Description of Measurement Facilities

An 8 $\times$ 8 array of 4-in. square, graphite lattice cells were stacked into the PCTR test cavity to provide the test assembly for these measurements. The four center cells of the array contained rods of either the solid solution set or the 350 μ set, depending on the measurement. Sixty cells surrounding the four center cells contained UO₂-2 wt% PuO₂ homogeneous fuel (Figure 3.5).

The reactor was made critical by the addition of PCTR drivers around the test cavity. The neutron energy spectrum and flux pattern were balanced across the test lattice by adjustment of the PCTR driver locations in the radial and axial directions, and by the addition of natural uranium and copper flux modifiers at the edge of the test lattice.

The match of the spectrum and the flux distributions were determined by irradiating bare and cadmium covered gold foils throughout the test lattice, and repositioning the flux modifiers until no significant energy or flux gradients existed across the cell boundaries around the four center test cells. This standard technique was used prior to measuring infinite medium neutron multiplication factors in the PCTR.

Description of the Particle Fuel Rods and Test Cells

Separate sets of four UO₂-2 wt% PuO₂ (8% ²⁴⁰Pu) fuel rods, one set containing plutonium-oxide in a solid solution state,

FIGURE 3.5. Fuel Arrangement in and Around PCTR Test Cavity

and another set containing 350 μ particles, were manufactured for the PCTR. These 0.566-in. OD rods were clad with 0.030-in. thick zirconium.

Two of the rods were built as separable units with a 0.005-in. thick cap at one end and a thin sleeve to contain

foils for irradiation in the rods in the center of the PCTR Test Cavity. Therefore, for each irradiation measurement, the four-rod center cell contained two complete fuel rods of a specific particle size and two sets of similar but separable particle rods containing foils at the axial center of the rods. The separable cells are depicted in Figure 3.6. However, the copper neutron absorbers at the cell boundary, depicted in the figure, were not present when those measurements were made.

Description of Task

Special plutonium foils were fabricated for the measurements in the 350 μ particle fuel rods. Aluminum discs (0.5-in. OD) were drilled with holes, each sufficiently small to contain a single 0.014-in. diameter PuO_2 particle. Each disc could contain approximately 45 particles. After insertion of the particles, the discs were clad in 0.005-in. thick aluminum covers. A diagram of the particle foils with respect to the particle rods is shown in Figure 3.7.

The plutonium reaction rate measurements in the solid solution rods were determined from a homogeneous foil of Al-1.8 wt% Pu, also encased in a 0.005-in. thick aluminum cover as depicted in Figure 3.8.

The ^{235}U reaction rates were determined from homogeneous foils of Al-7 wt% ^{235}U similar to the plutonium-aluminum foils but without the aluminum covers.

Theory of Measurement

Prior to the experiment, the fission foils used in this experiment were irradiated simultaneously on a graphite rotator in the PCTR thermal column.⁽⁴⁾ The flux in the thermal column was sufficiently constant so that any differences in count rate could be attributed to differences in the mass of fissile

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FIGURE 3.6. PCTR Center Test Cells with Separable Fuel Element and Foil Holders

FIGURE 3.7. Diagram of Particle Foils and Particle Fuel

FIGURE 3.8. Diagram of PuAl Foils and Solid Solution Fuel

material in the foils, provided the foils were of similar geometry and material. Therefore, the relative masses of the foils were determined by this method prior to the reaction rate measurements in the particle fuel.

The relative reaction rates of the particle foils with respect to the PuAl foils were also determined from this thermal column measurement. The results of the simultaneous irradiations of all the foils in a similar location were required by the analysis process. The measured parameters were as follows:

$$\frac{A_{\text{Fuel (ss)}}^{\text{PuAl}}}{A_{\text{Thermal Column}}^{\text{PuAl}}} = \text{Relative activity of PuAl foil in solid solution fuel rod to a similar foil irradiated simultaneously in the PCTR thermal column}$$

$$\frac{A_{\text{Fuel (350)}}^{350 \text{ Pu}}}{A_{\text{Thermal Column}}^{350 \text{ Pu}}} = \text{Relative activity of 350 } \mu \text{ particle foil irradiated in 350 } \mu \text{ particle fuel rod to a similar foil irradiated simultaneously in the PCTR thermal column.}$$

The differences due to the mass variations in these quantities could be eliminated by the data derived in the thermal column irradiations conducted prior to the experiment.

After correcting the measured data both for the residual counts taken before the irradiation and for background and flux peaking or dipping due to the presence of the foil (all these are mentioned later), essentially the following term was derived:

$$\frac{A_{\text{Fuel 350}}^{350 \text{ Pu}}}{A_{\text{Thermal Column}}^{\text{PuAl}}} = \text{Relative activity of 350 } \mu \text{ particle fuel in 350 } \mu \text{ particle rod to PuAl foil in thermal column}$$

This parameter cannot be directly measured because of the self-shielding differences due to the particle effect. From the following equation, one can see that the absolute mass of the foils must be known.

$$\frac{A_{\text{Fuel } 350}^{350 \text{ Pu}}}{A_{\text{Thermal Column}}^{\text{PuAl}}} = \frac{A_{\text{Fuel } 350}^{350 \text{ Pu}}}{A_{\text{Thermal Column}}^{350 \text{ Pu}}} \times \frac{A_{\text{Thermal Column}}^{350 \text{ Pu}}}{A_{\text{Thermal Column}}^{\text{PuAl}}} \frac{(\text{Mass PuAl in Thermal Column})}{(\text{Mass } 350 \mu \text{ foil in Thermal Column})}$$

The relative reaction rates of the ^{239}Pu isotope then could be determined by comparison with a similar parameter measured in the solid solution fuel rods, namely:

$$\frac{A_{\text{Fuel}}^{\text{PuAl}}}{A_{\text{Thermal Column}}^{\text{PuAl}}},$$

where the mass differences in this parameter can be directly removed by the previous and simultaneous irradiation of both foils in the thermal column.

It should be noted that the absolute weight of the fissionable isotopes in the foils must be known quite accurately in order to determine the relative reaction rates. Because these masses are on the order of 2 to 3 mg, most of the errors in the results were due to uncertainties in the masses.

Description of Measurements and Irradiations

Because only two rods in each set of four were separable, a plutonium foil was irradiated in one of the rods and the uranium and copper foils were irradiated together in the second rod but separated by a 0.001-in. thick aluminum catcher foil.

Identical foils were irradiated in the PCTR thermal column during each irradiation for normalization purposes.

The initial irradiations were done with bare foils, followed by irradiations with cadmium (0.040-in. thick) covering the foils in the separable rods. The foils in the PCTR thermal column were irradiated bare.

Each irradiation lasted 2 hr at a power level of 100 W. Counting started 8 hr after the irradiations so that the short-lived fission products had decayed to a negligible level. With only six foils per irradiation, the foils could be counted for a sufficient time and frequently enough to permit acquisition of excellent counting data.

The fission product decay gamma rays from the fission foils and the copper-64 decay gamma rays from the copper foils were counted using a sodium iodide detector and associated digital counting equipment. The energy discriminator was set to eliminate all counts from gamma rays with energy less than 600 keV.

Corrections to Foil Data

The unmodified relative foil activities were derived with respect to the foil activity in the thermal column by using the foil analysis code APDAC.⁽⁵⁾ This code corrects the data for background, subtracts residual counts derived by counting the fission foils before the irradiation, corrects for mass differences and decay time, and calculates the activation rates with respect to the thermal column foil with a statistical analysis of the results.

The data was exceptionally good even with the complicated decay rates associated with fission product activities. The error was less than 0.4% for the bare foils, and approximately 0.6% for the cadmium covered foils.

The results from the APDAC analysis needed further correction before use because of the peaking corrections in the fuel due to the presence of the foils, aluminum covers, and

end caps. Therefore, R - Z geometry transport theory calculations were performed for the foil and fuel diagrams depicted in Figures 3.8 and 3.9 using the DOT - SN⁽⁶⁾ Code. Cross sections for DOT were calculated with the THERMOS-GRANIT Code. The flux enhancement factor due to the presence of the foil in the fuel was obtained from the calculated ratio of the integrated flux in the foil area to a similar calculation using the same R - Z geometry, but with all fuel material in place of the foil and fuel. Thus, the correction factor was derived as follows:

$$\frac{A_o}{A_{\text{foil}}} = \frac{\Phi_{\text{fuel}}}{\Phi_{\text{foil}}}$$

where: A_o = The activity of the foil for flux in the area not affected by the presence of the foil
 A_{foil} = Measured foil activity
 Φ_{foil} = Integrated flux in foil region calculated by DOT - SN with fuel materials in all regions
 Φ_{foil} = Integrated flux in foil region calculated by DOT - SN with foil and other materials as defined in Figures 3.8 and 3.9.

Similar R - Z calculations for the copper foils and the Al-7 wt% ²³⁵U foils irradiated in the fuel were also performed to obtain the flux enhancement factors in the fuel necessary to derive correct measurements of the reaction rates of the ²³⁵U and the nonfissionable isotopes of uranium and plutonium. These corrections amounted to approximately 2% of the unmodified-measured foil activities in the fuel.

