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REACTOR PHYSICS QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER 1968

FEBRUARY 1969

**AEC RESEARCH &
DEVELOPMENT REPORT**

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Reactor Technology

REACTOR PHYSICS QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER 1968

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FORWARD

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

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1.0 SUMMARY

REACTOR THEORY AND CODE DEVELOPMENT

A foil flux peaking or dipping approximation coupled with basic particle size theory methods has been used to obtain an approximate expression for correcting the thermal fission rate density measured by a special particle size fission foil to the unperturbed fuel particle fission rate density.

A one-dimensional diffusion theory program to compute resonance-shielded and collapsed-group cross sections, 1DX, and a computer code to calculate group constants from currently available cross section data, ETOX, have been written for use in fast-reactor analyses.

Major sections of the multigroup three-dimension synthesis diffusion analysis code, RSYN, have been completed and the computer strategy for solving the synthesis equations has been selected.

THERMAL REACTORS

A detailed theoretical analysis is being made of the Batch Core fueled experiments conducted in the Plutonium Recycle Critical Facility (PRCF). The theory-experiment correlations include multiplication factors, moderator level coefficient, thermal reaction rate traverses, and kinetic parameters.

Burnup calculations of the PRTR Batch Core Experiment have been made using the ZODIAC G code and the results are compared to values obtained from the ZODIAC(2+2) code. The calculated results are in agreement; however, the calculated values are in disagreement with experiment.

The series of CAF measurements using UO_2 -4 wt% PuO_2 fuel in H_2O moderated lattices has been completed. A total of seven lattices were reported, with moderator to fuel volume

ratios ranging from 1.931 to 11.580, providing a valuable body of data for use in testing current calculational methods for plutonium-enriched light-water reactor systems.

The value of the ratio of epithermal to thermal neutron capture in ^{238}U , often called " ρ^{28} ," has been measured by foil activation techniques in the Physical Constants Testing Reactor (PCTR) water tank. Relative reaction rate measurements were made with foils placed in a separable fuel rod in a 1.0 in. square pitch lattice of 0.5 in. diameter rods (1) of UO_2 -2.0 wt% PuO_2 , and (2) of UO_2 -0.9 wt% PuO_2 , moderated by light water. Methods of correcting for foil and cadmium cover perturbations are included in the analysis. An improvement in the foil counting accuracy is planned for future experiments to provide data for theory-experiment correlation.

A numerical method for predicting the converged eigenvalue of a two-group, three-dimensional diffusion theory calculation is presented. The application of this method can result in large savings in computer time. Over the range of cases used, the accuracy was within 2%.

Test plates of Phoenix fuel (Al-20 wt% Pu, approximately 21 at.% ^{240}Pu) have been irradiated in the L-51 position in MTR to an average ^{239}Pu depletion of 28.8%. The maximum ^{239}Pu depletion observed in the test assembly was 47.7%. The plates were examined in detail using gamma scanning techniques and were destructively sampled at two points to obtain plutonium isotopic data and total fissions information.

FAST REACTORS

Static reactivity calculations have been made in support of the Preliminary Safeguards Analysis Report for the Nuclear Proof Test Reactor based on the annular conical core concept.

Analyses were completed and the results compared to experimental values obtained in the FTR critical experiments carried out in the ZPR-III facility, Assemblies 51 and 52.

An estimate was made of radiation levels that might be encountered in operation of the Nuclear Proof Test Facility from handling fuel in order that these levels would be considered in the design.

A completed study indicates that synthesis techniques are not appropriate for deep penetration calculations where the characteristic of interest is not averaged over the entire system.

Calculations were made to determine the effectiveness of a reactor hot cell shield following various operating incidents up to a design basis accident.

The gamma intensity as a function of distance and shield thickness was calculated for different size pipes containing radioactive closed-loop sodium to provide a basis for routing these lines in the region above the reactor cover.

A number of analyses were made to provide instrument engineers with information required to form a basis for the conceptual design of a failed fuel element detection and location system for the FTR.

CRITICAL MASS PHYSICS

The expandable-slab critical assembly has been used for experiments on $\text{Pu}(\text{NO}_3)_4$ solutions with 18.4 wt% ^{240}Pu at concentrations of 412 and 159 g Pu/liter. Data were acquired for critical systems which were bare and laterally reflected with either water or a 1 in. thickness of Lucite. The effect of a large movable Lucite reflector on the critical height of a slab of constant thickness was found to depend on which side of the assembly it was placed. This result is attributed to

the position of the slab-tank within the hood: the walls of the hood are not equidistant from the faces of the slab and thus are not equally effective reflectors. Analysis of the data yielded extrapolation lengths and bucklings which are functions both of concentration and configuration. It is suspected that the variations in buckling are due ultimately to the spatial dependence of the neutron flux spectrum within the thin slabs.

3.0 REACTOR PHYSICS

A DIPPING AND PEAKING CORRECTION FOR PARTICLE SIZE FOILS USED IN PARTICLE FISSION RATE DENSITY MEASUREMENTS IN GRANULAR FUELED LATTICES

C. L. Bennett

In performing lattice parameter measurements by foil activation techniques, it is important to correct for the peaking or dipping of the flux in the foil with respect to the surrounding media. Such correction methods normally only apply to solid solution (or single element) foils surrounded by a nonparticulate-structured medium such as coolant, fuel, moderator, etc. One such method for disk shaped foils that is both simple to apply and reasonably accurate is that derived by Dunn.⁽¹⁾ A similar method was needed for the case when both the foils and surrounding media had particulate structure. The following derivation couples the Dunn approximation with the basic particle size methods of References 1, 2, and 3 to obtain a simple expression for correcting the thermal fission rate density measured by a special particle size foil to the unperturbed fuel particle fission rate density.

The Dunn foil flux peaking or dipping correction can be rewritten in the following form:

$$\frac{\phi_{\text{fuel}}(E)}{\phi_{\text{foil}}(E)} = 1 - \beta(E), \quad (1)$$

where $\phi_{\text{fuel}}(E)$ and $\phi_{\text{foil}}(E)$ are, respectively, the volume averaged unperturbed fuel and perturbed foil fluxes, and

$$\beta(E) \equiv [\Sigma_{\text{fuel}}(E) - \Sigma_{\text{foil}}(E)] t \ln(L/t), \quad (2)$$

where L and t are the foil diameter and thickness. For particulate structured fuel, the fuel absorption cross section used in Equation (2) is

$$\Sigma_a^{\text{fuel}}(E) = \frac{\phi(E) V_p \Sigma_a^{\text{part}}(E) + \phi(E) [1-V_p] \Sigma_a^{\text{dil}}(E)}{\phi(E) V_p + \phi(E) [1-V_p]} \quad (3)$$

where V_p is the volume fraction of particles in the fuel and dil refers to the fuel diluent. The special foil is assumed to contain the same composition particles as in the fuel, but not necessarily of the same size or at the same volume fraction as in the fuel. The foil diluent material would definitely not be fissile and will be assumed different than the fuel diluent.* The foil absorption cross section in Equation (2) can be approximated by the following expression

$$\Sigma_a^{\text{foil}}(E) \approx P'_0(E) V_p \Sigma_a^{\text{part}}(E) + [1-V_p] \Sigma_a^{\text{dil}}(E) \quad (4)$$

where $P'_0(E)$ is the average first flight escape probability for the foil particles and the prime is used to denote foil quantities.

The total fission rate of the particles in the fuel volume dV and in the energy interval dE is $\phi^{\text{part}}(E, r) \Sigma_f^{\text{part}}(E) V_p dV dE$, where $\phi^{\text{part}}(E, r)$ is the flux per unit energy in the fuel particles at energy E and position r . The fuel averaged particle fission rate density per unit energy, $f_p(E)$, then is

$$f_p(E) = \phi^{\text{part}}(E) \Sigma_f^{\text{part}}(E). \quad (5)$$

* Such particle fission foils were conceived and designed by D. F. Newman for use in a series of particle size experiments performed in the Physical Constants Test Reactor. (5)

The average thermal fission rate density in the fuel particles F_p then is just equal to the energy integral of $f_p(E)$ from zero to some thermal cutoff E_c , i.e.,

$$F_p = \int_0^{E_c} f_p(E) dE = \phi^{part} \bar{\Sigma}_f^{part} \quad (6)$$

The total fission rate of the particles in the foil volume dV' and in the energy interval dE is just the total collision rate in the foil, $\phi^{foil}(E, r) \Sigma^{foil}(E) dV' dE$; times the conditional probability that a particle^{tot} collision will result from a foil collision, $P_o^{part}(E) \Sigma^{part}(E) / \Sigma^{tot}(E)$; times the further probability that a fission will occur given a particle collision, $\Sigma_f^{part}(E) / \Sigma^{part}(E)$; which reduces to $\phi^{foil}(E, r) P_o^{part}(E) \Sigma_f^{part}(E) V' dV' dE$. The foil averaged particle fission rate density per unit energy, $f_p'(E)$, then becomes

$$f_p'(E) \approx \phi^{foil}(E) P_o^{part}(E) \Sigma_f^{part}(E) \quad (7)$$

Equation (1) can be used to eliminate the perturbed foil flux from the above, resulting in

$$f_p'(E) \approx \frac{\phi^{fuel}(E) P_o^{part}(E) \Sigma_f^{part}(E)}{1 - \beta(E)} \quad (8)$$

The average thermal fission rate density in the foil particles, F_p' , then becomes

$$F_p' \approx \phi^{fuel} \bar{\Sigma}_f^{part} \left(\frac{P_o^{part}(E)}{1 - \beta(E)} \right)_{thermal} \quad (9)$$

where

$$\left(\frac{P_o^{part}(E)}{1 - \beta(E)} \right)_{thermal} = \frac{\int_0^{E_c} \frac{P_o^{part}(E)}{1 - \beta(E)} \phi^{fuel}(E) \Sigma_f^{part}(E) dE}{\int_0^{E_c} \phi^{fuel}(E) \Sigma_f^{part}(E) dE} \quad (10)$$

and

$$\bar{\Sigma}'_{f, \text{part}} = \int_0^{E_c} \phi(E) \Sigma'_{f, \text{part}}(E) dE / \phi_{\text{fuel}} \quad (11)$$

Since both the foil and fuel particles are assumed to have the same composition, $\Sigma'_{f, \text{part}}(E) = \Sigma'_{f, \text{fuel}}(E)$. Also the average thermal fission cross sections of the foil and fuel particles are equal, i.e.,

$$\bar{\Sigma}'_{f, \text{part}} = \bar{\Sigma}'_{f, \text{fuel}} \quad , \quad (12)$$

since as defined, both are averaged over the same spectrum, that of the fuel.

Finally, from Equation (6), (9), and (12), the ratio of the unperturbed thermal fission rate density in the fuel particles to the perturbed fission rate density measured by the foil particles can be expressed by

$$\frac{F_p}{F'_p} \approx \frac{\phi_{\text{part}} / \phi_{\text{fuel}}}{\left(\frac{P'_0(E)}{1 - \beta(E)} \right)_{\text{thermal}}} \quad (13)$$

The numerator of the RHS of Equation (13), the fuel particle thermal flux depression factor, is readily obtainable from the GRANIT code, a cylindrical geometry version of THERMOS⁽⁶⁾ modified by this author to explicitly calculate the thermalization in lattices containing granular materials, i.e., one or two different types of particles embedded in a matrix of some other diluent material. The denominator of the RHS of Equation (13) could also be obtained from GRANIT after some coding modification, since most of the information necessary for its calculation is part of the normal output.

Using Equation (5) and (8), the energy dependence of the relative flux peaking or dipping between the fuel and foil particles can also be determined.

$$\frac{\phi_{\text{part}}(E)}{\phi_{\text{part}}'(E)} = \frac{f_p(E)}{f_p'(E)} \approx \frac{\phi_{\text{part}}(E) / \phi_{\text{fuel}}(E)}{P_0'(E) / (1 - \beta(E))} \quad (14)$$

The various approximations inherent in the above results may be expected to start to break down under the following conditions:

- (1) The fuel or foil particles volume fraction is large (~10% or more);
 - (2) The foil edge effects become important, i.e., $t/L \rightarrow 1$;
 - (3) The particles are much better moderators than the diluent;
 - (4) The foil or fuel become too black.
- Most of these effects are discussed in References 1 through 4.

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1DX, A ONE-DIMENSIONAL DIFFUSION THEORY CROSS SECTION GENERATING CODE

R. W. Hardie and W. W. Little, Jr.

1DX, a one-dimensional diffusion theory program designed to compute resonance-shielded and collapsed-group cross sections, has been developed at PNL for use in fast-reactor analyses. Resonance-shielded cross sections are calculated using data (infinite dilution cross sections and resonance shielding factors) in the "Russian" format. Interpolation schemes are used to compute shielding factors applicable to specific compositions. 1DX will also accept input cross sections in the "DTF" format (i.e., resonance-shielded cross sections). Collapsed-group cross sections by reactor zone are calculated using flux-weighting.

The spatial flux profiles and eigenvalue are computed by standard source-iteration techniques. Convergence is accelerated using fission source over-relaxation. Adjoint calculations are performed by transposing the scattering matrix and fission source and then inverting the group order of the input data.

Criticality searches can be performed on time absorption (α), material concentrations, region dimensions, and buckling. Alpha and k_{eff} can be used as parametric eigenvalues.

The format of the input data (e.g., cross sections, flux dumps, geometry and composition specifications) is compatible with the Los Alamos one- and two-dimensional transport codes DTF-IV and 2DF, the BNW two-dimensional diffusion theory code 2DB, and the BNW perturbation code PERT-IV. All five codes use the same input module.