THERMOS-GRANIT Calculations

The experimental lattice cells were mocked-up in cylindrical geometry for calculation using the THERMOS-GRANIT Code. This code employs a special geometry routine to treat finite resonance shielding of particles embedded in cylindrical media by the method of Lane, Nordheim, and Sampson.⁽⁷⁾ The relative

reaction rates of the fission isotopes ^{235}U and ^{239}Pu were calculated for five different particle sizes comprising solid solution, 50 μ , 100 μ , 200 μ , and 350 μ , respectively, to derive a functional description of the reaction rates with respect to particle size. To obtain this function, the calculated reaction rates for each isotope with respect to a specific particle size were normalized to the average flux in the fuel region. The differences in the average flux in the fuel region between each particle size case were corrected by a factor derived from the measured copper foil data.

It was assumed that the relative activities of the copper foils irradiated in the fuel would indicate the differences in the flux with respect to the particle size fuel rods and, therefore, would also be directly proportional to the correction factors necessary to normalize the calculated quantities for comparison purposes.

Measured Results

The results of the foil measurements are listed in Tables 3.9, 3.10, and 3.11, for the plutonium, uranium, and copper foils, respectively. The interpolated activities are also listed for similar foil types in the 50, 100, and 200 μ rods, assuming a self-shielding effect linear with respect to the particle size. The same parameters calculated with the THERMOS-GRANIT Code are almost linear, as depicted in Figures 3.9 and 3.10. These same quantities are currently being used in the derivation of the infinite medium neutron multiplication factors by the PCTR techniques.

TABLE 3.9. Relative Plutonium Foil Activities with Respect to PuO_2 Particle Size

	Solid Solution	Particle Size, μ			
		Interpolated Values			
		50	100	200	350
Sub Cadmium	10.37	10.09	9.81	9.26	8.42
	± 0.10				± 0.14
Epi Cadmium	0.564	0.559	0.555	0.546	0.533
	± 0.006				± 0.009
Total	10.93	10.65	10.37	9.80	8.95
	± 0.08				± 0.14

TABLE 3.10. Relative ^{235}U Foil Activities with Respect to PuO_2 Particle Size

	Solid Solution	Particle Size, μ			
		Interpolated Values			
		50	100	200	350
Sub Cadmium	7.64	7.69	7.74	7.84	7.99
	± 0.05				± 0.05
Epi Cadmium	0.666	0.665	0.664	0.662	0.658
	± 0.006				± 0.006
Total	8.30	8.35	8.40	8.50	8.65
	± 0.04				± 0.05

TABLE 3.11. Relative Copper Foil Activities with Respect to PuO_2 Particle Size

	Solid Solution	Particle Size, μ			
		Interpolated Values			
		50	100	200	350
Sub Cadmium	7.777	7.825	7.873	7.970	8.115
	± 0.037				± 0.039
Epi Cadmium	1.247	1.247	1.247	1.246	1.245
	± 0.010				± 0.010
Total	9.024	9.072	9.120	9.216	9.360
	± 0.036				± 0.038

FIGURE 3.9. Comparison of Calculated and Measured Self-Shielding Effects in Plutonium with Respect to Particle Size Fuel Rods (UO₂-PuO₂)

FIGURE 3.10. Comparison of Calculated and Measured Self-Shielding Effects in Uranium with Respect to Particle Size Fuel Rods of UO₂-PuO₂

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4.0 FAST REACTORS

FAST REACTOR EXPERIMENTS IN THE PCTR

D. F. Newman

The Physical Constants Testing Reactor (PCTR) has proven its usefulness in past and present neutronics experiments on thermal reactor lattices.^(1,2) The experimental techniques developed in these studies are applicable to reactors of any energy range, and the application of the PCTR method to fast reactors is merely an extension of the type of experiments previously performed. A facility was developed in the center of the PCTR in which the neutron spectrum can be altered from a typical thermal reactor spectrum to a spectrum quite similar to that in large dilute fast breeder reactors. This coupled fast-thermal critical assembly is currently being studied to determine to what extent experiments of this type can benefit the operation of the Fast Test Reactor (FTR).

Description of the Fast Neutron Cavity

A large Fast Neutron Cavity (FNC) assembly has been designed, constructed, and operated in the central region of the PCTR. This assembly transforms the thermal reactor neutron spectrum from the surrounding PCTR "driver" region into a spectrum typical of a fast reactor by absorption of low energy neutrons and production of fast neutrons in a uranium metal region. The slightly enriched (0.85% ^{235}U) uranium buffer absorbs a large fraction of incident thermal and epithermal neutrons, while most of the incident fast neutrons pass through into the cavity region. Additional fast neutrons are produced in the neutron spectrum buffering process due to fission in the uranium. The fast neutron spectrum in the central cavity is degraded from the fission spectrum due to inelastic scattering of fast neutrons in the uranium buffer. As shown by neutron spectrum studies with proton-recoil counter and multiple foil

activation methods,^(3,4) the neutron flux per unit lethargy, $\phi(u)$, peaks at a neutron energy about 300 keV in the central cavity. The total flux in the FNC is 1.5×10^7 neutrons/cm²-sec-W, with 95% of the integral flux above a neutron energy of 1 keV.

The design concept of the FNC is similar to that of the Karlsruhe STARK Fast Thermal Argonaut Reactor.⁽⁵⁾ However, the central fast neutron spectrum zone of the FNC is approximately five times larger than STARK. The concentric cubic design of the PCTR/FNC facility is illustrated in Figure 4.1, and the nuclei densities and volume in each region of the critical reactor are summarized. This fast neutron cavity is believed to be the largest in operation. It can provide the required volume to equilibrate the neutron spectrum in a zone of fast reactor medium inserted in the central cavity. Calculational studies using the multigroup transport theory program, DTF-IV, and the Russian 26 neutron energy group set of cross sections have shown that meaningful reactivity experiments could be performed on fast reactor cores in the PCTR/FNC facility.⁽⁶⁾ Zoned fast reactor experiments are most accurate for those cores for which full-sized critical experiments would be exceedingly costly^(7a) because of the large volume of core material needed, with 6030 liters^(7b) for a 1000 MWe capacity as perhaps typical. Thus, it is concluded that neutronics experiments for large dilute fast breeder reactor media could be conducted in the FNC at large cost savings for fuel, when compared with fast critical assemblies.

FNC Design and Operation

The Fast Neutron Cavity (FNC) is designed to utilize the space in the PCTR test region fully.⁽⁸⁾ A cutaway drawing (Figure 4.2) shows the assembly scheme using 1.5-in. diam aluminum tubes stacked four rows thick enclosing a 1/2-in. thick

Region	Region Volume, liters	Material	Nuclei Density, nuclei/cm ³ × 10 ⁻²⁴
Cavity	376.2	N	3.98 × 10 ⁻⁵
		O	1.07 × 10 ⁻⁵
Stainless Steel	41.2	Fe	6.03 × 10 ⁻²
		Cr	1.73 × 10 ⁻²
		Ni	8.09 × 10 ⁻³
Uranium Buffer	274.6	²³⁸ U	3.36 × 10 ⁻²
		²³⁵ U	2.86 × 10 ⁻⁴
		Al	5.00 × 10 ⁻³
Transition Zone	353.4	²³⁹ Pu	3.72 × 10 ⁻⁵
		²⁴⁰ Pu	3.16 × 10 ⁻⁶
		²³⁵ U	1.18 × 10 ⁻⁴
		²³⁸ U	1.62 × 10 ⁻³
		O	5.14 × 10 ⁻³
		Al	1.05 × 10 ⁻²
		Zr	1.23 × 10 ⁻³
		C	6.58 × 10 ⁻³
Driver	1676.5	H	6.58 × 10 ⁻³
		²³⁵ U	3.53 × 10 ⁻⁵
		²³⁸ U	2.62 × 10 ⁻⁶
		Al	2.71 × 10 ⁻³
		O	1.12 × 10 ⁻⁴
		C	7.30 × 10 ⁻²
Reflector	8249.3	H	1.19 × 10 ⁻³
		C	8.28 × 10 ⁻²
		N	1.17 × 10 ⁻⁵
		O	3.15 × 10 ⁻⁶

FIGURE 4.1. Geometry and Composition of the PCTR Fast Neutron Cavity

Neg 0680893-2

FIGURE 4.2. Cutaway View of the Fast Neutron Cavity

stainless steel box (29.0-in. by 29.0-in. by 24.7-in. inner dimensions) on all sides. A removable stepped door in the front buffer provides access to the center of the FNC. The outside dimensions of this door (10.5-in. by 10.5-in.) will allow it to be removed through the 11.25-in. by 11.25-in. removable section in the moving face without opening the face. A 6-in. by 6-in. test sample can be removed through an access hole once the door has been withdrawn. The removable stepped door can also be removed when the movable face is fully opened (as shown in Figure 4.3). This mechanically stable rack of aluminum tubes containing uranium metal fuel is assembled as a liner of the central cavity in the PCTR. The inner two rows of tubes contain the uranium metal buffer fuel. The outer two rows of tubes contain high fissile content fuel and lucite rods. This undermoderated material ($H/^{239}\text{Pu} + ^{235}\text{U} \approx 24$) provides a transition zone in the reactor between the uranium buffer and the driver zone. This transition zone increases the efficiency of the thermal-to-fast neutron conversion due to its large thermal-neutron macroscopic fission cross section. The thermal neutron sink created by the large surface area of the 4000 kg metallic uranium buffer, without the highly enriched transition zone, could not be overcome by fully loading the PCTR driver fuel.