All programming is in FORTRAN-IV. Variable dimensioning is used to make maximum use of the available fast memory.

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

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

Computer code ETOX has been written to calculate group constants for fast reactor calculations from currently available cross section data. The code is specifically designed to use the Evaluated Nuclear Data File (ENDF/B)⁽¹⁾ as input for microscopic cross section values. Output from this code includes punched cards in the Russian (Bondarenko)⁽²⁾ format which can be used as input to the fast reactor codes FCC-IV⁽³⁾ and 1DX.⁽⁴⁾ The output includes infinite-dilute group cross sections, inelastic scattering probability matrices, and temperature dependent self-shielding factors for arbitrary values of σ_0 (total cross section per atom).

A document describing ETOX is being prepared.

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A MULTIGROUP 3D SYNTHESIS CODE

H. S. Zwibel

RSYN is a multigroup 3D synthesis diffusion analysis code. Full 2D diffusion analysis is provided by the diffusion analysis routines in 2DB. The flux distribution in the third dimension is synthesized from a 1D spatially dependent linear combination of the two dimensional fluxes obtained by analyzing 2D sections with 2DB. Therefore, basic to the analysis is the assumption that, for the n^{th} energy group,

$$\phi_n(x,y,z) = \sum_{i=1}^M A_{ni}(z) X_{ni}(x,y) \quad ,$$

where the $X_{ni}(x,y)$ are the M 2DB solutions for the n^{th} energy for each of M two-dimensional sections of the core perpendicular to the z axis.

Applying variation methods to optimize the choice of the $A_{ni}(z)$'s generates a set of synthesis equations which have the form of 1D diffusion equations. These equations are complicated relative to the usual set by having coupling through the diffusion constants

$$-D_n \nabla^2 \phi_n \rightarrow - \sum_j D_{nij} \frac{d^2 A_{nj}(z)}{dz^2} \quad \begin{matrix} i^{\text{th}} \text{ equation} \\ n^{\text{th}} \text{ group} \end{matrix} \quad .$$

However, the same noniterative techniques used to solve the 1D equations are still applicable. The only complication is that matrices, of size MXM , must be inverted. Since M will generally be less than 10, this presents no difficulties. The coefficients of the synthesis equations are obtained by appropriate averaging of cross sections over the $X_{ni}(x,y)$.

The appropriate routines in 2DB have been modified to serve as subroutines in RSYN. Logic flow specifications, computer storage allocations, input routines, output routines, and routines for calculating coefficients for the synthesis equations are essentially completed. Computer strategies for solving the synthesis equations have been investigated for the one group case and a final selection of strategy has been made. The jobs remaining are programming and debugging of the routines for solving the synthesis equations and checking the program with test cases. Expected availability is February 1969. The major problem area is uncertainty over the success of the selected strategy for a large number of energy groups and a large number of 2D sectional cuts.

3.0 THERMAL REACTORS

THEORETICAL CORRELATION OF THE PRTR BATCH CORE EXPERIMENT -

D. L. Prezbindowski and M. A. Mannan*

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

Some preliminary calculated results of k_{eff} , fuel element worths, boron reactivity coefficient, and intracell flux traverses were presented in the last quarterly.⁽⁷⁾ This report presents results of theory-experiment correlations of k_{eff} , β/λ , moderator level reactivity coefficients and intracell flux traverses.

Effective Multiplication Factor

The effective multiplication factors, k_{eff} , of the 7, 13, and 28-element core with 0.93, 4.64, and 11.25 wppm ^{10}B in the moderator have been calculated using the HAMMER, HRG-THERMOS-HFN (HTH), and HRG-THERMOS-EXTERMINATOR codes. Three different energy group structures have been used for the HTH System to evaluate the effect of energy groups on k_{eff} . The results are shown in Table 3.1. It may be observed that the difference

* On assignment to PNL under an International Atomic Energy Agency Fellowship

TABLE 3.1. Comparison of Experimental and Calculated k_{eff}

	NUMBER OF ENERGY GROUPS	EFFECTIVE MULTIPLICATION FACTOR		
		7 ELEMENT 0.93 wppm ^{10}B	13 ELEMENT 4.46 wppm ^{10}B	28 ELEMENT 11.25 wppm ^{10}B
EXPERIMENT	--	1.009	1.006	1.002
HAMMER (1-D)	4	1.0248	1.0208	1.0093
HTH (1-D)	4	--	1.0222	--
	6	1.0225	1.0193	1.0110
	17	1.0218	1.0190	--
EXTERMINATOR (2-D)	6	1.0297	1.0240	1.0133

between the experimental and calculated k_{eff} decreases with the increase of core size.

Moderator Level Coefficient

The moderator level coefficients of the 13-element core loading have been calculated using EXTERMINATOR. The comparison of the experimental and the calculated results is shown in Figure 3.1. The calculated values are consistently higher than the experiment. This may be due to the fact that the scattering material above the moderator level was neglected in the calculation.

Thermal Reaction Rate Traverse

It was reported in the Plutonium Utilization Program Technical Activities Quarterly Report for June, July, and August 1968⁽⁷⁾ that the sources of discrepancy between the calculated and experimental Lu traverse would be investigated. As a result of the investigations the experimental ^{176}Lu reaction rate has been corrected for the flux depression

FIGURE 3.1. Moderator Level Coefficient of Reactivity for 13 Element Core

and peaking in the foil. The calculated reaction rate has been corrected for the curvature of the flux in the core. White boundary condition was used in this calculation rather than the reflecting boundary condition. The correlation of the experiment is very good. The result is shown in Figure 3.2.

Kinetic Parameters

The kinetic parameters of the PRCF 7 and 13-element cores with 0.93 and 4.64 wppm ^{10}B , respectively, were calculated using HRG-THERMOS and HFN-KINPAR codes. The effective delayed neutron fraction β , and the average neutron lifetime λ were calculated using the 6 and 17 broad energy group structures shown in Table 3.2.

FIGURE 3.2. PRCF Fuel Cell Activation Traverse

The six group structure was chosen to accurately represent the energy of the delayed neutrons which fall in Group 2.

The calculated results are compared with experimental values in Table 3.3.

It can be seen from Table 3.3 that the calculated values of β/ℓ are about 10 to 15% lower than the experimental values. From the table it does not appear that the 17 group $\hat{\beta}/\ell$ calculation offers any substantial improvement over the 6 group calculation.

TABLE 3.2. *Six and Seventeen Energy Group Structures*

<u>17 Group No.</u>	<u>Upper E, Ev</u>	<u>Upper E, Ev</u>	<u>6 Group No.</u>
1	1.0 +7	1.0 +7	1
2	2.23 +6		
3	1.05 +6		
4	4.98 +5	4.98 +5	2
5	2.35 +5	2.35 +5	3
6	1.11 +5		
7	5.25 +4		
8	2.48 +4		
9	1.17 +4	1.17 +4	4
10	5.53 +3		
11	2.61 +3		
12	1.23 +3		
13	2.75 +2		
14	6.14 +1		
15	1.32 +1		
16	3.06	3.06	5
17	0.683	0.683	6

TABLE 3.3. *Comparison of Calculated and Experimental β/ℓ*

<u>Boron-10, wppm</u>	<u>Fuel Elements</u>	<u>Calculated</u>		<u>Experimental</u>
		<u>6 Group</u>	<u>17 Group</u>	
0.93	7	--	4.42	5.0 $\pm$ 0.3
4.64	13	8.24	8.17	9.4 $\pm$ 0.5
11.25	28	12.38	--	12.9 $\pm$ 0.7

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THERMOS MODIFICATION

D. L. Prezbindowski and W. L. Purcell

The capabilities of the THERMOS Code are currently being extended in the areas of calculation of grain effects, multi-thermal group cross-section edits, and number of space and energy groups available. The extension of the number of

available space and energy groups is being accomplished by storing the space and energy transfer matrices on drum rather than in core. Every time the space or energy transfer matrices are required in the program they must be read from and written back to drum. Subroutines YOYO, YOTRI, and TRIYO have been written and debugged for this purpose. In order to minimize the additional computer time required for reading and writing to drum, a version of THERMOS has been written and debugged which is logically equivalent to the original version, but the coding has been rearranged to minimize the number of times that the drum must be accessed. The coding and testing of an extended space and energy point THERMOS utilizing drum storage is proceeding.

PRTR BATCH CORE BURNUP CALCULATIONS

U. P. Jenquin, T. M. Traver, and L. D. Williams

The results of previous burnup calculations⁽¹⁾ for the PRTR Batch Core Experiment⁽²⁾ were obtained with the ZODIAC(2+2)⁽³⁾ code. The calculations were repeated using the improved burnup code ZODIAC G⁽⁴⁾ to assess the differences in calculated values relative to measured values. The most significant difference between the two burnup codes is that the THERMOS⁽⁵⁾ code is utilized in ZODIAC G whereas the TEMPEST⁽⁶⁾ code is utilized in ZODIAC(2+2) for the thermalization calculation.

The representation of the core assumed for the calculation is the same as for the previous calculation except for the central position where the FERTF water-cooled loop⁽⁷⁾ and element replace a Batch Core Element⁽⁸⁾ (UO_2 -2 wt% PuO_2). Surrounding the FERTF element are 73 Batch Core elements represented by three burnable regions. On the edge of the core are 7 irradiated mixed oxide fuel elements which are assumed to represent the various low enrichment UO_2 and UO_2 - PuO_2 fuel elements being test irradiated.

The initial boron-free excess reactivity available for burnup is calculated using the ZODIAC G code to be 1.6% $\Delta k/k$ less than the estimate from ZODIAC(2+2)⁽¹⁾ and 7% $\Delta k/k$ less than the measured value.⁽¹⁾

The FERTF element is worth -1.2% $\Delta k/k$ relative to a Batch Core element. Hence, the excess reactivity available for burnup is essentially in agreement between the two codes. The reasons for the discrepancy between the calculated and measured multiplications are being investigated.

The core lifetime is calculated to be 26,000 MWd using ZODIAC G, at which time the 55-element Batch Core will contain an average exposure of 12,300 MWd/MTM. This can be compared to the ZODIAC(2+2) prediction of 11,300 MWd/MTM⁽¹⁾ and the experimentally predicted value of about 11,800 MWd/MTM.⁽¹⁾ The magnitude of the agreement between the calculated and experimentally predicted value is the result of compensation between inaccurate calculations of initial excess reactivity and reactivity loss rate with exposure. The boron content necessary to control the excess reactivity as a function of burnup as calculated with ZODIAC G is shown in Figure 3.3.

Single element (no leakage) burnup calculations were done for Batch Core fuel and UO_2 -4 wt% PuO_2 fuel to obtain estimates of isotopic concentrations, macroscopic cross sections, and reactivity worths at low, intermediate, and high exposures.

FIGURE 3.3. PRTR Boron Content of the Full Core at Operating Conditions Versus Average Exposure on the 55 Element Batch Core

The UO_2 -4 wt% PuO_2 fuel will probably be used to drive the Batch Core fuel to high exposure. The plutonium composition of this fuel was taken as 75.6/18.1/5.1/1.2 at.% Pu-239/240/241/242.

The calculated variation in reactivity of the two fuels as a function of exposure is shown in Figure 3.4. To irradiate a Batch Core fuel element to an average exposure of 35,000 MWd/MTM will require a large amount of supplemental reactivity. The macroscopic cross sections have been used for reactivity calculations to determine the necessary fuel shuffling and quantity of fuel required to drive 12 of the Batch Core elements to high exposures.⁽⁹⁾

The isotopic variation with exposure is shown in Figures 3.5 and 3.6 for the UO_2 -2 wt% PuO_2 fuel, and Figures 3.7 and 3.8 for the UO_2 -4 wt% PuO_2 fuel. The data plotted are the ratio of the isotopic concentration to the initial elemental (i.e., U and Pu) concentration at various exposures. For purposes of material accountability estimates, the data have also been fit to the equation:

$$\frac{N_x}{N} = A(0) + A(1)y + A(2)y^2 + A(3)y^3 + A(4)y^4$$

with program LEARN, a simplified version of program LIKELY.⁽¹⁰⁾ N_x/N is the isotopic concentration divided by the initial concentration of the element, y is the exposure in megawatt days per tonne of metal (MWd/MTM), and $A(i)$ are the fitted coefficients. For isotopes with little curvature, the higher order coefficients have been set to zero in the fit. The resulting coefficients are tabulated in Table 3.7.

The reasons for the disagreement between measured and calculated values of the initial excess reactivity and reactivity loss rate with exposure are not currently known. A possible reason is an inaccurate calculation of resonance absorption rates caused by having to use a guesstimated spatial

FIGURE 3.4. Variation of Fuel Reactivity with Exposure

self shielding factor for the 19-rod cluster. The relative effect of overestimating the absorptions occurring in ^{238}U in the cluster would be to underpredict k_{eff} initially and overpredict ^{239}Pu buildup which in turn would account for the smaller slope of k_{eff} versus exposure. Additional studies using the Battelle Monte Carlo Code will help resolve the cluster geometry effects. The detailed data to be obtained from the Batch Core Experiment (e.g., isotopics, power distributions, etc.) will be needed to firmly resolve the problem.