The thermal-to-fast neutron spectrum conversion was accomplished without significant reduction of the integral flux in the central cavity, when compared to the integral flux in the driver region, as shown in Figure 4.4. A large depression of the integral flux in the central cavity would reduce the sensitivity of a fast thermal-coupled reactor to small sample reactivity perturbations, and decrease the precision of fast reactor experiments in the central zone. Calculations show that such a depression would occur if an absorber material were used instead of fissionable material for the neutron spectrum buffer.

Neg 0681645-1

FIGURE 4.3. Removal of the Stepped Door to the FNC

FIGURE 4.4. Calculated Region Averaged Neutron Spectra in the PCTR/FNC Critical Facility

On the other hand, the number of fissions in the neutron spectrum buffer and the central fast zone should be a small fraction of the total number of fissions in the reactor. When most of the fissions are caused by the thermal neutrons in the driver region, the neutronic characteristics of the overall reactor are essentially unchanged from previous operating experience. Reactivity control is provided by the existing safety system in the thermal zone. The prompt neutron lifetime in the PCTR ($\ell_p \approx 1.5 \times 10^{-3}$ sec)⁽⁹⁾ is several orders of magnitude longer than the prompt neutron lifetime in a full sized fast critical assembly ($\ell_p \approx 10^{-7}$ sec).⁽¹⁰⁾ Thus, the power level rise in the PCTR would be over a much longer period than that of a fast reactor if the excess reactivity should exceed prompt criticality.

PCTR/FNC Calculational Model

A one-dimensional spherical calculational model of the PCTR was developed as an aid in understanding the spectral buffering requirements in the FNC. One-dimensional diffusion theory code, 1DX,⁽¹¹⁾ was used for generating effective nuclear cross sections using data in the 26-group Russian format. These cross sections were input to the S_n multigroup transport theory program, DTF-IV,⁽¹²⁾ using the P_0 scattering approximation and S_n order 4. Equivalent concentric spherical radii were obtained from the actual concentric cubic dimensions (Figure 4.1) using the geometric buckling equivalence relation:

$$R = \left(\frac{\sqrt{3}}{3}\right)S$$

where R is the radius of a sphere, and S is the side of a cube. Correlation of these calculations with measured critical fuel loadings and neutron spectra in the center of the FNC, both with a 30 wt% uranium (93% ^{235}U) in an aluminum fast core assembly,

installed and with the cavity completely empty, shows reasonable agreement. The calculated spatial variation of the neutron spectrum when the FNC was empty is shown in Figure 4.4. The distribution of fissions calculated for the empty FNC was 70.4% in the driver zone, 20.1% in the transition zone, and 9.5% in the uranium buffer. When the UA1 fast core containing 13.3 kg ^{235}U was installed in the center of the FNC (see Figure 4.5), the distribution of fissions in the driver zone (69.8%), the transition zone (19.9%), the uranium buffer (9.6%), and the fast core (0.7%), showed little change. The total reactivity worth of the eight PCTR control rods, measured in this critical loading, was 2.92\$. These results indicate that fast reactor cores large enough to equilibrate the neutron spectrum in the central region of the fast zone can be operated in the FNC without reducing the total control rod worth below the 2.0\$ minimum required by the PCTR Operating Specifications.

FIGURE 4.5. Aluminum-30 wt% Uranium (93% ^{235}U) Fast Core Installed in the Center of the FNC (Shown with the FNC Front Buffer Removed).

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NUCLEAR AND PROCESS TECHNOLOGY

R. E. Heineman

Nuclear Analysis Section

W. W. Little, Jr.

Criticality Experiments with 25 wt% PuO₂-U(Nat)O₂ Fuel Pins in Water

C. L. Brown and C. A. Rogers

Critical approach and exponential experiments are in progress to measure material bucklings, extrapolation distances, and critical masses as a function of lattice spacing for 25 wt% PuO₂-U(Nat)O₂ fuel pins in water. The data obtained from these experiments will provide a firm basis for criticality safety for the handling, storage, and shipment of FFTF fuel pins. The composition and dimensions of the fuel pins used in experiments are given in Table 4.1.

The experiments are being performed in a 5-ft diam × 6-ft high water tank located in the 305 B Building, Critical Approach Facility (CAF). Fuel pins are systematically loaded into vertical, triangular pitch lattices, which are reflected by water on all sides. Pin spacings are maintained by Lucite locator plates at the top and bottom of the lattices. A typical arrangement of pins is shown in Figure 4.6. Results of the lattices measured and evaluated to date are shown in Table 4.2. The error limits indicated in the table represent 95% confidence limits.

TABLE 4.1. Average Composition and Dimensions of 309 Fuel Pins Used in CAF Experiments

Clad OD, in.	0.230
Clad ID, in.	0.200
Clad length, in.	28.7
Clad material, in.	316SS
Fuel diam, in.	0.195
Fuel length, in.	27.25
Fuel density, g/cm ³	10.31 ± 0.22 (93 ± 2% TD)
Fuel form	PuO ₂ -UO ₂ - pellets
Fuel pin composition	
Fuel, g	138.40 ± 1.33
PuO ₂ , g	34.87
Pu, g	30.75 ± 0.30
U, g	91.16 ± 1.03
O, g	16.52 ± 0.17
²⁴⁰ Pu Content of Pu, wt%	11.93 ± 0.06
²³⁵ U Content of U, wt%	0.71
Enrichment, wt% PuO ₂	25.20

FIGURE 4.6. Fuel Pin Lattice in Water (Spacing Between Pins is 0.45 in.)

TABLE 4.2. Results of Criticality Experiments with 25 wt% PuO_2 (Nat) O_2 Pins in Water

Critical Approach Experiments

<u>Lattice Pitch, in.</u>	<u>Water to Fuel Volume Ratio</u>	<u>No. Pins in Experi- mental Lattice</u>	<u>No. Pins for Critic- ality (extrapolated)</u>
0.779	15.423	148	156.2 ± 0.8
0.900	21.007	141	149.8 ± 0.6

Exponential Experiments

<u>No. Pins in Experi- mental Lattice</u>	<u>b_{11}, cm^2</u>	<u>Material Buckling M^{-2}</u>	<u>Extrapo- lation Distance, cm</u>
78	-12.72 ± 0.10	160.9 ± 1.3	6.88 ± 0.06
90	-15.05 ± 0.12	160.8 ± 1.4	6.89 ± 0.06
Average		160.9 ± 1.0	6.89 ± 0.04
66	-12.10 ± 0.16	138.3 ± 1.3	6.91 ± 0.09
78	-14.27 ± 0.11	138.6 ± 1.0	6.89 ± 0.06
Average		138.5 ± 0.8	6.90 ± 0.05

Core Analysis Section

P. D. Cohn

Metal Swelling Effects on FTR Core ComponentsR. J. Jackson, C. L. Wheeler, G. R. Waymire,
and W. W. Little

The purpose of this study was to develop an understanding of the predicted major interaction between FTR core components caused by fast neutron induced metal swelling, to identify the most severe problem areas, and to provide guidance to the RPD for core design. A summary of the results is tabulated in Table 4.3. It may be noted that there are swelling induced interactions between all major core components. The document reporting the results of the study is presently in process of publication.

The limitation on fuel lifetime may be determined by the structural performance of core components instead of fuel behavior. Therefore, it is necessary to evaluate the following potential burnup limiting criteria:

- Reactivity Change - The net reactivity change appears to be positive. If this reactivity increase is predictable, it will not be a burnup limitation.
- Refueling - The limitation of fuel burnup life associated with refueling will be related to a maximum permitted distortion which is, at present, undefined.
- Duct Contact - Duct-to-duct contact is considered as a burnup limitation. Immediately after duct-to-duct contact, a reverse bending would begin in a duct in the fluence range of 10^{22} range. Its ductility is immeasurably small for all practical purposes. Therefore, duct contact from a combination of dilatation and bending distortion undoubtedly constitutes a burnup limitation.

TABLE 4.3. Metal Swelling Effects on FTR Core Components

<u>Component</u>	<u>Description of Effects</u>
General	<ol style="list-style-type: none"> 1. More swelling is predicted for solution-treated 304 than for cold worked 316. 2. Temperature reduction, core length reduction, and duct rotation ameliorate the swelling effects.
Driver Ducts	<ol style="list-style-type: none"> 1. Dilatation is the primary cause of duct-to-duct contact between Rows 1, 2, and 3. 2. Duct bending is largest in Rows 5 and 6. 3. Duct top deflection is almost entirely restrained by the relatively straight reflectors in Rows 7, 8, and 9 during refueling. 4. Duct length increase can be accommodated by minor design changes in the instrument tree.
Control Rod	<ol style="list-style-type: none"> 1. Large diametral clearances are required between the guide tube and the inner duct for control rod life greater than 6 months.
Closed Loops	<ol style="list-style-type: none"> 1. Large volume increases are predicted for the closed loop components based upon ultimate operating temperatures. 2. The dilatation and bending associated with the 316 cold-worked closed loops causes duct-to-duct contact between driver ducts in Row 5 and the closed loop in Row 6 at about 280 days (approximately 40,000 MWd/MTM in Row 5 at this time). 3. A crushing interference between the flow tube and the flow tube liner is predicted because of differential metal swelling of these two coaxial components in the core region.

Rotating the ducts 180 degrees at intervals during their life greatly reduces the predicted bending deflection. This effect is illustrated in Figure 4.7 which compares the predicted displacement for a rotated and nonrotated 316 cold-worked, Row 6 driver duct. This figure presents the duct movement in terms of displacement instead of clearance because some assumption regarding fuel management is necessary before clearances can be calculated. The clearance can be considered as (0.100-displacement) if the adjacent duct wall is assumed as a fixed straight wall.