FIGURE 3.5. Variation of Uranium Isotopic Concentrations Versus Burnup for UO_2 -2 wt% PuO_2 Fuel

Neg 0684509-4

FIGURE 3.6. Variation of Plutonium Isotopic Concentrations Versus Burnup for UO_2 -2 wt% PuO_2 Fuel

Neg 0684509-5

FIGURE 3.7. Variation of Uranium Isotopic Concentrations Versus Burnup for UO_2 -4 wt% PuO_2 Fuel

FIGURE 3.8. Variation of Plutonium Isotopic Concentrations Versus Burnup for UO_2 -4 wt% PuO_2 Fuel

TABLE 3.7. *Fitted Coefficients for Isotopic Variation as a Function of Burnup*

$$\frac{N_x}{N} = A(0) + A(1)y + A(2)y^2 + A(3)y^3 + A(4)y^4$$

y = Exposure (MWd/MTM)

N = Element Concentration at no Exposure

N_x	<u>UO₂-2 wt% PuO₂</u>				
	<u>A(0)</u>	<u>A(1)</u>	<u>A(2)</u>	<u>A(3)</u>	<u>A(4)</u>
²³⁵ U	7.200-3	-2.561-7	1.379-12	2.066-17	0.0
²³⁶ U	0.0	5.053-8	-6.463-13	0.0	0.0
²³⁸ U	9.978-1	-4.918-7	-5.705-12	0.0	0.0
²³⁹ Pu	9.0514-1	-3.590-5	5.563-10	-2.156-15	0.0
²⁴⁰ Pu	8.154-2	1.471-5	-4.018-10	3.037-15	0.0
²⁴¹ Pu	1.265-2	5.160-6	-1.781-11	-3.840-15	5.281-20
²⁴² Pu	6.60-4	1.547-7	5.949-11	3.769-16	-1.740-20

N_x	<u>UO₂-4 wt% PuO₂</u>				
	<u>A(0)</u>	<u>A(1)</u>	<u>A(2)</u>	<u>A(3)</u>	<u>A(4)</u>
²³⁵ U	7.200-3	-1.687-7	4.129-13	9.060-18	0.0
²³⁶ U	0.0	3.580-8	-3.236-13	0.0	0.0
²³⁸ U	9.9280-1	-4.154-7	-3.325-12	0.0	0.0
²³⁹ Pu	7.5611-1	-2.015-5	1.539-10	2.559-16	0.0
²⁴⁰ Pu	1.8145-1	5.072-6	-1.099-10	2.558-16	0.0
²⁴¹ Pu	5.089-2	2.118-6	-3.369-12	-1.122-15	1.044-20
²⁴² Pu	1.155-2	4.107-7	1.281-11	1.688-16	-3.660-21

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APPROACH-TO-CRITICAL EXPERIMENTS WITH UO_2 -4 wt% PuO_2 - H_2O LATTICES

W. P. Stinson and J. H. Lauby

A series of subcritical experiments to determine critical masses and bucklings of UO_2 -4 wt% PuO_2 - H_2O lattices has been completed. These experiments are part of a series of experiments that utilize fuels having different plutonium enrichments and various concentrations of the isotope ^{240}Pu . These data provide a rigorous test of the analytical methods used in calculating parameters of light water moderated systems which contain plutonium.

The experiments were conducted in the Critical Approach Facility (CAF)⁽¹⁾ with hexagonal lattices that are reflected with an effectively infinite thickness of H_2O in all directions. The lattice plates are constructed from 0.75 in. thick lucite. A pair of identical lattice plates, spaced vertically one-fourth the length of a fuel rod from each end of the core, is used to support the fuel rods in a vertical position. The weight of the fuel rods is supported by a 0.75 in. thick lucite plate. A Be-Pu neutron source containing 240 g of plutonium is positioned on the vertical axis of the core, 2 1/2 in. beneath the lower end of the core. Three BF_3 proportional counters which are placed in the reflector and spaced at about 120° intervals around the core are used to collect the experimental data. A BF_3 proportional counter placed in the geometrical center of the core (replacing a fuel rod) is used in a few of the experiments for correlation purposes.

The UO_2 -4 wt% PuO_2 fuel is vibrationally compacted into tubes of Zircaloy-2 whose wall thicknesses are 0.0335 in. and whose inside diameters are 0.4975 in. Each tube contains 1084.5 g of UO_2 - PuO_2 in a height of 36.0 in. The resulting average fuel density is 9.46 g/cm^3 . The isotopic composition of the fuel is natural uranium and plutonium with 0.28/75.38/18.10/5.08/1.15 at% of $^{238}\text{Pu}/^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$, respectively.

The experiments were conducted using critical approach and exponential measurement techniques. The relative multiplication of the loadings were measured as the fuel is added up to 96% of the critical mass with an infinite reflector on the top and also with the moderator level 7 in. below the top of the fuel. Data for a clean core is obtained by removing the control and safety rod thimbles, replacing the center counter by a fuel rod, and unloading the fuel in appropriate increments. The critical number of fuel rods is obtained by fitting a straight line, by the method of least squares, to $NI_0(AR)/\phi$ plotted against the number of fuel rods, where

I_0 = A modified Bessel function of the first kind

A = Normalizing factor

R = Core radius = $\ell \sqrt{\frac{N\sqrt{3}}{2\pi}}$

ϕ = Neutron flux as detected by the 3 BF_3 proportional counter and averaged

ℓ = Lattice spacing

N = Number of fuel rods in the core.

Data for the exponential experiments were taken by removing the central fuel rod and obtaining a neutron flux traverse with a BF_3 proportional counter along the vertical axis. Similar traverses with indium foils were taken on some of the cores for comparison purposes.

The material bucklings are computed from the relaxation lengths obtained from the exponential data and are equated to the geometrical bucklings for the critical loadings using the equation

$$B_m^2 = B_g^2 = \left(\frac{J_0}{R_e + \lambda} \right)^2 - \gamma^2 = \left(\frac{J_0}{R_c + \lambda} \right)^2 + \left(\frac{\pi}{H + 2\lambda} \right)^2$$

where

B_m^2 = Material buckling

B_g^2 = Geometrical buckling

γ = Relaxation length of neutron flux on the vertical axis

H = Length of fissionable material in the fuel rods

R_e = Radius of the bare core during exponential measurement

R_c = Radius of the bare core when critical.

Values of reflector savings and extrapolation length (λ) are calculated by an iteration process using the preceeding equation. It is assumed that the value of λ is the same in the axial and radial directions.

The results obtained are summarized in Table 3.8. The error on the critical number of rods results from the goodness of fit of the approach data fitted to a straight line by the method of least squares. The error on λ is caused by the errors on the critical number of rods and on the relaxation length. The latter error is calculated by a generalized least squares computer program (LEARN) which fits the data from the exponential experiments to a hyperbolic sine function.

The agreement between the pre-experiment calculations⁽²⁾ and the measurements is fairly good, with the calculations being conservative.

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TABLE 3.8. Results of Reflector Savings and Extrapolation Length (λ) Calculations

Lattice	Maximum Number of Fuel Rods in Loading		Number Fuel Rods at Critical		Moderator to fuel Volume Ratio	Atom Ratio H/Pu	B^2, m^{-2}	λ, cm	γ, cm^{-1}
	Exponential	Critical Approach	Experiment	Calculated ⁽²⁾					
0.85	126	241	252.6 $\pm$ 0.5	246.57	1.931	152.8	94.7 $\pm$ 0.3	7.89 $\pm$ 0.04	-0.0637 $\pm$ 0.0002
0.93	90	169	178.9 $\pm$ 0.2	177.68	2.566	203.3	107.9 $\pm$ 0.3	7.55 $\pm$ 0.03	-0.0678 $\pm$ 0.0002
1.05	72	130	138.9 $\pm$ 0.4	132.78	3.624	288.7	108.7 $\pm$ 0.5	7.54 $\pm$ 0.04	-0.0658 $\pm$ 0.0002
1.143	60	118	122.4 $\pm$ 0.1	118.10	4.533	413.9	107.9 $\pm$ 0.1	7.29 $\pm$ 0.01	-0.0700 $\pm$ 0.0001
1.386	60	112	123.6 $\pm$ 0.7	117.72	7.271	578.1	88.4 $\pm$ 0.8	6.46 $\pm$ 0.06	-0.0665 $\pm$ 0.0001
1.60	90	163	180.9 $\pm$ 2.1	169.80	10.874	804.8	59.5 $\pm$ 1.1	5.30 $\pm$ 0.17	-0.0533 $\pm$ 0.0002
1.70	138	259	271.9 $\pm$ 0.5	249.22	11.580	922.0	41.1 $\pm$ 0.2	5.36 $\pm$ 0.09	-0.0388 $\pm$ 0.0003

MEASUREMENT OF ρ^{28} IN $\text{UO}_2\text{-PuO}_2$ RODS MODERATED BY LIGHT WATER -
D. F. Newman

Introduction

The value of the ratio of epithermal to thermal neutron capture in ^{238}U , often called " ρ^{28} ," has been measured in the Physical Constants Testing Reactor (PCTR) Water Tank.⁽¹⁾ Relative reaction rates were obtained from foil activation measurements in a 1.0 in. square pitch lattice of 0.5 in. diameter rods of (1) $\text{UO}_2\text{-2.0 wt\% PuO}_2$, and (2) of $\text{UO}_2\text{-0.9 wt\% PuO}_2$, moderated by light water.^(2,3) Values of ρ^{28} obtained from both bare and cadmium covered ^{239}Np activities induced in uranium foils, irradiated in the fuel rods, have been compared with values of ρ^{28} obtained from bare ^{239}Np activities and subcadmium copper activities using a substitution technique to eliminate the perturbation caused by the insertion of cadmium in the rod.

Results

Experimental values for the bare specific activity of ^{238}U captures in a fuel rod were measured, with 2 to 3% uncertainty, relative to the subcadmium specific activity of ^{238}U capture in the PCTR thermal column. Determination of the measured values for the epithermal specific activity of ^{238}U capture in a fuel rod, relative to the subcadmium specific activity of ^{238}U capture in the thermal column, must include a correction to the measured values for the perturbation caused by the insertion of cadmium in the rod. Measured subcadmium specific activities in a fuel rod require a correction for thermal flux peaking in the foils used for the activation measurements. Similarly, subcadmium specific activities in the thermal column require a correction for thermal flux dipping in the activation foils. The value for ρ^{28} should be independent of any effects due to the insertion

of foils (both bare and cadmium covered) such as flux enhancement, flux depression, differences in resonance shielding, and neutron source depression.

Elimination of the differences in resonance self-shielding and the neutron source depression caused by the insertion of cadmium in the fuel rod can be accomplished by the utilization of copper activation data.

The subcadmium specific activity of ^{238}U capture in the fuel rod relative to that in the thermal column is inferred from the copper activation data. The epicadmium specific activity of ^{238}U capture measured in the thermal column can be obtained from activation measurements with bare uranium foils, by subtracting the relative subcadmium specific activity of ^{238}U capture (inferred from copper data). Thus, the relative subcadmium and epithermal reaction rates for ^{238}U capture can be obtained without the insertion of cadmium in the fuel rod containing uranium foils. However, cadmium covered uranium foils were inserted in the fuel rod in a separate irradiation from those used to measure ρ^{28} so the effect of the insertion of cadmium could be evaluated. Results of these measurements are listed in Table 3.9.

The large uncertainty ($\sim 6\%$) in the measured values for ρ^{28} is primarily due to the 2 to 3% uncertainty in the bare specific activity of ^{238}U capture in a fuel rod relative to the subcadmium specific activity of ^{238}U capture in the thermal column. In these first experiments, no special attempt was made to improve the accuracy of the uranium capture activity measurements. However, these results for ρ^{28} do not affect the results reported⁽³⁾ for k_{∞} and $\bar{\alpha}^{49}$, since the overall neutron balance in the lattice was analyzed in terms of total reaction rates obtained from bare foil activation measurements. An improvement in the accuracy for the values of ρ^{28} can be obtained by use of improved uranium foil counting techniques in another experiment.

TABLE 3.9. Calculation of ρ^{28}

	UO ₂ -2.0 wt% PuO ₂ Rods	UO ₂ -0.9 wt% PuO ₂ Rods
$\left(\frac{SA_{rod}^{Cu}}{SA_{column}^{Cu}} \right)_{sub-Cd}$	6.41 ± 0.03	6.84 ± 0.03
$\left(\frac{SA_{rod}^{238}}{SA_{column}^{238}} \right)_{sub-Cd}$ (Substitution Method)	6.40 ± 0.03	6.83 ± 0.03
$\left(\frac{g_{column}^{238}}{g_{rod}^{238}} \right)_{THERMOS \text{ calculation}}$	0.998079	0.998943
$\frac{(SA_{rod}^{238})_{Bare}}{238}$		
$(SA_{column}^{238})_{sub-Cd}$	15.05 ± 0.50	11.74 ± 0.25
$\frac{(SA_{rod}^{238})_{epi-Cd}}{238}$ (Substitution Method)		
$(SA_{column}^{238})_{sub-Cd}$	8.65 ± 0.51	4.91 ± 0.25
ρ^{28}	1.35 ± 0.08	0.72 ± 0.04
ρ^{28}	0.917 ± 0.006	0.910 ± 0.006
$\frac{(SA_{rod}^{238})_{epi-Cd}}{238}$ (Cd covered foils)		
$(SA_{column}^{238})_{sub-Cd}$	9.32 ± 0.45	4.67 ± 0.17
$\frac{(SA_{rod}^{238})_{epi-Cd}}{238}$ (Cd covered foils)		
$(SA_{rod}^{238})_{epi-Cd}$ (Substitution Method)	1.077 ± 0.080	0.952 ± 0.060

Determination of ρ^{28} from Activation Measurements

The value of the ratio of epithermal to thermal neutron capture in ^{238}U , referred to as ρ^{28} , was determined from the bare specific activities of ^{238}U capture in the fuel relative to the subcadmium specific activity in the thermal column, and subcadmium copper relative activities. Since thermal neutron capture in ^{238}U has a slightly non-1/v cross section energy dependence, THERMOS⁽⁴⁾ calculated non-1/v factors for ^{238}U were used to infer the subcadmium ^{238}U capture relative activities from subcadmium copper relative activities from the equation:

$$\left(\frac{\text{SA}_{\text{rod}}^{238}}{\text{SA}_{\text{column}}^{238}} \right)_{\text{sub-Cd}} = \left(\frac{\text{SA}_{\text{rod}}^{\text{Cu}}}{\text{SA}_{\text{column}}^{\text{Cu}}} \right)_{\text{sub-Cd}} \left(\frac{g_{\text{column}}^{238}}{g_{\text{rod}}^{238}} \right)_{\text{THERMOS}} \quad (1)$$

where,

$$g_i^{238} = \frac{(\bar{\sigma}_c^{238})_i}{\left(\frac{\bar{\sigma}_{1/v}}{\bar{\sigma}_{o1/v}} \right)_i \sigma_{c_o}^{238}} \quad (2)$$

and the cross section averages are taken from transport theory as thermal spectrum averages over the region i (i.e., fuel rod region or PCTR thermal column region).

Corrections for thermal flux dipping and peaking in activation foils,⁽⁵⁾ listed in Table 3.10, were calculated using the equation:

$$\frac{A_{\text{foil}} - A_o}{A_{\text{foil}}} = t \left(\sum_{\text{medium}} a - \sum_{\text{foil}} a \right) \ln(L/t) \quad (3)$$

where,

t = foil thickness, cm

L = foil diameter, cm

$\sum_a$ medium = thermal spectrum averaged macroscopic absorption cross section for the medium in which the foil is placed

$\sum_a$ foil = thermal spectrum averaged macroscopic absorption cross section of the foil inserted in the medium

and the region averaged foil, fuel, and moderator cross sections were calculated by THERMOS.

The epicadmium specific activity of ^{238}U capture in the fuel relative to the subcadmium specific activity in the thermal column was obtained from the measured bare specific activity of ^{238}U capture in the fuel relative to the subcadmium specific activity in the thermal column. This ratio was obtained from Equation (1) by subtracting the relative subcadmium ^{238}U specific activity after corrections for thermal activation enhancement and depression were applied to the measured activities.

$$\frac{(SA_{\text{rod}}^{238})_{\text{epi-Cd}}}{(SA_{\text{column}}^{238})_{\text{sub-Cd}}} = \frac{(SA_{\text{rod}}^{238})_{\text{bare}}}{(SA_{\text{column}}^{238})_{\text{sub-Cd}}} - \left(\frac{SA_{\text{rod}}^{238}}{SA_{\text{column}}^{238}} \right)_{\text{sub-Cd}} \quad (4)$$

TABLE 3.10. Corrections for Thermal Flux Dipping and Peaking in Activation Foils

<u>1/2 in. Diameter Foil Location</u>	<u>Foil Type</u>	<u>Thermal Activation Correction Factor (A_o/A_{foil})</u>
Thermal Column	0.005 in. Cu	1.0167
Thermal Column	0.010 in. Natural U	1.0261
2.0 wt% PuO_2 Rod	0.005 in. Cu	0.9048
2.0 wt% PuO_2 Rod	0.010 in. Natural U	0.9183
0.9 wt% PuO_2 Rod	0.005 in. Cu	0.9724
0.9 wt% PuO_2 Rod	0.010 in. Natural U	0.9776

The value of ρ^{28} was then determined from the ratio of epicadmium activities as measured in the rod to the subcadmium activities as measured in the thermal column, Equation (4), and the ratio of subcadmium activities from Equation (1).

$$\rho^{28} = \frac{1}{\text{CdR}^{238}-1} = \frac{\frac{(\text{SA}_{\text{rod}}^{238})_{\text{epi-Cd}}}{(\text{SA}_{\text{column}}^{238})_{\text{sub-Cd}}}}{\left(\frac{\text{SA}_{\text{rod}}^{238}}{\text{SA}_{\text{column}}^{238}} \right)_{\text{sub-Cd}}} = \frac{(\text{SA}_{\text{rod}}^{238})_{\text{epi-Cd}}}{(\text{SA}_{\text{rod}}^{238})_{\text{sub-Cd}}} \quad (5)$$

Determination of the ^{238}U Resonance Escape Probability, p^{28}

A simple formula is obtained if all captures except ^{238}U resonance capture above the cadmium cutoff energy and neutron leakage are assumed negligible, then the resonance escape probability, p , can be expressed as follows:

$$p^{28} = \frac{\text{thermal absorptions}}{\text{total absorptions}} = \frac{\text{thermal abs.}}{\text{thermal abs.} + ^{238}\text{U resonance abs.}} \quad (6)$$

or,

$$p^{28} = \frac{(^{238}\text{U thermal abs.})(1/fG)}{(U^{238} \text{ thermal abs.})(1/fG) + ^{238}\text{U epithermal abs.}} \quad (7)$$

Dividing through both numerator and denominator of Equation (7) by $(^{238}\text{U thermal absorptions})(1/fG)$, one obtains:

$$p^{28} = \frac{1}{1 + \rho_{28} fG}, \quad (8)$$

where,

f = the thermal utilization

G = the ratio of the subcadmium ^{238}U capture to the sum of the subcadmium absorptions in the fuel;

ρ_{28} = the ratio of the epicadmium ^{238}U capture rate to the subcadmium ^{238}U capture rate.

This simple relation, in spite of its severe approximations, is quite useful for making comparisons of measurements made in similar systems.

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A METHOD FOR EXTRAPOLATING k_{eff} FOR WHIRLAWAY CALCULATIONS

H. B. Breaux

Introduction

Many reactor physics problems require multidimensional multienergy computer codes for solution. These calculations, by their very nature, are expensive to carry out. Frequently a cheaper but less accurate method is adequate: for instance, when one is adjusting the control rods to a critical bank height. In such a case, an approximation to the accurate version is desired so that once the approximate position has been estimated, the accurate calculation can be run in all its glory to provide a solid answer. There are other instances where less accurate answers to complex problems are satisfactory. As has been mentioned, streamlining the method or base code is one way, but this is time consuming and often very difficult because the numerics are not stable under highly accelerated convergence.

A second approach is described here. Briefly, the base code is run for a relatively short time, and a final answer is predicted using some integer values and a simple fitting technique.

Technique

In attempting to predict k_{eff} with fair accuracy while using only a small fraction of the required computer time for the WRLPWR⁽¹⁾ code, it was desirable to produce an equation which was for the most part linearly dependent upon k_{eff} . Such an equation can be obtained from the following relationship:

$$C_n = \frac{I_n F_n}{I_a F_a} \frac{K_n - K_{(n-10)}}{K_a} , \quad (1)$$

where I represents the iteration number, F represents the corresponding flux convergence, and K the corresponding eigenvalue. The subscript a is used to indicate that these are a selected set of values which correspond to a chosen iteration number. This choice of a is restricted only by the condition that the flux convergence has stopped oscillating. This set of subscript values for a remain constant for various C_n values. The subscript n is used to indicate that the values are variables. In selecting values for n it is desirable to let n correspond to the iteration numbers and all other n th values be those corresponding to the n th iteration choices. It is also desirable to let the values of $n > a$.

Using Equation (1), one calculates a few values for C_n . A plot of these values of C_n versus their corresponding K_n reveals that the function is practically linear. For practical purposes a linear equation of the form

$$C_n = MK_n + B \quad (2)$$

may be written, where M is the slope of the line ($\Delta C_n / \Delta K_n$), and B is the constant which may be determined by using three previously obtained values. By taking one of the values calculated for C_n [from Equation (1)], its corresponding value for K_n , and the previously determined slope M and substituting these values into Equation (2), one may easily determine, B .

After having obtained values for Equation (2) one then applies the condition that as k_{eff} converges the $K_n - K_{(n-10)}$ approaches zero. Applying this condition to Equation (1) readily shows that C_n must also approach zero. Thus, in seeking to determine a good approximation of the converged value of k_{eff} , one simply determines the K intercept of Equation (2) by setting $C_n = 0$.

Analytical Procedure

The use of the above relationship is illustrated by the following application involving a Phoenix fuel loading of an MTR core mockup. Several of these calculations for a WRLPWR run were made with the rods in their out position in order to see if the results were dependent upon the chosen a sub-scripted values. The results showed that there were very slight differences of the order of 10^{-4} . Even though the three plots of C_n versus K_n had different shapes, they all had a tendency to converge upon the same K intercept. This is illustrated if one graphs Tables 3.8A through 3.8C with extrapolated sections in the region where K approaches convergence.

TABLE 3.8A. *Results of WRLPWR for Rods in Out Position*

$$I_a = 48; F_a = 3.9501 \times 10^{-2}; K_a = 1.107917$$

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
60	3.6901-02	0.003866	1.112764	4.074691-03
70	3.2862-02	0.002626	1.11539	2.8756-03
80	2.6859-02	0.00186	1.11725	1.90255-03
100	1.4288-02	0.001076	1.119707	7.3186-04
120	6.7468-03	0.000724	1.121299	2.7904-04
140	5.3447-03	0.000531	1.122445	1.891426-04
160	2.5560-03	0.000279	1.123186	5.43161-05
179	2.2976-03	0.000215	1.123615	4.2093-05

The graphs of Tables 3.8A through 3.8C are shown in Figure 3.9. They adequately illustrate that the predicted converged value for k_{eff} is independent of the chosen set of a subscripted values.

The graphic technique yields a final k_{eff} of 1.121 in 90 iterations, while the code converges to 1.1236 in 180 iterations. This represents a savings of 1 hr and 95 sec of

TABLE 3.8B. Results of WRLPWR for Rods in Out Position

$$I_a = 60; F_a = 3.6901 \times 10^{-2}; K_a = 1.112764$$

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
70	3.2862-02	0.002626	1.11539	2.4518-03
80	2.6859-02	0.00186	1.11725	1.62218-03
100	1.4288-02	0.001076	1.119707	6.2053-04
120	6.7468-02	0.000724	1.121299	2.379168-04
140	5.3447-03	0.000531	1.122445	1.6127-04
160	2.5560-03	0.000279	1.123186	4.63119-05
179	2.2976-03	0.000215	1.123615	3.58900-05

TABLE 3.8C. Results of WRLPWR for Rods in Out Position

$$I_a = 30; F_a = 4.7508-02; K_a = 1.094342$$

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
70	3.2862-02	0.002626	1.11539	3.87298-03
80	2.6859-02	0.00186	1.11725	2.5622-03
100	1.4288-02	0.001076	1.119707	9.85695-04
120	6.7468-03	0.000724	1.21219	3.7581-04
140	5.3447-03	0.000531	1.122445	2.547445-04
160	2.5560-03	0.000279	1.123186	7.3155-05
179	2.2976-03	0.000215	1.1123615	5.66924-05

computer time for this problem. For the case with the rods at critical in Table 3.9A a savings of 2 hr 19 min 20 sec would have been made using this technique.

After having been satisfied that this linear function was not just a coincidence which occurred for the particular set of a 's chosen, it was decided to test its validity on WRLPWR jobs with different structural conditions. There were three cases whose data is listed in Tables 3.9A through 3.9C. The first case deals with a complete job with the rods at critical position, while 3.9B and 3.9C deal with two short restarts.

FIGURE 3.9. Graph of Rods Out of Position

TABLE 3.9A. *Results of WRLPWR for Rods at Critical*

$$I_a = 30; F_a = 2.0560-01; K_a = 8.282165$$

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
50	1.0636-01	0.03637	0.9271801	7.3189-02
70	9.3828-02	0.016129	0.9664659	4.0085-02
80	8.7752-02	0.012113	0.9785789	3.2177-02
90	8.1631-02	0.0096214	0.9882003	2.67480-02

TABLE 3.9B. *Results of WRLPWR for Rods at Critical*

$$I_a = 4; F_a = 3.6801; K_a = 0.9600294$$

Rods at Critical with Adjusted Amounts of Cd and Al

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
15	3.7774-02	0.0176731	0.980371	7.08586-04
20	2.7340-02	0.0118605	0.9858624	4.58909-04
25	2.1741-02	0.0084898	0.9888608	3.26522-04
30	1.8787-02	0.0062489	0.9921113	2.49217-04

TABLE 3.9C. *Results of WRLPWR for Rods at Critical*

$$I_a = 9; F_a = 9.2930-02; K_a = 0.5267735$$

Rods in with Adjusted Amounts of Cd and Al

I_n	F_n	$K_n - K_{n-10}$	K_n	C_n
21	4.3695-02	0.01836	0.5483811	3.44147-02
25	3.6196-02	0.013908	0.5531465	2.5709-02
30	2.9735-02	0.0112315	0.5582959	2.04667-02
34	2.5834-02	0.0099375	0.5619569	1.78306-02

Case 3.9B is also one with the rods at critical position, but the amounts of cadmium and aluminum have been adjusted. Case 3.9C deals with the rods in with adjusted amounts of Cd and Al.

With the technique previously described, the data in Tables 3.9A through 3.9C were used in determining the predicted values for k_{eff} . These values and their percent of accuracy

are listed in Table 3.10. Graphs of C_n versus K_n for Tables 3.9A through 3.9C were made to illustrate that these cases also approximate linear functions (Figures 3.10 and 3.11).