The duct wall displacement results from duct deflection (δ) and radial dilation (ΔR). The magnitude of these individual contributions to duct wall deflection are shown as dashed lines. The solid lines represent the combined effect of bending and radial dilatation. It may be noted that the rotated duct is predicted to have about a 0.040-in. maximum displacement over the 600-day period, compared with about a 0.250-in. displacement for the nonrotated duct. It may be noted that, after 600 days, radial dilatation is becoming significant in Row 6. This particular case is not optimum because very little reduction in deflection is gained by the rotation at 200 and at 400 days. Obviously, additional study to determine the tradeoff between reactor downtime and extending the burnup life of core components is required.

The major study conclusions are as follows:

- Burnup limiting duct-to-duct contact is predicted to occur before goal burnup in either a 304 solution-treated core, or a 316 cold-worked core. The dilatation effects are the primary cause of duct-to-duct contact in Rows 1, 2, and 3. Duct rotation or fuel shuffling will not extend the burnup limit appreciably in these rows. The duct bending effect, the primary cause of direct contact in Rows 5 and 6, could be reduced by duct rotation.

FIGURE 4.7. Effect of Duct Rotation on Displacement

- Duct top displacement during refueling is very uncertain. Driver ducts want to move radially outward, but are prevented from reaching their stress-free location by control rod guide tubes, closed loops, and reflectors. The relatively stiff closed loops play a dominant role in determining the driver duct top locations.
- Because of the large clearances required between the control rod guide tube and the inner duct, a balanced design is essential for either 304 or 316. This design would reach end of life in the fission gas plenum while simultaneously using up the interduct clearance.
- Row 6 cold-work, 316 closed loops are predicted to swell in such a manner that they make contact with adjacent driver ducts in Row 5 at 280 days. This is at a burnup of about 40,000 MWd/MTM. Rotation of the closed loop would increase this life limitation. The testing potential for a Row 6 closed loop needs analysis.
- There are significant uncertainties associated with predictions of swelling. The prediction is very sensitive to temperature. A decrease in metal temperature of 50 degrees lowers the predicted swelling by 17% in the driver ducts. Certain serious questions about the equation validity for U.S. produced stainless steel have been raised.
- Many additional analyses are needed to complete the understanding of the interaction between core components. The major areas are:
 - closed loop - driver duct interaction
 - flow tube - flow tube liner interaction
 - guide tube - inner duct interaction
 - refueling interactions
 - effect of support pad swelling on interactions

- Because metal swelling has a high sensitivity to temperature, further analyses are necessary to refine the estimates of component temperature distribution. This analysis is particularly needed in the control rods, closed loops, and reflectors.

Experimental Physics Section

R. A. Bennett

An Analysis of Subcritical Measurements of Peripheral Control Rods in ZPR-3 Assembly 56B

V. O. Uotinen

Introduction. Reactivity worths of control rods, inferred from responses of neutron sensors, located in various positions with respect to the rods, can be expected to differ. The magnitude of the variation is, of course, dependent upon the proximity of sensor and rod, and upon the magnitude of the worths of the rod. This has been demonstrated in recent FTR Critical experiments⁽¹⁾ performed in ZPR-3 Assembly 56B. The reactivity worth of four large peripheral B_4C rods, as inferred from detectors in different locations, ranged from -1766 ± 6 lh to -2021 ± 4 lh. Similarly, the worth of four small B_4C rods ranged from -925 ± 5 lh to -1000 ± 2 lh. The measured rod worths inferred from the standard out-of-core ZPR-3 flux detectors were larger in magnitude than those inferred from a detector located in the center of the reactor, by 12% and 3.3%, respectively, for the large and small B_4C rods. Intuitively, one would expect the flux at the center to be the least perturbed and, thus, that the central detector would yield the most correct result. However, on the basis of the experiment alone, one cannot definitely predict the measured worths. Furthermore, if, as suspected, the central detector gives the correct result, this implies that peripheral rod worths, measured using the standard ZPR-3 detectors, are consistently wrong. If this is the case, one should apply correction factors

to peripheral rod worths that have been measured in the past (Assembly 48A) with the standard detectors.

A theoretical analysis is essential so that the measured rod worths can be interpreted meaningfully. The purpose of this report is to describe an analysis performed using two-dimensional diffusion theory, and to present the significant results and conclusions of the analysis.

Measured Reactivity Worths. The degree of subcriticality $\Delta k_{\text{ex}}(\text{sub})$ of the reactor is given by the relationship

$$\Delta k_{\text{ex}}(\text{sub}) = -Q/CR,$$

where Q is an effective source term which, when divided by the count rate, CR , yields $\Delta k_{\text{ex}}(\text{sub})$. The source term Q may be inferred using two techniques. The first is a period technique involving count rate recording as a function of time when the reactor is on a positive period, then applying inverse kinetics analysis to the data to obtain Q . The second method is a two-power level technique in which the subcritical count rate is measured at two levels of subcriticality differing by a known reactivity amount (e.g., movement of a calibrated control rod). The source constant Q can be evaluated by taking the difference between the Δk in the two cases.

The effective source term is different for each detector, and even for the same detector when it is in different locations, because the source term is proportional to the absolute detection efficiency (counts detected/total events in reactor) of the detector. Once Q has been measured for a given detector in a given location, the degree of subcriticality is obtained by simply dividing Q by the observed count rate.

This simple theory assumes that the effective source and the detection efficiency remain constant as a perturbation is introduced to change the k of the reactor. In practice, this

is not necessarily a valid assumption. Any change in fuel mass changes the effective source because the spontaneous neutrons from the fuel material constitute the source. Also, any change altering the composition and/or geometry at the periphery of the core will cause a perturbation which will change the detection efficiencies of detectors. These perturbations may be negligible in some cases, but this is not necessarily the case. One must consider each measurement carefully and, if necessary, apply calculated correction factors to assure that measured results are interpreted correctly.

Reference 1 (ANL-7577) contains a detailed description of these "peripheral poison control composition worth experiments." The poison control drawers, containing platelets of B_4C and of Na, are illustrated in Table 4.4. Homogenized material densities are listed in the Table.

The measurements were performed by first measuring the reactivity for the base case. Then the reactivity was measured after poison control material replaced radial reflector material in four single matrix locations (marked I in Figure 4.8). After this, radial reflector was replaced by poison control material in 12 additional matrix locations (marked II in Figure 4.8). Finally, the reactivity of the base case was again measured. Corrections were made for control rod movement and differences in temperature, so that all cases could be compared on the basis of common reference condition. The following description of the various detectors is from ANL-7577:

"The standard flux detectors are the two current chambers labeled 4a and 4b in Table 4.5 (see also Figure 4.8). These chambers, filled with BF_3 gas at 1 atm, are 7.6 cm in diameter with an active length of 19 cm. The two chambers are connected in parallel, and the current is measured with a high-stability electrometer. The electrometer output is digitized by a voltage-to-frequency

TABLE 4.4. Drawer Loading for Peripheral Poison Drawer

	Na		Ni
	B ₄ C		Na
	Na		Ni
			Ni
	B ₄ C		Na
	Na		Ni

0 in. 8 in. 14 in. 18 in. 21 in.

Drawer Loading Used for Measurements in the Radial Reflector

Homogenized Material Densities (a/b-cm) for Poison Control Drawers

Na	0.008736
B-10	0.007145
B-11	0.02897
C	0.009358
Fe	0.01054
Cr	0.002651
Ni	0.001177
Mn	0.0001192
Si	0.0001391

S - SAFETY ROD T - SOURCE TUBE I - POISON CONTROL I LOCATIONS
 C - CONTROL ROD N - NEUTRON LEVEL DETECTOR II - POISON CONTROL II LOCATIONS

NOTE: ALL DIMENSIONS IN CENTIMETERS

FIGURE 4.8. Detector and Matrix Locations for the Peripheral Poison Control Worth Measurements

TABLE 4.5. Detector Locations for Peripheral Boron Rod Worth Experiments

Detector No.	Type ^a	Signal	Matrix Element	Geometric Location	
				Radius, ^b cm	Distance from Midplane, ^b cm
1	PC	Pulse	1-P-16	0	7.6
2	PC	Pulse	2-P-16	0	63.0
3	PC	Pulse	1-T-12	31-3	63.0
4a + 4b	IC	Current	External	~110	15.0

^a All detectors were BF_3 gas-filled chambers. PC--proportional counter; IC--ion chamber.

^b Distance to center of the active volume.

converter, and the resultant pulses are summed in a precision EPUT meter to yield a digital count rate proportional to the chamber current.

"The three additional detectors were all BF_3 gas-filled proportional counters. Detector No. 1 was 0.9 cm in diameter with an active length of 5.1 cm. Detectors 2 and 3 were 2.5 cm in diameter with active lengths of 15 cm. The data for Detector 1 was taken on a pulse-height analyzer. The locations of the detectors were chosen to evaluate their influence upon the individual count rates as interpreted by the standard subcritical technique. Detector 1 measured the change reaction rate at the core center; hence, a simple analysis based on results from this detector would seem intuitively to yield the most nearly correct answer. Detector 2 predominantly measured the change in axial leakage; hence, it could be expected to yield too small a change and, therefore, too small a reactivity worth for the poison control compositions. Detectors 4a and 4b, which measured the change in radial leakage, could be expected to yield

too large a change and, therefore, too large a reactivity worth for the poison control composition. The actual worth could be expected to be between those calculated from the change in axial leakage and the change in radial leakage. Detector 3 measured a mixture of many effects with possible localized effects due to the proximity of the poison control composition itself; hence, the results based on this detector should be the most difficult to interpret as to physical meaning."