TABLE 3.10. Predicted k_{eff} for Tables 3.9A, 3.9B, and 3.9C

<u>Case Number</u>	<u>Given k_{eff}</u>	<u>Predicted k_{eff}</u>	<u>Predicted k/Given k</u>
3.9A	1.036	1.02335	98.78%
3.9B	1.000	0.9984815	99.85%
3.9C	0.5800	0.57463	99.07%

When analyzing the graph of Table 3.9C one finds noticeable deviation from a straight line. This case represents one of the restarts. In an attempt to determine why such deviations existed and whether or not they were factors of the selections of a and n subscripted values, several plots were made for this case using various a and n terms. The same deviated pattern was observed on all of the plots. Consequently, it was concluded that the deviations were due to the fact that the entire case exists within the region approaching final convergence. The graphs of Tables 3.8A through 3.8C adequately support this conclusion, since the deviations of the three cases follow similar patterns and only exist in the regions approaching convergence. Unfortunately, a final computer-converged results was not available for determining the actual percent of k convergence for that case. However, k_{eff} is given experimentally at about 0.58, which is within 1% of the predicted value.

FIGURE 3.10. Graphs of Tables 3.9A and 3.9B

FIGURE 3.11. Graph of Table 3.8C

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PHOENIX FUEL REACTOR PROGRAM

E. C. Davis, Jr.

MTR-Phoenix Fuel Experiment

Scans of the power distributions have been made on the first Phoenix fuel test element (GEH 14-801). This element was irradiated in position L-51 of the MTR and returned to BNW for post-irradiation analysis. The test assembly contained six Al-20 wt% Pu (21 at.% ^{240}Pu) fuel plates which were assembled into two bundles of 3 plates each, as shown in Figure 3.12. The basic features of the Phoenix test plate is shown in Figure 3.13 along with the details of the taper at the end of the fuel core. Initial gamma scanning measurements were made on a bundle of three plates, still pinned together as they were during irradiation. For these scans, the $^{95}\text{Zr}/^{95}\text{Nb}$ activities were monitored using a NaI detector and the resultant profiles were similar in shape to those given in this report. The lateral profile of this initial scan was completed to within approximately 1/4 in. of one edge of the fuel core in this bundle. This was the edge of the bundle that was located farthest from the MTR core. The profile was only partially completed due to insufficient horizontal travel on the special plate holder of the gamma scanning equipment. The holder had to be modified to allow an additional inch of travel and will now accommodate the larger size plates of the MTR. For subsequent scans ^{137}Cs was monitored for total burnup using a Ge(Li) detector. The bundle of 3 plates was separated, and measurements of the activities of single plates were obtained. These profiles were determined to be inaccurate due to a belt-drive malfunction of the horizontal traversing mechanism on the plate holder of the gamma scanning equipment. The belt slippage invalidated the calibration of position across the plate. Hence, the data obtained were not accurate enough to generate power peaking factors for the fuel plate

FIGURE 3.12. Phoenix Plate Test Assembly - Top View

FIGURE 3.13. Phoenix Test Plate

edge. The drive mechanism of the plate holder had to be modified to a combination gear and belt-drive. This modified equipment was used along with the PNL Gamma Scanning Facility which had been adapted to permit plate scanning with high spatial resolution to measure the fission product activation profiles obtained from two of these plates. The spatial distribution of fuel burnup in these plates were measured by gamma scanning them using a Ge(Li) detector to obtain the activity profiles of ^{137}Cs and ^{106}Rh . Complete axial and horizontal profiles were generated for the interior plate (Plate 46). A horizontal profile only was determined for the exterior plate (Plate 51). The two horizontal profiles were measured at approximately 1/2 in. below the element core midplane, and the axial profile was measured over the length of the fuel core along the vertical centerline.

The axial power shape, as shown in Figure 3.14, was observed to be inverted relative to the power shape obtained in the PRCF.⁽¹⁾ The peak occurred above the core midplane instead of below, as previously observed. The source of the flux shift is believed to be due to a boron sleeve that is located in position L-41. This sleeve extends from approximately +6 1/3 in. to -10 2/3 in. from the core horizontal midplane. This traverse showed the expected reduction of the power peaking at the bottom of the plate due to the tapered fuel cores. The peak-to-average for this traverse was observed to be 1.15. The power at approximately the midplane is about a factor of 2 larger than the power at the bottom of the plate.

The horizontal profile obtained from Plate 51 is given in Figure 3.15 and has a shape that is similar to those measured for PRCF Phoenix fuel plates.⁽¹⁾ The peak-to-average

FIGURE 3.14. Vertical Power Profile in Plate 46 of the Test Assembly GEH-14-801 Located in Position of L-51 of the MTR

FIGURE 3.15. Horizontal Power Profiles in an Interior Plate (No. 46) and Exterior Plate (No. 51) of the Test Assembly GEH-14-801 Located Position L-51 of the MTR

for this traverse was determined to be 2.76 with an estimated uncertainty of $\pm 43\%$. The large uncertainty of this peaking factor is directly related to the uncertainty in determining the position of the fuel edge.

The horizontal profile from Plate 46 has a shape that is similar to Plate 51 and this is also shown in Figure 3.15. However, at approximately 1/16 in. from the edge of the fuel core, the power profile reverses its slope and the magnitude decreases steadily out to the edge. The cause of this anomalous behavior has not been identified.

Measurement of the specific activities of the cobalt and nickel flux monitor wires included in this test assembly has also been completed. Each monitor plate of the assembly contained a cobalt and a nickel wire. There were six vertical wires and two horizontal wires within the assembly, and each was encased within a 304 SS sheath. Unnormalized flux profiles were obtained for eight of the encased wires by counting over the ^{60}Co and ^{58}Co activation peaks. These profiles showed that the flux at the core midplane is almost a factor of two greater than the flux at the bottom of the fuel plate. The shapes of these curves are consistent with data previously obtained from gamma scanning the fission product activities in the test fuel plates.

To obtain normalized profiles, 1/4 in. long samples were cut from selected locations of each wire. Four of these small samples were obtained from the center monitor plate and were weighed and absolute-counted to obtain the specific activity for each sample. The integrated nvt and the fast and thermal fluxes for two locations of flux monitor Plate 3 in this test element have been determined from the specific activities using the SPACT⁽²⁾ code and are listed in Table 3.11. These results indicate that the values of the thermal and fast flux will be uncertain by approximately 50% and approximately 15%, respectively.

TABLE 3.11. *Integrated nvt and Fast and Thermal Flux for Two Location on Plate 3*

Detector Material	Position Above Bottom of Plate, in.	SPACT Calculations		$R = \frac{\phi_{\text{Fast}}}{\phi_{\text{Thermal}}}$
		nvt	ϕ	
Nickel	~11	$(3.27 \pm 0.47) \times 10^{20}$	$(5.38 \pm 0.77) \times 10^{13}$ fast	1.11 ± 0.58
Cobalt	~11	$(2.95 \pm 1.48) \times 10^{20}$	$(4.85 \pm 2.43) \times 10^{13}$ thermal	
Nickel	~1/2	$(9.93 \pm 1.05) \times 10^{19}$	$(1.63 \pm 0.17) \times 10^{13}$ fast	0.48 ± 0.24
Cobalt	~1/2	$(2.07 \pm 1.03) \times 10^{20}$	$(3.40 \pm 1.70) \times 10^{13}$ thermal	

Post-Irradiation Analysis

Plates 46 and 51 from the test assembly GEH-14-801 were sectioned at the elevations at which the lateral gamma-scan traverses were performed (about 12 in. above the bottom of each fuel plate). A 4X macrophotograph of the polished, unetched, cross section obtained from Plate 51 is given in Figure 3.16. A determination of the fuel location and its dimension was made for each plate. The results of this work are listed in Table 3.12. The larger uncertainties assigned to the post-irradiation measurement results are a direct result of the uncertainty of the thickness of the saw cut which cannot be estimated accurately.

Small samples of the fuel cores were taken at the vertical centerline of Plate 46 ($0.04 \times 0.285 \times 0.575$ in.) and at the edge of Plate 51 ($0.04 \times 0.138 \times 1.635$ in.), for radio-chemical analysis. The lined regions on both plates, shown in Figures 3.14 and 3.15, respectively, represent the length and width of the samples of the fuel cores that were sectioned for destructive analysis. The results of this analysis for changes in Pu-isotopic concentrations are listed in Table 3.13. Additional results were obtained from the burnup analysis using the computer program ISODIL⁽³⁾ and are reported in Table 3.14. The total ^{137}Cs atoms have been corrected for

Neg 17350

4X

FIGURE 3.16. *Macro photograph of the Cross Sections of Plate 51 As-Polished*

TABLE 3.12. *Fuel Locations and Dimensions for Plates 46 and 51.*

	Pre- and Post-Irradiation Width, in.		Post-Irradiation Width, in.	
	Plate	Fuel Core	Aluminum Sections	
			Farthest away from MTR core	Closest to MTR core
Plate 51	Pre-	2.314 $\pm$ 0.001	1.847 $\pm$ 0.001	
	Post-	2.314 $\pm$ 0.005	1.845 $\pm$ 0.005	
Plate 46	Pre-	2.313 $\pm$ 0.001	1.850 $\pm$ 0.001	
	Post-	2.313 $\pm$ 0.005	1.845 $\pm$ 0.005	
			0.268 $\pm$ 0.001	0.203 $\pm$ 0.001
			0.239 $\pm$ 0.001	0.225 $\pm$ 0.001

TABLE 3.13. *Changes in Pu-Isotopic Concentration*

	<u>238</u>	<u>239</u>	<u>240</u>	<u>241</u>	<u>242</u>	<u>Total Pu</u>
Initial Composition, at.%	0.50	69.27	20.95	6.47	2.81	100.0
Plate 46 Sample, at.%	0.48	47.26	24.78	7.30	3.62	83.4
% Change $\left(\frac{N_i - N_o}{N_o}\right) \times 100$	-4.0	-31.8	+18.3	+12.8	+28.8	-16.6
Plate 51 Sample, at.%	0.47	36.20	24.76	7.88	4.09	73.4
% Change	-6.9	-47.7	+18.2	+21.8	+46.0	-26.6

TABLE 3.14. *Final Results from the Destructive Analysis of Plates 46 and 51*

<u>Sample</u>	<u>Total Weight of Fuel Core and Clad, g</u>	<u>Amount of Pu by Weight, g</u>	<u>¹³⁷Cs Meas. Atoms/cm³ Fuel Core</u>	<u>Fission Density Fissions/cm³ Fuel Core</u>	<u>²³⁹Pu Capture to Fission Ratio α-239</u>
Plate 46	0.637	0.060	1.825×10^{19}	2.78×10^{20}	0.543
Plate 51	1.696	0.098	4.114×10^{19}	6.27×10^{20}	0.438

decay during and after irradiation, and the fission densities were computed by assuming ¹³⁷Cs has a yield of 6.56% for plutonium fission.

The test assembly average values were determined using the results of the burnup analysis from Plate 46 and integrating over the profiles obtained from the gamma scans of Plates 46 and 51 and from eight flux monitor wires which were distributed throughout the assembly. The results obtained from this analysis are listed in Table 3.15 and indicate that an average ²³⁹Pu depletion of 28.8% was achieved for this element after being irradiated in the MTR for an equivalent 70.5 days at full power.

A second Phoenix test assembly (GEH-14-802) containing up to Al-22% Pu (21 at.% ²⁴⁰Pu) has been irradiated in the MTR for an equivalent full-power operational time of 82.94 days. These

TABLE 3.15. Average Burnup Data for Test Assembly GEH-14-801

	<div>Fission Density, Fissions/cm³ × 10⁻²⁰</div>	<div>²³⁹Pu Depletion, %</div>	<div>Total Heat Output, Btu × 10⁻⁹</div>	
Plate {	51	2.76	31.8	0.222
	47	2.43	28.6	0.194
	43	2.50	29.3	0.191
	54	2.42	28.5	0.196
	46	2.23	26.6	0.180
	48	2.37	28.0	0.191
Element GEH-14-801	2.45	28.8	1.174	

test plates have been operated at heat fluxes and temperatures that are approximately 10 to 12% greater than those achieved in the previous irradiation of GEH-14-801. Based on the higher weight percent of Pu, this second test assembly has a calculated average ²³⁹Pu depletion of 35.5%. The depletion of ²³⁹Pu atoms from both test assemblies has been plotted as a function of full power operation time in the MTR in Figure 3.17. Based on this curve, element GEH-14-802 would require an estimated 48 additional days at full power to achieve an average ²³⁹Pu burnup of 50%. At present, GEH-14-802 is not being irradiated. Consideration is being given to further irradiation of this assembly to achieve this goal.

References

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FIGURE 3.17. *MTR-Phoenix Fuel Test Assembly Burnup Curves
for the Depletion of ^{239}Pu Atoms*

4.0 FAST REACTORS

NPTR-PSAR Static Reactivity Calculations

S. L. DeMyer

Static reactivity calculations have been made in support of the Preliminary Safeguards Analysis Report (PSAR) for the Nuclear Proof Test Reactor (NPTR). The results of these calculations, together with the methods and models used, are presented in this report.

For the purposes of these preliminary calculations, the Annular Conical Core⁽¹⁾ midplane dimensions and loading pattern were assumed. General areas covered by these calculations are reactivity worths of loading errors and reactivity worths of the NPTR control systems. A summary of the calculated results is presented in Table 4.1.