The measured worths deduced from the various detectors did indeed follow the pattern that was expected. The measured results are presented in Table 4.6.

Calculated Reactivity Worths. The problem of calculating the change in flux and reaction rate at all detector locations is a three-dimensional problem. However, a two-dimensional analysis such as has been performed here allows one to compare the central detector (expected to yield the most correct worths) with the standard ZPR-3 detectors. Such a comparison can shed some light on some of the important questions involved.

Two-dimensional diffusion-theory calculations in X-Y geometry were performed using a slightly modified version of the code, 2DB.⁽²⁾ The modified version, 2DB,⁽³⁾ was developed for FTR shielding calculations, and one of its features is the ability to directly calculate activations of interest. The reactor model was a quarter (37×35 mesh) of the reactor as shown in Figure 4.9. In the model the y-dimension of the ZPR-3 was extended out beyond the physical boundary of 85.746 cm to 106 cm. This was done by extending the dilute stainless steel "mesh" in order to obtain fluxes and reaction rates at the locations of the ZPR-3 standard flux detectors, which are on top of the reactor. This extension of the mesh has a negligible effect on the calculated reactivity.

TABLE 4.6. Results of the Peripheral Poison Control Composition Worth Experiments (Replacing Radial Reflector Composition)

		Experiment	ρ (lh), Core Excess Reactivity			
		Reference 1:	34.2			
		Reference 2:	32.4			
		Average:	33.3			
		Units	1	2	3	4a + 4b
Poison Control, Configuration I						
Effective Source (Q)	ct/sec		$1.692 \pm 1\%$	$38.56 \pm 1\%$	$9.395 \pm 1\%$	$271.0 \pm 1\%$ ^a
Corrected Count Rate ^b	ct/sec		$188.9 \pm 0.5\%$	$4,510 \pm 0.5\%$	$1,101 \pm 0.4\%$	$29,310 \pm 0.2\%$
$k_{\text{eff}}(\text{sub})^c$	% k		-0.896 ± 0.005	-0.855 ± 0.005	-0.903 ± 0.004	-0.925 ± 0.002
ρ (sub) ^c	lh		-939 ± 5	-896 ± 5	-947 ± 4	-971 ± 2
$\Delta\rho$ (Temp)	lh		4.6	4.6	4.6	4.6
Total $\Delta\rho^c$	lh		-968 ± 5	-925 ± 5	-976 ± 4	$-1,000 \pm 2$
Poison Control, Configurations I and II						
Corrected Count Rate ^b	ct/sec		$100.4 \pm 0.6\%$	$2,338 \pm 0.4\%$	$560.9 \pm 0.3\%$	$14,362 \pm 0.2\%$
$k_{\text{eff}}(\text{sub})^c$	% k		-1.685 ± 0.010	-1.649 ± 0.006	-1.675 ± 0.005	-1.887 ± 0.004
ρ (sub) ^c	lh		$-1,781 \pm 11$	$-1,743 \pm 7$	$-1,771 \pm 5$	$-1,998 \pm 4$
$\Delta\rho$ (Temp)	lh		9.9	9.9	9.9	9.9
Total $\Delta\rho^c$	lh		$-1,804 \pm 11$	$-1,766 \pm 7$	$-1,794 \pm 5$	$-2,021 \pm 4$

^aFor 10^{-8} scale on the electrometer.

^bUncertainties include statistics and uncertainty in the dead time correction to the count rates.

^cQuoted uncertainties are for count rate only. An additional 1% uncertainty should be added in quadrature to account for the uncertainty in the effective source term measured for the reference configuration.

The one-dimensional diffusion-theory code, 1DX,⁽⁴⁾ was used to generate 13-group cross sections⁽⁵⁾ from a modified 26-group Russian set. Cross sections for materials in each zone were averaged over the 1DX spectrum appropriate for that zone. Each of the 13 broad groups contained two of the narrow groups.

An eigenvalue calculation and a source calculation were performed for each of three reactor configurations: 1) a base case, no peripheral poison; 2) poison in four single matrix locations; and 3) poison in four groups of four matrix locations.

For the source calculations, a neutron source uniformly distributed in the fuel zones was included to account for spontaneous fissions in isotopes of plutonium and uranium. The following source strengths (in neutrons/g-sec) were used for

FIGURE 4.9. Map of a Quarter of the Reactor, Showing the Percent Overestimation of the Worth of Four Large B_4C Rods as a Function of Position. (The lines are lines of "equal-overestimation").

$^{238}\text{Pu}/^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}/^{238}\text{U}$, respectively: 2420.0/0.022/1012.0/0.022/1804.0/0.014. An α -n reaction in oxygen, which constitutes an additional source of neutrons in plutonium oxide fuels, was not present in these experiments because the plutonium was metallic and clad.

Activations were calculated for absorptions in ^{10}B and for fissions in ^{235}U , using both cross sections averaged over the core spectrum and cross sections averaged over the reflector spectrum. The axial buckling was adjusted to $5.2391 \times 10^{-4} \text{ cm}^{-2}$ to give a k_{eff} of 0.9995 for the base case.

It might be argued that one could never hope to calculate accurate fluxes at points as far from the core ($\sim 110 \text{ cm}$) as the locations of the standard detectors. However, we are not interested here in the absolute value of the flux at these locations but rather in the ratio of fluxes from two cases. It is maintained that the ratios of fluxes are meaningful even though the absolute values are systematically wrong. The calculated results give credence to this contention.

Reactivity worths deduced from calculated fluxes and reaction rates at detector locations are presented in Table 4.7. A ratio R is also given, which is the ratio of the worth obtained from reaction rates to the "correct" worth given by the eigenvalues. In the center of the reactor, all three, i.e. total flux, $\sigma_a^{10\text{B}}$, and $\sigma_f^{235\text{U}}$ reaction rates, yield reactivity worths in good agreement with the worths based on calculated eigenvalues. This agreement indicates that the insertion of peripheral control rods introduces a negligible distortion at the center of the reactor.

At the locations of the standard ZPR-3 flux detectors, all three calculated ratios overestimate the change in reactivity. These results indicate that the insertion of peripheral control rods introduces rather large distortions in the flux

TABLE 4.7. Calculated Reactivity Worths Deduced from
Calculated Reaction Rates of Various Detectors

Detector Location	Detector	4 Small B ₄ C Rods		4 Large B ₄ C Rods	
		Calculated Worth (lh)	R*	Calculated Worth (lh)	R*
Center of reactor	Reaction rate in ¹⁰ B	1005	1.000	1923	1.003
	Fission rate in ²³⁵ U	1003	0.998	1915	0.999
	φ Total	1002	0.997	1909	0.996
Location of standard ZPR-3 detectors	Reaction rate in ¹⁰ B	1056	1.051	2255	1.176
	Fission rate in ²³⁵ U	1052	1.047	2215	1.155
	φ Total	1045	1.040	2153	1.123

* *R is the ratio of the worth deduced from reaction rates to the worth as obtained from the calculated eigenvalues.*

at the locations of the standard ZPR-3 detectors. This, of course, was what the measured results seemed to indicate also. With regard to the question of how much the reactivity worth is overestimated when deduced from the standard detectors, the calculated results agree fairly well with the measured results. The calculated total flux gives the closest agreement with measurements indicating an overestimation of 13% and 4.3%, respectively, for the large and small B₄C rods, as compared with measured overestimations 12% and 3.3%. This is surprising at first thought, because one would rather expect the calculated ¹⁰B reaction rate to give the best agreement with measurement since BF detectors were used. However, the standard BF₃ detectors are surrounded by a hydrogenous material which thermalizes the fast neutrons, thus enhancing the count rate. Therefore, it is not surprising that the total flux would give results

that agree most closely with measured results. The amount of overestimation (%) of the worth of 4 large control rods as a function of detector position is shown in Figure 4.9.

It appears that the calculated ratio of total flux (i.e. base case/case with peripheral control rods) at the standard ZPR-3 detector location, when compared to the same ratio at the center of the reactor, provides a reasonable correction factor which can be applied to measured worths deduced from the standard detectors.

Summary and Conclusions. Two-dimensional diffusion-theory calculations in x-y geometry have been made using the code 2DBS. Reactivity worths were deduced from the changes in calculated fluxes and reaction rates at detector locations as peripheral control drawers replaced radial reflector. The results of these calculations indicate that a detector in the center of the reactor yields correct results, and that the standard ZPR-3 flux detectors can be expected consistently to overestimate the worth of such peripheral rod configurations. Furthermore, it appears that two-dimensional diffusion theory calculations can be used to calculate reasonable correction factors which can be applied to peripheral rod worths that have been inferred from the standard ZPR detectors.

Finally, a rather obvious suggestion arising from the present analysis is that, in future subcritical measurements of peripheral rod worths, one should place a detector in the center of the reactor where possible.