TABLE 4.1. *Reactivity Worths of Loading Errors and Reactivity Worths of NPTR Control Systems*

<u>Calculation</u>	<u>Worth, Δk</u>
Maximum in-core fuel loading error	0.0113 ($\sim 3.5\%$)
Maximum reflector loading error	0.0027 ($< 1.0\%$)
Total strength of reflector control system	0.0368
Total strength of in-core control system	> 0.0702
Total strength of tertiary control systems	Design and composition dependent

A midplane diagram of the core arrangement is shown in Figure 4.1. For calculational purposes, the hexagonal sub-assemblies were rectangularized, preserving cross sectional area of the core and each individual subassembly, as shown in Figure 4.2. As the core has approximate mirror symmetry about

- REFLECTOR
OPEN TEST
- REFLECTOR
CONTROL
- REFLECTOR
- IN-CORE
CLOSED LOOP
- IN-CORE
OPEN POSITION
- IN-CORE
CONTROL
- DRIVER

FIGURE 4.1. Reference Core Arrangement

FIGURE 4.2. Task Force Reference Annular Conic Core
(Rectangularized)

the Y axis, the calculations were made in X-Y geometry, using a half core with a reflective boundary at the Y axis. This calculational model, with zone detail, is shown in Figure 4.3.

Evaluations of the model described above were made with the two dimensional diffusion code 2DB,⁽²⁾ using the X-Y geometry option. Axial leakage effects, as a function of position, were simulated through the use of a constant axial buckling and X and Y position dependent diffusion coefficients. Different five energy group cross section sets were used for the core and reflector regions. For each set, the cross sections were energy-averaged over spectra characteristic of the region of interest using the computer code FCC⁽³⁾ and a 26 group cross section set. The latter is the Russian set⁽⁴⁾ as modified by PNL.⁽⁵⁾

The core cross section set was used for all core zones and all zones adjacent to the core, except for subassemblies of "reflector" composition. The reflector cross section set was used for all reflector regions. The assumed material volume fractions are listed in Table 4.2. The core composition used in FCC was a volume weighted average of the Driver I and Driver II compositions of Table 4.2.

In-Core Fuel Loading Errors

This error results when an in-core position is accidentally filled with a subassembly of higher worth than intended for that position. To assure calculation of the maximum worth accident of this type, a Driver II subassembly has been placed in two core locations.

The first set of calculations established the change in k_{eff} caused by replacing the regular central loop material, Na and SS (see Table 4.2, Central Loop) with a Driver II subassembly. To insure that this location is, in fact, the highest worth location

FIGURE 4.3. NPTR-Half Core Calculational Model

TABLE 4.2. *Material Volume Percents*

<u>Closed Loop Test</u>		<u>Driver II</u>	
PuO ₂	1.236	PuO ₂	7.644
U ^N O ₂	4.564	U ^N O ₂	26.156
Void	0.700	Void	4.200
316 SS	46.750	316 SS	22.650
Na	46.750	Na	39.350
<u>Open Loop Test (Core)</u>		<u>C.R. Follower</u>	
PuO ₂	5.403	316 SS	40.000
U ^N O ₂	19.947	Na	60.000
<u>Central Loop (And Ref. Open Tests)</u>		<u>Control Rods</u>	
316 SS	65.000	Void	1.000
Na	35.000	316 SS	30.740
<u>Driver I</u>		Na	41.100
PuO ₂	6.369	B ₄ ^N C	27.160
U ^N O ₂	27.156	<u>Radial Reflector</u>	
Void	4.200	316 SS	75.000
316 SS	22.650	Na	25.000
Na	39.350	<u>Tertiary Control</u>	
		B ₄ ^N C	100.000

for the accidental placement of a Driver II subassembly, a second set of calculations was made in which control rod follower material in Zone 31 is replaced by a Driver II subassembly.

Apart from the materials in zones referred to above, the basic calculations have the following materials from Table 4.2 in the given zones of Figure 4.3. The results are listed in Table 4.5.

<u>Zone</u>	<u>Material</u>
1	Central loop
2,3,49,51	Closed loop test
4,5,6	C. R. follower
7→11	Core open loop test
12,13,14	Ref. open tests
15→21	C. R. follower
22,40→48	Driver II
26→39	Driver I
23,24,25	Reflector
50	Dilute Oxygen (Not listed in Table 4.2)

TABLE 4.3. *In-Core Fuel Loading Error Worths*

<u>Calculation</u>	<u>k_{eff}</u>	<u>Worth, %Δk</u>
Base Case, Na and SS in Zone 1	1.0407	-
Driver II in Zone 1	1.0520	
Change in k		1.13
Base Case, Control rod follower in Zone 31	1.0252	
Driver II, in Zone 31	1.0451 ^(a)	
Worth, per subassembly, Driver II in Zone 31		1.00

a. *Two exchanges because of symmetry and half core calculational model.*

Reflector Loading Errors

It is physically possible for a NPTR core fuel sub-assembly to be placed in a reflector or open test position in the first row of the reflector, which is the reflector row of highest worth. The worth of a highly enriched Driver II sub-assembly, replacing a reflector element in Position 20 of Figure 4.3, was found to be 0.00272 Δk.

With reference to the zones listed in Figure 4.3, and the compositions in Table 4.2, the base configurations for these calculations contains the following materials in the given zones:

<u>Zone</u>	<u>Material</u>
1	Central loop
2,3,49,51	Closed loop test
4-6	Control rod follower
7-11,14	Core-open loop test
12,13	Reflector-open loop test
15-19	Control rod follower
20,21,23-25	Reflector
22,40-48	Driver II subassembly
26-39	Driver I subassembly
50	Oxygen

Tertiary Control System Strength

A tertiary control system for the NPTR is under investigation. Its composition and location are to be defined. For these initial calculations, it has been assumed that the control material is B_4C , enriched up to about 100 at.% ^{10}B . Control rod compositions are as given in Table 4.2.

One possible location for the tertiary control system of the NPTR is the central position. The reactivity changes produced by replacing the NPTR central loop composition with 26.2 at.%, 50 at.%, and 91 at.% ^{10}B enriched rods are shown in Figure 4.4.

These results are in substantial agreement with previous calculations.⁽⁶⁾

Alternate positions for rods of a tertiary control system are the outer annuli of the closed test loops. In the FTR, these loops contain liquid Na coolant, whereas, in the NPTR loops the space can be made available for extra control purposes when the NPTR is shut down.

FIGURE 4.4. Central Loop Rod Worth Versus Enrichment
Rod Contains 27.2 vol% B_4C

To establish the maximum worth for such a control system, the Na-SS content of the outer annuli was replaced with 100 vol% B_4C . The three annuli used in the calculations are the zones numbered 49 in Figure 4.3.

The worth of three enrichments of B_4C , 19.78, 50, and 91 at.% was computed and a worth versus enrichment curve is shown in Figure 4.5.

Reflector Control Rod Worth

Calculations were made to determine the worth of 24 reflector control rods in the reflector zone adjacent to the core. The unrodded reference base configuration core has a k_{eff} of 1.04166. The reference core with control rod follower replaced by the natural B_4C rods, described in Table 4.2, has a k_{eff} of 1.00489. Hence, the worth of the 24 reflector control rods is -0.03677.

Zoning and Control Strength

Changing the fissile density in the vicinity of control elements can change their reactivity worth. This is particularly true if a change were made from a single zone core to a two zone core. To illustrate these effects, the worth of a single in-core rod, in Position 4 of Figure 4.3, and the worth of 18 reflector control rods in Zones 15, were calculated and are listed in Table 4.4.

TABLE 4.4. *Zone 15 Calculations for 1 In-Core Position 4 Rod and 18 Reflector Rods*

<u>Control Configuration</u>	<u>Number of Core Zones</u>	<u>Worth, Δk</u>
One in-core control rod	1	0.01132
	2	0.0117
18 reflector control rods	1	0.02416
	2	0.02807

FIGURE 4.5. Worth Versus ^{10}B Enrichment of Three B_4C Annuli Surrounding the Central Closed Test Loops

The zones for the two zone core are those of Driver I and Driver II as listed in Figure 4.3. For the single zone calculations the core composition was 6.85 vol% PuO₂ and 26.95 vol% UO₂.

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PHASE B FTR CRITICAL EXPERIMENTS PROGRAM

W. R. Young and J. V. Nelson

Experimental work on ZPR-III Assemblies 51 and 52 is complete. Assembly 51 was an irregular right cylinder whose driver zone and reflector compositions approximated those quantities in the split-core reference FTR.⁽¹⁾ Figure 4.6 shows a cross section of Assembly 51 in the ZPR-III. This assembly, which had a critical mass of 212 kg ²³⁹Pu in a 307 liter volume, was the object of an extensive experimental program.⁽²⁾ Assemblies 52a through 52f had the same driver zone and reflector compositions as Assembly 51 but, in addition, these cores had simulated test zones of various shapes and volumes.^(3,4) The experimental program of Assemblies 52 did not provide for the detailed nucleonics measurements of Assembly 51 but concentrated on the effect of test zones on critical mass, the relative flux to power ratio, and the peak to average flux ratio.^(3,6)

Reporting of experimental results from Assembly 51 is not complete at this time. Important results that have been received from ANL and analyzed by PNL are presented here.

In all analyses it has been assumed that $1\% \Delta k/k = 1036 \text{ inhours (Ih)}$. There is some doubt concerning the delayed neutron fraction β , which corresponds to this conversion factor. Since most calculations are given as Δk , one can convert the inhour results, as reported by ANL, back to $\Delta k/k$ simply by multiplying by $10^{-6}/0.1036$.

Distributed Sodium Worth,⁽⁸⁾ Assembly 51

In ZPR-III Assembly 51, the worth of partial sodium removal was measured for several voided volumes. Computational results, using the computer program 2DB in an 8 group R-Z mode, are given for comparison with these experiments in Table 4.5. For measurement of axial dependence, the sodium was removed

FIGURE 4.6. Phase B ZPR-III Assembly 51

TABLE 4.5. Sodium Worths

Experi- ment	Axial Location of the Perturbation	No. of Drawers Perturbed		Material Removed, g		Stainless Steel Added, g	Reactiv- ity, Δk	Reactivity Change, Δk	
		Type A	Type A*	Sodium	Stainless Steel			Exp.	Calc.
	None	0	0	0	0	0	39.937	0	
I	0 to 7 in.	6		519.0	795.72	795.3	49.049	+9.112	12.0
			12	519.0	795.72	795.3			
II	0 to 17 in.	6		1237.08	1893.48	1934.6	21.309	-18.628	-4.0
			12	1237.08	1893.48	1934.6			
III	7 to 17 in.	6		718.08	1097.76	1139.39	15.051	-24.886	-17.0
			12	718.08	1097.76	1139.39			
IV	0 to 7 in.	22		1903.0	2917.64	2916.10	50.303	+10.366	+24.0
			20	865.0	1326.20	1325.50			

from nine drawers forming a volume that was centered at about the core midplane. The midplane is defined to be perpendicular to the core axis. In Experiment I, the volume extended from 0 to 7 in. on both sides of the midplane. In Experiment II, the volume extended from 0 to 17 in. on both sides of the midplane. This is the full core length. In Experiment III, the volume extended from 7 to 17 in. on both sides of the midplane.

Two measurements were done to define the radial dependence of sodium removal. In each, sodium was removed from 0 to 7 in. on both sides of the core midplane. Experiment I above, in which sodium was removed from the 9 drawers formed by a central 3×3 array, served as a base point. In Experiment IV, sodium was removed from the 21 drawers formed from a central 5×5 array with the four corner drawers excluded.

Reactivity changes with sodium void have a negative component from the increased leakage and decreased neutron moderation. They have a positive component from decreased absorptions. The combination of these effects causes the reactivity change with sodium removal to go from positive at the core center, through zero, and negative as one goes out. Axial removal, at 0 to 17 in.,

extended the full length of the core. Table 4.5 shows that for these cases (Experiments II and III), the integrated reactivity change has gone through zero and is negative. Experiments I and IV involve a central axial and radial region such that the integral worth of sodium void has not gone negative. Calculations show that if additional sodium, in the 0 to 7 in. axial position, were removed, the integrated reactivity change will eventually go negative. The size and shape of the surface which encloses the volume of zero sodium void worth is of primary interest in calculating the energy release in the DBA.

Sodium Substituted for Central Driver Drawer⁽⁸⁾

The plutonium, uranium, stainless steel plates, and all other materials were removed from central drawer, P16, of Assembly 51 and replaced with sodium filled cans. This sodium section extended through the axial reflectors and was 58 in. long, excluding drawer ends and spring gaps. Table 4.6 summarizes the sodium and stainless steel loadings that were added.

TABLE 4.6. *Composition of P-16 Matrix Position, Na Substituted*

	<u>Half One</u>		<u>Half Two</u>		<u>Total</u>
	<u>Reflector</u>	<u>Core</u>	<u>Core</u>	<u>Reflector</u>	
Na, g	580.2	830.0	829.9	579.9	2820.0
SS, g	863.2	1255.2	1255.7	856.4	4230.5

The measured reactivity change was -763 Ih^* , whereas, the calculated change was -891 Ih , giving an E/C ratio of 0.856.

Calculations were carried out with the two dimensional diffusion code 2DB. The central drawer was cylindricized and the problem was analyzed in R-Z geometry with 8 energy groups.

* $1\% \Delta k/k = 1036 \text{ Ih}$

Worth of Uranium and Plutonium in Central Fuel Drawer⁽⁸⁾

The reactivity worth of the uranium and plutonium in the P16 matrix position was measured. The ZPPR fuel, Pu metal, U_3O_8 , and depleted uranium plates were removed from P16. Empty stainless steel cans and rearrangement of the normally present stainless steel plates took up the vacated fuel positions. The material change and associated reactivity changes, experimental and calculated, are shown in Table 4.7.

The E/C for the reactivity change on bulk fuel removal from P16 in Assembly 51 is 0.872 as shown by the values of Table 4.7. This can be compared with small sample central worths calculated from perturbation theory that yield E/C of 0.791 and 0.650 for ^{239}Pu and ^{238}U , respectively. Coworkers have calculated small sample worths in Assembly 51 with E/C of 0.77 and 0.653 for ^{239}Pu and ^{238}U .⁽⁹⁾ Though the same cross section sets were used, β values and calculational techniques differed slightly. The most probable causes of the discrepancy are the delayed neutron fraction, β , used in converting from $\Delta k/k$ to inhours, and resonance self-shielding factors.

TABLE 4.7. Uranium and Plutonium Worth Experiment

TABLE 4.7. Uranium and Plutonium Worth Experiment										
	Weight of Material, g						Stain- less Steel, (a) g	Reactiv- ity Change		E/C
	^{239}Pu + ^{241}Pu	^{240}Pu + ^{242}Pu	^{235}U	^{238}U	Mo	0		Meas.	Calc.	
(Reference base materi- al to be removed)	2076.34	179.56	12.83	5978.0	94.48	150.86	627.75			
(Perturbed base materi- al to be added)	0	0	0	0	0	0	650.24	-752	-863	0.872

a. Stainless Steel Analysis: 73.4% Fe, 17% Cr, 8.4% Ni, 0.75% Mn, and 0.45% Si by weight

Fuel Compaction⁽⁸⁾

The effect of fuel slumping in the FTR was simulated by a fuel compaction experiment. A 5 in. section of plutonium metal, depleted uranium, and ZPPR fuel was moved from the 12 to 17 in. axial position in Drawer 1 P16 to the 0 to 5 in. position of 1 P16. Sodium was moved from the 0 to 5 in. position to the 12 to 17 in. position. The total worth of this redistribution was measured to be 125 lh and calculated to be 146 lh, giving an E/C of 0.858.

A summary of the material movement is given in Table 4.8.

TABLE 4.8. *Material Redistribution in Fuel Compaction Experiment*

	<u>Moved from 12-17 in. Axial Position to 0-5 in., g</u>	<u>Moved from 0-5 in. Axial Position to 12-17 in., g</u>
²³⁹ Pu	301.495	-
²⁴⁰ Pu	26.117	-
²⁴¹ Pu	3.0882	-
²⁴² Pu	0.21685	-
²³⁸ U	751.69	-
²³⁵ U	1.61	-
Na		59.84
SS	93.766	91.48
Al	1.93	-

The E/C for the fuel compaction experiment is consistent with previous experience in calculating fuel worth in the central column of Assembly 51. Additional problems arise from the changing fuel lump size and contribute to the complexity of the analysis.

Central Fission Ratios

Central fission ratios, measured in Phase B experiments performed in ZPR-III Assembly 51, have been calculated and

compared with the measurements. The results are listed in Table 4.9. A new matrix of ^{238}U scattering cross sections has been used in other calculations to give slightly better agreement. The FCC calculation was a fundamental mode, 26 group, diffusion calculation. DTF results are from a 26 group, S4, transport calculation.

TABLE 4.9. *ZPR-III Assembly 51--Fission Ratios*

<u>Fission Ratio</u>	<u>Experiment</u>	<u>Calculated</u>	
		<u>FCC</u>	<u>DTF</u>
U-238/U-235	0.0309	0.033	0.033
Pu-239/U-235	1.003	1.012	1.014
Pu-240/U-235	0.240	0.240	0.233

Assemblies 52a through 52f^(3,4,5)

Experiment-theory correlations of the critical mass for the balance of the cores of the revised Phase B Critical Experiments⁽⁶⁾ program have been completed. In the previous Quarterly Report the correlations for cores having axial fuel-free zones of circular cross sections were given. Correlations for cores having zones of rectangular cross sections are reported here in Table 4.10. The earlier results for the circular holes are included for completeness.

An examination of Table 4.10 shows that the critical masses of the reactors with central circular holes, which simulate test zones with 80 vol% Na, 20 vol% SS, are consistently calculated 3 to 5% higher than the measurements. Those with test zones approximating rectangular central slots do not show this systematic agreement.

All critical calculations were made with the two-dimensional diffusion code, 2DB. While the cores with central

TABLE 4.10. *Critical Masses of Some Fast Pu Assemblies Having Central Fuel-Free Volumes (Test Zones)*

Assembly Number	Test Zone Areal Profile	Test Zone Vol., liters	Critical Mass, kg of fissile		E/C
			Experiment	Calculated	
52a		0	199	210	0.948
52b	Square, 3 × 3	24	236	249	0.948
52c	Square, 5 × 5 - less corner drawers	66	280.0 ± 0.5	290	0.966
52f ^(a)	Square, 5 × 5 - 3 drawers each corner	98	309.2 ± 0.5	323	0.957
52d ^(a)	Rectangle, 3 × 11	88	275.2 ± 0.5	303	0.908
52e ^(a)	Rectangle, 3 × 15	119	282.0 ± 0.5	299	0.943

a. *Large safety rods*

holes could be successfully cylindricized, those with rectangular central slots could not. As a consequence, those assemblies with central holes could be analyzed in an R-Z mode. In contrast, Assemblies 52d and 52e had to be analyzed in an X-Y mode and required an estimation of the axial buckling in order to calculate the axial leakage. It was assumed, for the above calculations, that the axial buckling of these assemblies depends only on the volume of the simulated test zone since the driver composition remains constant. Therefore, cores having equal volume test zones were assumed to have equal axial buckling. The assumed axial buckling values for Cores 52d and 52e are listed in Table 4.11. These were deduced from the axial buckling of cylindrically symmetric cores.

The relatively poor agreement between experiment and calculation for the critical assemblies with slotted test zones is

undoubtedly due to the axial buckling estimate. Until a better method of estimating this parameter is found, one can expect as much as 10% error in the calculated critical mass of the split, conical core concept of the FTR which has a slotted core.

TABLE 4.11. *Estimated Axial Buckling of Slotted Assemblies*

<u>Assembly</u>	<u>Test Zone Volume, liters</u>	<u>Axial Buckling, cm</u>
52d	88	634×10^{-6}
52e	119	627×10^{-6}

B₄C and Ta Poison Rod Worths

The relative and absolute worths of B₄C and Ta were measured in Assembly 52 at the core center (P16 matrix position)⁽⁸⁾ and at the core edge (P22 matrix position).⁽³⁾ Figure 4.7 shows the locations of the poison rods in ZPR-III. The poison samples, 1/2 × 2 × 36 in. long were placed in a typical 2 × 2 in. drawer which then consisted of about 25 vol% poison, about 50 vol% sodium, and about 25 vol% stainless steel. The reference drawer loading consisted of the same quantity of stainless steel and sodium. However, the stainless steel was rearranged to form a 1 in. square axial void centered 1/2 in. above the midpoint of the drawer. The compositions of the poison samples are listed in Table 4.12.

Analysis of these experiments was done in X-Y geometry. using the two-dimensional diffusion code 2DB⁽⁷⁾ with an 8 group cross section set. The control rod cross sections were flux-volume weighted to correct for heterogeneity. A comparison of experimental and calculated values for the absolute and relative worths of B₄C and Ta can be found in Tables 4.13, 4.14, and 4.15.

FIGURE 4.7. Assembly 52-12 Half 2

TABLE 4.12. B_4C and Ta Sample Compositions

		Composition, wt%			
		$^{10}_B$	$^{11}_B$	C	Ta
B_4C	1262.2 g, 96 pieces 1/4 × 1/2 × 3 in.	13.67	63.31	23.02	0
Ta	9432 g, 72 pieces 1/8 × 2 × 2 in.	0	0	0	100.0

TABLE 4.13. Natural B_4C and Ta Worth in P16

Test Region	Excess Reactivity of Core, Ih	Reactivity Change Relative to Void, Ih		
		Exp.	Calc.	E/C
Reference	-740	0	-	-
B_4C	-1646	-905	-832	1.09
Ta	-1313	-573	-538	1.07

TABLE 4.14. Natural B_4C and Ta Worth in P22

Test Region	Excess Reactivity of Core, Ih	Reactivity Change Relative to Void, Ih		
		Exp.	Calc.	E/C
Reference	+15.3	0		
B_4C	-198.8	-214.3	-213	1.01
Ta	-106.0	-121.3	-147	0.86

TABLE 4.15. Relative Worths of B_4C and Ta

Test Position	B_4C/Ta		
	Exp.	Calc.	E/C
P-16	1.58	1.55	1.02
P-22	1.78	1.45	1.23

The central worth of small samples of B_4C have been calculated with an E/C of 1.01. This agreement between experiment

and calculation is also true for the bulk B_4C worths shown in Tables 4.12 and 4.13. The small sample tantalum worth has not been measured.

Tantalum, ^{239}Pu and ^{238}U all have appreciable resonance shielding of cross sections while B_4C does not. This suggests that the calculated self shielding factors for some resonance absorbers may be in error.

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RADIATION LEVELS NEAR NPTR FUEL

H. C. F. Ripfel

The NPTR will use fuel containing plutonium of a variety of isotopic compositions; hence, radiation dose levels are a matter of concern in the area of fuel handling where they might put limitations on the operation of the facility. It is necessary to assess these levels to provide the designer with radiation exposure information upon which to base the design.

Areas which have a potential for significant radiation can readily be identified from the fuel handling scheme, Drawing No. SK-3-14226, to be:

1. Single fuel pin handling - preferably manually - in the pin exchange hood.
2. Observation of the hood operations - preferably visually.
3. Handling of fully assembled modules by crane for transportation to and from the facility, the pin exchange hood, the storage vault, the cell. This includes insertion into and removal from the core - preferably with close-up observation.

For present purposes, consideration is restricted to external exposure only, as the fuel is seal welded within several mils of steel. This cladding also provides enough shielding to eliminate radiation other than gammas and neutrons.

It is envisaged that the NPTR will have two types of fuel which can be characterized by the isotopic composition of the plutonium, i.e., one will have a ^{240}Pu content of about 10% (Material I), and another of about 23% (Material II). Detailed compositions are shown in Table 4.16.

TABLE 4.16. Isotopic Compositions of Plutonium

<u>Material</u>	<u>^{238}Pu</u>	<u>^{239}Pu</u>	<u>^{240}Pu</u>	<u>^{241}Pu</u>	<u>^{242}Pu</u>
I	0.04	88.3	10.3	1.22	0.16
II	0.5(0.53)	65.3(65.79)	23.1(22.6)	7.6(7.78)	3.5

Some realistic assumptions have been made in anticipation of already known or predictable design features. The pin exchange hood is designed as a glove box (to provide protection in the case of an accidental breaching of the fuel pin cladding). The gloves consist of Neoprene, 33 mils thick. The cladding of the fuel pins is 16 mils of stainless steel. The observation window of the hood consists of Lucite 3/8 in. thick. The fuel modules have cans of a wall thickness of 150 mils of stainless steel. The operator is to wear gloves of the same kind as described above. The reactor loading device provides a concentric steel shielding around the modules of 0.5 in. wall thickness. An additional shielding of 2 in. of Boron-polyethylene is considered to reduce the neutron radiation level.

Dose rates, for single pins and pin bundles of these materials, at the surface and at a distance of 2 ft, which is the approximate position of the operator relative to the material, were found in Reference 1. When these are used together with gamma attenuation factors and neutron shielding efficiencies, given in the same reference, the radiation dose rates for the NPTF could be estimated and are given in Table 4.17

The gamma dose rate will increase as a function of time due to ^{241}Am buildup. This effect could become troublesome for material that contains appreciable amounts of ^{241}Pu . Unfortunately the data on the time behaviour are rather poor. Therefore, an adaptation had to be made to apply them to the problem. A second data source⁽²⁾ refers to a material which has essentially

TABLE 4.17. *Contributing and Total Dose Rates Considering Working Conditions and Shielding as Applicable for Fuel Consisting of a Mixture of 25 wt% PuO₂ and 75 wt% UO₂*

Material	Dose Rate, mrem/hr ^(a)			
	Neutrons	Gamma	Neutrons	Gamma
	Total		Total	
	I		II	
Single Pin:				
Surface	3.6	107	10	250
Through 16 mils of steel and 33 mils of Neoprene	111		260	
At 2-ft distance from side.	0	0.175	0.03	0.41
Through 16 mils of steel and 3/8-in. of Lucite	0.175		0.44	
217 Pin Subassembly:				
Surface	47	5	150	12
Through 166 mils of steel and 33 mils of Neoprene	52		162	
At 2-ft distance from side	1.6	-	4.4	0.01
Through 0.666 in. of steel and 2 in. of Boron-polyethylene	1.6		4.41	

the same isotopic composition as Material II, see figures in parentheses in Table 4.16, but is significantly older: 485 days compared to 120 days. In both cases the surface dose rates from PuO₂ were measured and are displayed in Figure 4.8, Curve I. From these results, and with the knowledge of a point corresponding to a material aged for 120 days, the corresponding curve for the 25% PuO₂-75% UO₂ mixture was obtained considering:

FIGURE 4.8. Dose Rates at the Surface of PuO_2 (I) and at 25% PuO_2 -75% UO_2 Mixture (II)

- a. The energy distribution of the source material is not time dependent, because after 120 days more than 99% of the gammas already originate from ^{241}Am .
- b. The physical properties of the material (self-shielding, source distribution) which also determine the slope of the curve can be taken care of using the relationship:

$$\frac{\Delta J_1}{J_1} = \frac{\Delta J_2}{J_2} = \text{const} \quad ,$$

where ΔJ is the increase in dose rate in a given interval of time and J is the dose rate itself. Curve II of Figure 4.8 shows the result of these calculations.

From Curve II the relative increase of the dose rates was taken and used to obtain the results shown in Table 4.18 for Material II, for separation times up to 5 years.