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Expected Count Rates in the Shutdown FTR and the Question of an Extraneous Startup Source

V. O. Uotinen

Introduction. Calculations have been performed to determine what count rates can be expected from in-vessel sensors in the shutdown ($k \approx 0.9$) FTR, and whether a startup source will be necessary in order to obtain a statistically meaningful count rate. Earlier conclusions, stated in the CSDD for System 95,⁽¹⁾ and in the FFTF Monthly Informal Technical Report, March 1969,⁽²⁾ were conflicting, and were based on preliminary calculations. A more exact calculation has been performed, and it is concluded that an extraneous startup source will not be required in the FTR. This conclusion agrees with that of an earlier evaluation.⁽³⁾ Some background on this problem is reviewed, and some recent conclusions are presented and discussed.

Background. It was stated in the CSDD for System 95⁽¹⁾ that the expected count rate at shutdown ($k = 0.9$) of a low level neutron detector (2 in. ^{235}U fission chamber) located at a radius of 96 cm on the core midplane is 10-20 counts/sec when operating in a gamma-field of 10^5 R/hr. In a gamma-field

of 10^6 R/hr, the expected count rate is reduced to less than 5 counts/sec. On this basis, it was concluded⁽⁴⁾ that a start-up source would probably be needed. These expected results were based on preliminary calculations of radial reaction rates for a reactor at 400 MWt and with no peripheral control rods. An estimated ratio of 2×10^{11} was used to extrapolate the calculated reaction rate to the shutdown reactor ($k = 0.9$) and an estimated correction (40%) was applied to account for a flux depression caused by peripheral control rods.

Other preliminary investigations⁽²⁾ indicated that expected count rates would be considerably greater than those predicted in Reference 1.

Summary and Conclusions of Recent Calculations. Subcritical source calculations have been performed using the code 2DBS to obtain reaction rates directly for the subcritical FTR at $k = 0.9$. Thus, no extrapolation to shutdown was necessary. Also, the reactor model contained peripheral control rods. Thus, no correction was necessary for the flux depression caused by the peripheral control rods. Activations using fission cross sections for ^{235}U averaged over a spectrum characteristic of the shield were calculated.

These calculations indicate that the expected count rate at 96 cm, using a commercially available 2-in. ^{235}U fission chamber, is 50 counts/sec when operating in a gamma-field of 10^6 R/hr. The much smaller earlier estimate was due, in part, to a less exact calculational model. However, the main cause of the low estimate was an incorrect extrapolation from 400 MW to the shutdown condition. A factor of 2×10^{11} had been assumed as the ratio of the flux at 400 MW to flux at $k = 0.9$. Our recent calculations indicate a more accurate estimate of this ratio, 4×10^{10} , to be in agreement with other recent calculations⁽⁵⁾ also. The estimate of 50 counts/sec in a gamma-field of 10^6 R/hr is considered to be a conservative estimate of the expected count rate.

Furthermore, the predicted count rate of 50 counts/sec is more than adequate for monitoring the subcritical reactivity. In fact, according to stated design requirements, a count rate as low as 10 counts/sec would be acceptable.⁽⁶⁾ On the basis of this study, it is concluded that a startup source will not be required for the FTR.

Description of Calculations. Two-dimensional diffusion-theory calculations in R-Z geometry were performed using the code 2DBS. The two-zoned reactor model was essentially the same as that used by Sheff⁽⁵⁾ in his subcritical calculations except that, in this case, the densities recommended by Westinghouse⁽⁶⁾ (4/30/69) were used. The one-dimensional diffusion-theory code 1DX was used to generate 13-group cross sections (from a modified 26-group Russian set) for materials in the various zones. The calculational method is the same as that used to analyze current FTR critical experiments. The calculated eigenvalue for this model of the shutdown FTR was 0.908.

A distributed neutron source was inserted in all zones containing fuel to account for spontaneous fissions and α -n reactions with oxygen. The major contributor to the source (about 65% of the total) is the spontaneous fission of ^{240}Pu , for which the half-life is known better than $\pm 10\%$. The yields from α -n reactions are less well known, but the uncertainty in the total source strength, due to uncertainties in neutron yields, is estimated to be ± 10 - 15% . There is an additional uncertainty due to the uncertainty of isotopic concentrations, particularly that of ^{240}Pu which was assumed to be 12%. If this concentration is uncertain by $\pm 1\%$ one can arrive at an estimate of $\pm 20\%$ for the uncertainty in the total source strength due to spontaneous fission and α -n reactions in the fuel pins.

For a $k_{\text{eff}} = 0.9$, the ^{235}U reaction rate at a radius of 96 cm is calculated to be 1.3×10^5 reactions/sec per 10^{24} atoms. A commercial 2-in. diam fission chamber containing 1.8 g of ^{235}U (2 mg/cm^2) is equivalent to ~ 600 reactions/sec. Assuming a conservative,⁽⁷⁾ 50% efficiency, and 15% absorption loss, we have 250 counts/sec. When one compensates for a gamma-field of 10^6 R/hr, the efficiency of the detector is reduced⁽⁸⁾ to 20% of its original value, thus, giving us 50 counts/sec.

Experience in using this calculational method to calculate traverses in FTR critical experiments has shown a consistent underestimation of reaction rates at large radii. This is a further reason why the estimated 50 counts/sec is considered a conservative estimate of the expected count rate.

According to the calculations, one could gain about a factor of two in count rate by moving the detector in to a radius of ~ 80 cm. To move it in as far as 72 cm would not result in a further gain. In fact, it could result in a loss if the detector is shielded by a peripheral control rod, as indicated by R- θ calculations of this same problem.

Calibration of Subcritical Reactivity Monitor. Although an extraneous source may not be required in order to obtain a statistically meaningful count rate, another argument for an extraneous source is that it would provide a more convenient calibration for the reactivity monitor. If the extraneous source is much larger than the built-in natural source, and if it has a long half-life, the calibration of the source strength could be essentially independent of irradiation and fuel loading changes. In the absence of an extraneous source, one would have to recalibrate after every significant fuel loading change, and/or apply a calculated correction to an earlier calibration.

However, it would seem that calculated corrections can be made with acceptable accuracy, and that periodic recalibration of the source strength is not a major inconvenience. Thus, even for purposes of calibration, one is not justified in saying that an extraneous source is a requirement.

Expected Gamma Intensities. Immediately after shutdown, the gamma field is expected to be above the proposed 10^6 R/hr discrimination level. Thus, the detector will be indicating counts due to gamma pile-up. However, if the gamma-field were as large as 10^7 R/hr (at least a factor of two larger than anticipated⁽¹⁰⁾ for saturation gamma intensities), the expected count rate due to gamma pileup would be only ~ 600 counts/sec.⁽⁹⁾ This gamma-caused count rate will, of course, decay rapidly in the first few hours following shutdown according to the usual sum-of-exponentials decay scheme, leveling off at an intensity dictated largely by the -intensity of Co^{60} (5.26-year half-life). There is some uncertainty regarding the magnitude of the saturated Co^{60} -intensity, but it can certainly be limited. For example, one can either specify a low cobalt content in the steel surrounding the detector or shield the detector in some way.

Verifying the Response of Chamber to Neutrons. Against the decaying gamma background in the hours following shutdown, one should verify that the detector is also responding to neutrons. This can be done quite easily by taking several one-minute counts after shutdown to establish the decay curve, then removing a safety rod or two and observing a change in the smooth decay curve. After a few hours, the gamma intensity will have decayed to below 10^6 R/hr. When this point has been reached, the measured count rate is, of course, constant with time because the chamber is responding only to neutrons.

A Large Source to Replace In-Vessel Sensors. It has been suggested that a large enough extraneous source could conceivably permit the proposed in-vessel sensors to be moved outside the vessel and thus provide a savings on the cost and engineering effort required to locate sensors inside the vessel. However, our recent calculations indicate that, in order to obtain acceptable count rates at positions outside the vessel, the extraneous source must have intensity of $\sim 5 \times 10^{10}$ n/sec. A ^{252}Cf source of such intensity would cost \$300,000 - \$500,000 at projected costs for the early 70's. An americium beryllium-curi-um (ABC) source of such intensity could conceivably be obtained at a lower cost, but would have the disadvantage of a short (163 days) half-life. Furthermore, detectors located outside the vessel will measure the spontaneous fission source of the stored fuel as well as neutrons originating in the core. At shutdown, the signal measured by an out-of-vessel detector would be mostly from the stored fuel.⁽¹⁰⁾ In summarizing the discussion on this point, it appears that the disadvantages outweigh the advantages.

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Reactivity Effects of Stainless Steel Swelling

S. L. DeMyer

SS 304 is expected to exhibit significant swelling as a result of exposure to neutron flux in the FTR. Differential swelling is expected where there are neutron flux and temperature gradients across the shroud tubes. The bowing magnitude of the individual shroud tubes and the resultant whole core distortion is a function of the steepness of the gradient, the flux, the temperature, and the diameter of the shroud tube.

For the reference design of the FTR,⁽¹⁾ and 30,000 MWd/t exposure, the distortion is characterized by a maximum radial contraction at the midplane of 0.29%, or 0.1886 cm on a radius of 65.8686 cm, and a maximum expansion along the central axis of 0.32%, or 0.1473 cm on a half height of 45.7200 cm.

It has been calculated that this distortion will yield a reactivity effect of $+0.34\% \text{ dk} \pm 0.04\% \text{ dk}$. This uncertainty reflects the precision of the convergence but not the limitations of the calculational model. Thirteen energy group cross sections were used in an R-Z geometry, two-dimensional diffusion calculation. The computer code 2DB⁽²⁾ was used.

The cross sections were collapsed from the 26-group revised Russian set⁽³⁾ over zone average spectra in 1DX.⁽⁴⁾ The atom densities used in the 1DX calculation and the undistorted core calculation are listed in Table 4.8.

Figure 4.10 represents the 2DB calculational model in R-Z geometry of the undistorted core. Zones 1-4, 7-10, and 13-16 are the inner core, while Zones 5, 6, 11, 12, 17 and 18 are the outer core. The calculation contains 37 radial and 31 axial mesh intervals in an evenly distributed pattern.

Figure 4.11 represents the calculational model of the distorted core. There are three levels of axial expansion and six levels of radial contraction allowed. As a result, 18 core zones are formed, 12 in the "inner" core, and six in the "outer" core. The resultant atom densities in these zones, determined by the ratio of "initial" to "final" volume in these zones, are enumerated in Table 4.9. The radial expansion data at the core midplane is summarized in Figure 4.12. From data describing the total axial expansion,⁽⁵⁾ zone dependent expansions were inferred by assuming that:

- Maximum axial expansion occurs at the core center. This allows the use of calculations for a reflected, one-half core. The actual maximum occurs above the center, as the thermal profile is not symmetric about the core midplane.
- The expansion along the axis can be described by a cosine centered at the midplane. While the axial expansion is not linear, it is not a cosine either, although the function does resemble the latter.
- The decrease in the total axial expansion with increasing radius can be described as a cosine centered at the core center.

TABLE 4.8. FTR III Atom Densities

<u>Material</u>	<u>Atomic Density</u>
<u>Inner Core</u>	
239 & 241 _{Pu}	0.99399998-03
240 _{Pu}	0.84999999-04
235 _U	0.99999999-05
238 _U	0.45500000-02
O	0.11752000-01
Fe	0.13463000-01
Cr	0.32590000-02
Ni	0.14120000-02
Na	0.10249000-01
<u>Outer Core</u>	
239 & 241 _{Pu}	0.14930000-02
240 _{Pu}	0.17500000-03
235 _U	0.11000000-04
238 _U	0.50820000-02
O	0.14925000-01
Fe	0.18880000-01
Cr	0.31280000-02
Ni	0.13550000-02
Na	0.87379999-02
Mo	0.43000000-03
C	0.10530000-02
<u>Radial Reflector</u>	
Fe	0.86189999-02
Cr	0.24190000-02
Ni	0.47647000-01
Na	0.66319999-02
<u>Axial Reflector</u>	
Fe	0.14802000-01
Cr	0.40430000-02
Ni	0.29866000-01
Na	0.92429999-02

FIGURE 4.10. 2DB Calculational Model in Undistorted Core R-Z Geometry

FIGURE 4.11. Calculation Model in Distorted Core
R-Z Geometry

TABLE 4.9. Zone Dependent Atom Density Modifiers

<u>Zone</u>	<u>Ring</u>	<u>Initial Volume, cm³</u>	<u>Final Volume, cm³</u>	<u>Initial/Final</u>
1	1-A	} 1716.777	1707.379	1.0055
7	1-B		1707.519	1.0054
13	1-C		7710.166	1.0039
2	2-A	} 13,734.060	13,665.740	1.0050
8	2-B		13,665.999	1.00498
14	2-C		13,684.593	1.0036
3	3-A	} 27,346.548	27,354.966	0.9997
9	3-B		27,340.201 + 0.0208	1.002
15	3-C		27,376.855 + 0.09877	1.007
4	4-A	} 40,881.360	41,100,139	0.9947
10	4-B		41,050.147	0.9959
16	4-C		40,954.782	0.9982
5	5-A	} 55,378.167	54,803.746	1.0105
11	5-B		54,867.781 + 0.0472	1.0093
17	5-C		55,066.400 + 0.2200	1.0057
6	6-A	} 68,669.976	68,474.73	1.0028
12	6-B		68,485.491	1.0027
18	6-C		68,541.860	1.0019

FIGURE 4.12. Core Midplane Radial Expansion

The assumptions also were made that:

- A core could be used with radius equal to the radial distance across 6 rows of subassemblies lined up flat to flat (65.8688 cm) rather than the average core radius. As a result, calculated core k_{eff} 's would be larger than for the FTR, but the change in k should be little different.
- Initial (unburnt) atom densities could be used rather than those representative of 30,000 MWd/t.

The calculated increase in k_{eff} produced by distortion in the core is 0.4% dk. The convergence on k in the calculations is 2 parts in 10^{-4} in k , thus introducing an uncertainty of the order of $\pm 0.03\%$ dk. However, a larger uncertainty is introduced by the use of different mesh structure. Both calculations have the same number of mesh intervals, but the distribution is varied. An estimate of this effect on k would be that the k of the distorted calculation is about 0.06% dk too high.⁽⁶⁾ This uncertainty, however, would be about $\pm 0.02\%$ dk. As a result, the reactivity increase should be stated as 0.40% dk - 0.06% dk $\pm$ 0.04% dk, or 0.34% dk $\pm$ 0.04% dk.

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5.0 CRITICAL MASS PHYSICS

CRITICALITY OF TRANSURANIUM ACTINIDES - UNDERMODERATED SYSTEMS

S. R. Bierman and E. D. Clayton

Of the 14 elements making up the actinide group, ^{239}Pu and the uranium isotopes 233 and 235 have, in the past, received the bulk of the attention with respect to criticality.

However, some of the lesser known actinides are beginning to find the application for which they are particularly adapted, and consequently are being, or will be, produced in significant quantities with respect to criticality.

Most of the applications employ the particular actinide in some high density form either as an oxide or as a metal and generally in fairly pure isotopic concentrations. However, in the production, purification, and fabrication to forms required for a given application, these actinides exist in various isotopic concentrations and are subjected to the entire spectrum of neutron moderation. Thus, material not generally considered to present a criticality problem can cause one, either by isotopic contamination with a fissile isotope or by some event which results in a shift in the neutron spectrum.

This section covers actinide isotopes that can sustain criticality at high concentrations. Specifically, the isotopes covered are ^{237}Np , used as a target material in producing ^{238}Pu which, in turn, is used as an α decay heat source, ^{241}Am , used as a neutron target for producing other actinides and for its 60 keV gamma ray, ^{244}Cm , used as a heat source and starting material for producing other actinides such as ^{252}Cf , and ^{240}Pu , present in high exposure fuels.

All of these isotopes contain an even number of neutrons in their nucleus and, characteristically, exhibit a rather sharp threshold in their fission cross sections, but with

little or no probability for sub-threshold fission. This phenomenon is illustrated in Figure 5.1 where the fission-to-absorption ratio, i.e. the fractional probability for neutron fissioning, is shown for ^{237}Np as a function of neutron energy.

At energies above about 600 keV, neutron absorption results almost entirely in fission, but a sharp decrease is observed at lower energies to a level where only about 1% or less of the absorptions result in fission. As the neutron energy spectrum is degraded into the sub-threshold region, criticality becomes increasingly more difficult to achieve.

As moderating material is added to a metal-atoms-in-water system, k_{∞} decreases steadily, as shown in Figure 5.2, from that for a pure metal at 20.45 kg/l through a value of 1 at a ^{237}Np concentration of about 12.7 kg/l. Below 12.7 kg/l, k_{∞} is less than one and criticality is no longer possible. Over the concentration range where criticality is possible, the critical radius for a bare sphere of metal atoms in water varies uniformly from a minimum of 9.3 cm (68.6 kg) for the pure metal to a maximum value approaching infinity as the minimum critical concentration of 12.7 kg/l is approached.

The effect of energy degradation also appears in the reflector savings of such a system. A good moderating material such as water returns relatively thermal neutrons to the core and its effectiveness as a reflector is limited to a few percent instead of the more commonly encountered factors of 2 or more. On the other hand, scattering materials such as steel are quite effective as reflectors, since the neutrons scattered back into the core experience little loss in energy, i.e. the energy spectrum of the reflected neutrons is, for the most part, still above the sub-threshold region for fission.

FIGURE 5.1. ^{237}Np Fission-to-Absorption Ratio

FIGURE 5.2. Criticality of Homogeneous ^{237}Np -Water Systems

These calculations for Np were made with a set of nuclear data compiled and evaluated by B. R. Leonard of BNW. The fission cross sections above 5 keV coincide with those of White, as recommended by Davey.⁽¹⁾

The calculations for Np, as with the other actinides covered in this paper, were carried out by a computational technique that has proved to be very satisfactory for ^{239}Pu systems. The method consists of averaging the nuclear data over 68 quarter-lethargy units between 10 MeV and 0.414 eV, and 0.01 eV energy intervals between 1 eV and 0.05 eV, and 0.001 eV energy intervals to zero. These are further averaged and reduced in the computer code GAMTEC-II to produce one thermal group having an upper energy of 0.683 eV, and 32 broad epithermal groups appropriately structured to define the respective cross sections as a function of energy. These multigroup constants were then used in the neutron transport theory code DTF-IV for making the criticality calculations. The epithermal grouping consisted basically of a single broad energy group over the high energy fission plateau, quarter lethargy groups over the threshold region, and the best description of the sub-threshold structuring as the remaining epithermal groups would permit.

The adequacy of this calculational model is illustrated in Figure 5.3. The circular and triangular points shown are calculated values and the squares are experimental values for ^{239}Pu -water. As can be seen, these points are located pretty close about the best-estimate-for-critical line except in the region near 1 kg/l where the diffusion and transport theory results begin to differ. However, in the higher density region above 1 kg/l, the transport theory calculations accurately predict the experimentally determined critical radii of ^{239}Pu -water systems. Consequently, this theoretical model should be adequate for calculating other undermoderated aqueous actinide systems.

FIGURE 5.3. Criticality of Homogeneous ^{239}Pu -Water Systems

One of the more interesting actinides is ^{244}Cm , an excellent heat source (2.79 W/g) with a relatively long half-life (18.1 years for α -decay). This actinide is also a logical precursor to Californium and the heavier actinides, since it is the heaviest produced in appreciable quantities during normal operation.

As with ^{237}Np , the fission cross section for ^{244}Cm is typical of the even neutron actinides in that it exhibits a fission threshold, as illustrated in Figure 5.4, with a plateau a couple of orders of magnitude above the sub-threshold level. As shown in Figure 5.5, the k_{∞} for ^{244}Cm metal atoms in water decreases uniformly with dilution to a value less than unity at a metal concentration of about 6.7 kg/l. Again, the water is seen to be ineffective as a reflector, and the steel worth is only a few percent. Data obtained from Barton⁽²⁾ of LASL indicate the critical radius for the bare metal sphere to be slightly higher at 7.88 cm, as compared with the calculated value of 7.5 cm shown here.

Although aqueous mixtures of ^{244}Cm cannot be made critical if the Cm concentration is below about 6.7 kg/l, the most practical method of producing ^{244}Cm by successive neutron capture results in the presence of some heavier fissile isotopes of Cm. Calculations by H. K. Clark⁽³⁾ of SRL have indicated that only 42 g of ^{245}Cm at 15 g/l are needed for criticality. Consequently, any criticality evaluation of a Cm system must consider the relative concentrations of both types of isotopes.

The fission-to-absorption ratio of ^{241}Am is quite similar to that of ^{237}Np and ^{244}Cm . The fission threshold of ^{241}Am is in the vicinity of about 1/2 MeV, with the sub-threshold level about an order of magnitude lower than the high energy fission plateaus (Figure 5.6). As seen in Figure 5.7, k_{∞} decreases

FIGURE 5.4. ^{244}Cm Fission-to-Absorption Ratio

FIGURE 5.5. Criticality of Homogeneous ^{244}Cm -Water Systems

FIGURE 5.6. ^{241}Am Fission-to-Absorption Ratio

FIGURE 5.7. Criticality of Homogeneous ^{241}Am -Water Systems

uniformly with concentration to a value less than one at about 7.6 kg/l. The corresponding spherical critical radii approach infinity at this concentration from a minimum value of 11.3 cm (71.5 kg) for the steel-reflected metal at a theoretical density of 11.7 kg/l. However, the production of ^{241}Am results in contamination with ^{240}Pu and ^{241}Pu , and with some ^{242}Am , ^{242}Cm , and ^{238}Pu . These actinides should be considered in any criticality evaluation of a ^{241}Am system, since their presence may not be insignificant. This consideration would be particularly applicable for one of the highly fissionable fissile actinides such as $^{242\text{m}}\text{Am}$ which, even for the theoretical density metal, has a spherical critical mass of only 3.9 kg as compared to 71.5 kg for ^{241}Am .

The results of calculations for ^{240}Pu -water systems are shown in Figure 5.8. The k_{∞} is less than unity for metal concentrations below 12.5 kg/l and, as with the other even neutron actinides, the minimum critical size occurs for the theoretical density metal. The steel reflected sphere has a minimum critical radius of 10.6 cm (96.4 kg).

The results of calculations for ^{238}Pu -water systems, shown in Figure 5.9, are similar to those results obtained for the other actinides except that the concentration range over which criticality is possible is much wider and the effectiveness of water reflection seems to increase as the system becomes more thermalized. For ^{238}Pu concentrations below about 5.1 kg/l, k_{∞} is less than unity and the critical size decreases uniformly from infinity at this concentration to a minimum spherical radius of 3.8 cm (4.5 kg) for the steel reflected theoretical density metal.

Replacement type experiments⁽⁴⁾ by Stubbins, Barton, and Lonadier have indicated that the critical mass of a bare sphere of 19.6 kg/l ^{238}Pu should be slightly less than that for

^{239}Pu of the same density. With the cross sections used and the calculational model previously described, we obtain a value of 7.2 kg for the bare ^{238}Pu metal sphere as compared to 9.6 kg for the bare ^{239}Pu . Ottewitte and Kesishnian of Atomics International reported⁽⁵⁾ a calculated value of about 8.2 kg for the bare ^{238}Pu metal sphere at the 1966 Conference on Neutron Cross Section Technology. However, because better cross sections have become available since then, we feel that the calculated value of 7.2 kg is probably a little better value. Also, their calculated value for a ^{244}Cm metal sphere, reported at the same meeting, has since been shown by Barton, et al. to be about 10% high.

The fission-to-absorption ratio for ^{238}Pu , shown in Figure 5.10, is similar to the other even neutron actinides in that it exhibits a fission threshold. However, this threshold is not well defined and there is only a factor of 2 or 3 between the plutonium in the high energy region and the values in the subthreshold region. This relatively large probability for subthreshold fission probably accounts for the increased worth of the water reflector observed in the previous slide as the ^{238}Pu -water system became more thermalized.

Numerical values of critical radii and masses are tabulated in Table 5.1 for all the actinides covered in this paper.

The validity of these calculational results is difficult to estimate because of the almost complete lack of experimental data. However, because of some experimental confirmation of the results obtained for ^{238}Pu and ^{244}Cm metal, and because of our experience with ^{239}Pu , the calculations should be sufficiently conservative to permit their use in establishing operating procedures and limits to prevent criticality.

Experimental data are needed particularly for metal-water mixtures. Also, additional calculations are needed for mixtures of actinides.

FIGURE 5.8. Criticality of Homogeneous ^{240}Pu -Water Systems

FIGURE 5.9. Criticality of Homogeneous ^{238}Pu -Water Systems

FIGURE 5.10. ^{238}Pu Fission-to-Absorption Ratio

TABLE 5.1. Calculated Critical Radii and Masses of Actinide Metals in Spherical Geometry

Actinide	g/cm ³	H/M	k _∞	Bare Critical		Water Reflected (a) Critical		Steel Reflected (a) Critical	
				Radius, cm	Mass, kg	Radius, cm	Mass, kg	Radius, cm	Mass, kg
²³⁸ Pu	19.6	0	2.8055	4.443	7.20	4.085	5.596	3.800	4.505
	11.254	1	1.8680	8.206	26.05	6.73	14.367	6.652	13.876
	7.893	2	1.4108	13.924	89.25	8.043	17.202	11.27	47.326
	7.00	2.42	1.2828	--	--	8.671	19.116	14.512	89.596
	6.078	3	1.1435	26.733	486.40	9.586	22.427	22.51	290.387
	5.2	3.73	1.0080	--	--	10.157	22.824	--	--
	5.1	3.83	0.9925	--	--	--	--	--	--
	4.941	4	0.96646	--	--	--	--	--	--
²³⁹ Pu	19.6	0	2.8901	4.90	9.66	4.0	5.25	3.874	4.45
	19.74	0	--	--	--	(4.03) (b)	5.412	--	--
²⁴⁰ Pu	19.6	0	1.6339	12.457	158.44	12.18 (c)	148.35	10.55 (c)	96.40
	19.457	0.01	1.5908	12.687	166.43	12.45	157.28	10.742	101.02
	18.905	0.05	1.4058	13.670	202.29	13.43	191.82	11.623	124.34
	18.257	0.10	1.2027	15.073	261.890	--	--	12.881	163.44
	17.653	0.15	1.0419	16.703	344.581	--	--	14.375	219.375
	17.092	0.20	0.9171	--	--	--	--	--	--
²⁴⁴ Cm	13.5	0	2.4552	7.430	23.20	7.303	22.02	6.31	14.21
	10.872	0.5	1.7440	10.667	55.27	10.466	52.21	--	--
	9.055	1.0	1.3993	15.727	147.54	15.432	139.39	--	--
	7.758	1.5	1.1747	25.813	558.92	--	--	22.817	368.025
	6.787	2.0	1.0143	102.825	30907.43	--	--	--	--
²³⁷ Np	20.45	0	1.6378	9.287	68.613	9.118	64.935	7.956	43.138
	16.6	0.3	1.3094	14.368	206.247	14.129	196.125	12.501	135.841
	13.95	0.6	1.1016	25.359	952.928	25.032	916.538	22.872	699.159
	12.50	0.82	0.9871	--	--	--	--	--	--
	8.01	2	0.6494	--	--	--	--	--	--
²⁴¹ Am	11.7	0	1.9436	13.232	113.5	12.904	105.3	11.34	71.4
	9.601	0.5	1.3369	21.404	394.4	20.934	368.9	18.848	269.2
	8.141	1.0	1.0934	44.4 (est)	2985 (est)	42.997	2710.7	--	--
	6.24	2.0	0.83	--	--	--	--	--	--
²⁴² Am	11.7	0	3.0117	5.554	8.40	4.261	3.8	4.284	3.8
	3.237	6	2.2358	--	--	--	--	--	--
	0.262	100	2.5139	--	--	--	--	--	--

(a) 20 cm reflector (6)

(b) Experimental Value

(c) Extrapolated

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