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TABLE 4.18. *Contributing and Total Dose Rates from Material II as a Function of Time*

Time Since Separation	Dose Rate, mrem/hr					
	Neutrons		Gamma		Neutrons	
	Total		Total		Total	
	1 Year		2 Year		5 Year	
Single Pin:						
Surface	<10	428	<10	690	<10	1440
Through 16 mils of steel and 33 mils of Neoprene	438		700		1450	
At 2 feet distance from side	<0.03	0.77	<0.03	1.12	<0.03	2.4
Through 16 mils of steel and 3/8-in. of Lucite	0.8		1.15		2.43	
217 Pin Subassembly:						
Surface	<150	20.5	<150	32.8	<150	69
Through 166 mils of steel and 33 mils of Neoprene	170.5		183		219	
At 2-ft distance from side	<4.4	0.017	<4.4	0.027	<4.4	0.06
Through 0.666 in. of steel and 2 in. of Boron-polyethylene	4.42		4.43		4.46	

APPLICATION OF SYNTHESIS TECHNIQUES TO SHIELDING PROBLEMS

H. S. Zwibel

Variational synthesis methods have been successfully applied to a variety of problems. With this approximation technique, a multivariable problem is reduced to one of lesser dimensionality. The reduction is carried out by calculus of variation techniques with emphasis on producing optimum accuracy in some integral quantity, referred to as the characteristic of interest.

In order to investigate the possible applicability of these methods to deep penetration problems, an angle-space synthesis

approximation to the Boltzmann transport was applied to several problems.^(1,2) In each of these, the characteristics of interest (criticality constant for a homogeneous slab reactor and total flux in a noncritical slab with a uniform source) was calculated with high accuracy.

With this encouragement a shielding problem was then attempted. A source was impressed at the left end of a uniform slab and the emergent flux at the right end calculated, but in this case without success. That is to say, ordinary diffusion theory was just about as accurate. The reason for this failure of the variational synthesis method is probably due to the nature of the characteristic of interest. For the successful examples, the characteristic of interest was obtained by averaging the flux over the entire system. This exploits the full potential of the variational method in that errors in the flux are averaged out (see Reference 2 for a good example of this). The characteristic of interest for the shielding problems admitted no such cancellation. Because of this, synthesis methods as applied to such stringent problems are apparently not successful.

There are, however, areas in shielding for which one might profitably employ variational synthesis. For example, the decay heat in a shield is obtained by integrating the flux (times a cross section) over the entire shield. Such a characteristic of interest can probably be calculated by synthesis methods with good accuracy.

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HOT CELL SHIELDING CONSIDERATIONS

W. L. Bunch, C. A. Mansius, and L. D. O'Dell

Consideration has been given to the use of a hot cell above the FTR in which refueling operations would be carried out. Such a hot cell would serve several purposes, including a radiation barrier. Three studies were made to consider the effectiveness of a hot cell shield--from sodium spills, from sodium vapors, and from fission products following a design basis accident. It is concluded that a hot cell designed to accommodate irradiated fuel would be satisfactory from these three aspects.

For this study, a sodium spill was considered to contain about a pound of sodium and form a disc about 2 cm thick and 18 centimeters in diameter. Specific activity of primary system sodium has been calculated at 0.05 Ci of ^{24}Na and 10 μCi of ^{22}Na per cm^3 . The gamma intensity 100 cm from the disc source was calculated to be 50 R/hr and 3 mR/hr, respectively, from the two sodium isotopes. The ^{24}Na activity decays with a half life of 15 hr. Therefore, the hot cell could probably be entered within a few days following a spill to permit contact cleanup. If a pound of sodium also contained one-millionth of the equilibrium fission product inventory in a fuel subassembly, these would generate a gamma field of about 5 R/hr at zero decay time and about 0.5 R/hr a week later. Cleanup of a sodium spill might be complicated by the presence of fission and/or corrosion products.

If a hot cell is used above a reactor in which the cover shield is removed to facilitate refueling, then sodium vapors would evaporate from the pool and deposit on the cell walls. Calculations indicate that radioactivity of the sodium deposit would not prevent personnel access into the hot cell, though long-lived fission or corrosion products might cause

significant levels. This conclusion assumes that entry could be delayed sufficiently to permit the decay of ^{24}Na . Calculations were based on a pool temperature of 400 °F with an evaporation rate of 10^{-6} pounds of sodium per hour per square foot. Activity of the pool sodium was assumed to build up exponentially with a 2.6 year half life toward a saturated activity of 20 $\mu\text{Ci/g}$. Radioactive sodium was considered to be deposited during shut downs when the cover was removed, and the residual deposit then decayed during succeeding operating periods. The equilibrium gamma field was estimated to approach about 0.1 mR/hr due to the ^{22}Na , after about 10 years of operation. Depending on fuel failure history, fission products in the sodium might increase the gamma field by one or two orders of magnitude.

Following a design basis accident, it was assumed that all of the sodium and fission products would be vaporized and contained within a hot cell. The thirty-day integrated exposure received by an operator remaining at a control position located 20 ft from the cell was estimated based on gamma intensities calculated using the computer code ISOSHL. A relatively insignificant exposure would be received from all sources through a hot cell wall containing 400 g/cm^2 . If more realistically, it were assumed only the noble gases would remain suspended then a shield containing about 200 g/cm^2 would probably provide sufficient personnel protection. Actual shield requirements are dependent on the selected design basis accident together with tolerable radiation criteria yet to be established.

GAMMA INTENSITY NEAR CLOSED LOOP PIPES

C. A. Mansius

It will be necessary to shield personnel and equipment from the radioactive coolant, corrosion, and fission products contained in closed loop pipes both during operation of the loop

and following shutdown. Shield requirements are important in some concepts because of the limited amount of space above the reactor cover. This region must accommodate not only the closed loop piping, but also control rod drives, electrical lines, and fuel handling equipment, yet be shielded to provide access for maintenance and repair work.

The gamma intensity as a function of distance and shield thickness was calculated for two different size pipes containing radioactive sodium with a specific activity of 0.01 Ci of ^{24}Na per cm^3 . Both pipes were assumed to be made of stainless steel, with the 8 in. diameter pipe having 1/2 in. thick walls and the 3 in. diameter pipe having 1/4 in. walls. Specific activity of the sodium is dependent on design of the closed loop, and could be a significant factor different than the assumed value.

From Table 4.19 it can be seen that a little more than 3 in. of iron or about 1 in. of depleted uranium will reduce the gamma intensity by a factor of ten. For pipe diameters between the tabulated sizes, the gamma intensity will be proportional to the cross sectional area of the contained sodium. For distances in the range tabulated, the intensity is inversely proportional to the distance from the pipe surface.

If it is assumed that a particular component or piece of equipment will deteriorate after exposure to about 10^7 or 10^8 rads, then about 2 to 4 in. of iron or equivalent will be required to protect such equipment located in the vicinity of the closed loop piping. If personnel access for maintenance work near the sodium-filled pipes is delayed for about a week or ten days following shutdown, then this amount of shielding would also be adequate for the residual ^{22}Na activity. However, if personnel access is required during operation or immediately following shutdown, significantly thicker shields are required.

These calculations provide realistic estimates of shield requirements for conceptual design purposes. More detailed calculations will be required for specific system arrangements as the design progresses.

TABLE 4.19. *Gamma Intensity near Closed Loop Pipes, R/hr*

Pipe Diameter, in.	Distance from Pipe, ft	Shield Material and Thickness, in.			
		Air	Iron(2)	Iron(4)	Depleted Uranium(2)
8	1	1800	460	76	11.0
	5	370	100	24	2.9
	10	170	45	11	1.3
3	1	370	100	24.0	2.4
	5	64	17	4.6	0.53
	10	22	6.2	1.7	0.19

FAILED FUEL DETECTION

W. L. Bunch, C. A. Mansius, and L. D. O'Dell

It is desired that the FTR be instrumented to detect and locate failed fuel elements. A number of calculations were made to provide instrument engineers with information to assist in the selection of a conceptual design for the failed fuel element detection and location (FFEDAL) system. These calculations include an estimate of the gamma field below the cover, the delayed neutron source strength and the noble gas activity associated with fission products, and the effect of burnup on tag material.

The gamma field below the cover is from the radioactive sodium as well as the reactor system. If the depth of the sodium pool above the reactor core exceeds about ten feet, then the gamma field is due almost entirely to the radioactive sodium and is estimated to be greater than 10^5 R/hr. A field of this magnitude makes it impractical to locate FFEDAL detectors below the cover.

The fission product isotopes $^{87-90}\text{Br}$ and $^{137-139}\text{I}$ were assumed to account for those delayed neutrons with half lives greater than about 1.5 seconds. By using yield factors generated by Anderson⁽¹⁾ for fast fission of ^{239}Pu and neutron emission probabilities for each specific isotope based on a study by Hermann et al.,⁽²⁾ a neutron decay curve was calculated for the material generated by fission events in an average FTR fuel pin. The function plotted in Figure 4.9 is the relation

$$F(t) = f \sum_i Y_i P_i \exp(-\lambda_i t)$$

where f is the fission rate per second within the pin, Y_i is the yield of the i -th isotope, P_i is its neutron decay probability, and λ_i is its decay constant. The summation is taken over the seven isotopes of interest. The function can be used, together with release fractions and dilution factors, to estimate the neutron source strength within specific detector arrangements.

The noble gas inventory associated with an average FTR fuel pin was calculated using the computer code RIBD. Production rates, goal exposure inventories, and pertinent decay information for each of the isotopes are summarized in Table 4.20. Yield factors are based on Reference 1.

The three stable xenon isotopes 124, 126, and 129 have been suggested for use as tag materials whose isotopic ratio could be associated with a specific fuel subassembly. If burnup of the isotopes alters the initial ratio significantly prior to failure of the fuel cladding, then the tag material will not unambiguously locate the failure. Spectrum weighted, theoretical cross sections of 130, 61, and 320 millibarns for the three isotopes of xenon, respectively, were calculated by R. E. Schenter⁽³⁾ using an optical potential model. At a goal

FIGURE 4.9. Decay Function for Delayed Neutron Emission

TABLE 4.20. Noble Gas in FTR Fuel Pin

Isotope	Equil. Prod. Rate, atoms/sec	Inventory at Goal Exposure with No Decay		Half-Life	Decay Constant λ (sec ⁻¹)	β Particle		γ Ray		Total E _g , MeV
		Curies	Volume, cm ³			E _b , MeV	%	Avg. E _b , MeV	E _g , MeV	
Kr 80		0.00	1.296 - 08	∞	0	0	0	0	0	0
81		0.00	4.469 - 09	∞	0	0	0	0	0	0
82	1.739 + 12	0.00	1.040 - 02	∞	0	0	0	0	0	0
83*	2.542 + 12	7.012 + 01	9.321 - 04	1.86 hr	1.033 - 04	0	0	0	0.0322	100 0.0415
									0.0093	100
83	2.542 + 12	0.00	2.563 - 00	∞	0	0	0	0	0	0
84	3.479 + 12	0.00	3.746 - 00	∞	0	0	0	0	0	0
85*	4.596 + 12	1.256 + 02	3.949 - 03	4.39 hr	4.39 - 05	0.835	80	0.233	0.305	20 0.414
									0.1495	80
85	8.964 + 11	1.330 + 00	8.958 - 01	3.93 x 10 ³ d	2.04 - 09	0.695	99+	~0.232	0	0 0.232
86	6.021 + 12	0.00	6.243 - 00	∞	0	0	0	0	0	0
87	7.66 + 12	2.093 + 02	1.895 - 03	1.27 hr	1.516 - 04	1.27	25	1.01	2.3	<25 1.57
						3.63	75		1.89	<25
									0.41	<25
Kr 88	9.312 + 12	2.548 + 02	5.101 - 03	2.81 hr	6.85 - 05	0.52	70	0.331	2.40	35 2.07
						0.90	10		2.19	18
						2.7	20		1.85	15
									1.55	14
									1.20	4
									0.85	23
									0.365	5
									0.191	35
									0.163	7
									0.028	7
89	1.069 + 13	2.929 + 02	1.116 - 04	3.2 min	3.61 - 03	4.0	100	1.3	0	0 0
90	9.78 + 12	2.697 + 02	1.767 - 05	33. sec	2.1 - 02	3.2	100	1.07	0	0 0
91	7.165 + 12	1.974 + 02	3.918 - 06	10. sec	6.93 - 02	3.6	100	1.20	0	0 0
92	3.653 + 12	1.019 + 02	6.066 - 07	3. sec	2.31 - 01	5.0	100	1.67	0	0 0
Kr 93	1.231 + 12	3.494 + 01	1.388 - 07	2.0 sec	3.46 - 01	8.0	100	2.66	0	0 0
94	2.676 + 11	8.090 + 00	2.249 - 08	1.4 sec	4.95 - 01	6.4	100	2.14	0	0 0
95	3.345 + 10	5.875 + 00	1.166 - 08	1.0 sec	6.93 - 01	9.3	100	3.1	0	0 0
97		3.736 + 00	7.415 - 09	1.0 sec	6.93 - 01	10.6	100	3.5	0	0 0
Total Kr		1.576 + 03	1.347 + 01							

TABLE 4.20. (contd)

Isotope	Equil. Prod. Rate, atoms/sec	Inventory at Goal Exposure with No Decay		Half-Life	Decay Constant λ (sec ⁻¹)	β Particle			γ Ray		
		Curies	Volume, cm ³			E _b , MeV	%	Avg. E _b , MeV	E _g , MeV	%	Total E _g , MeV
Xe 128		0.00	8.943 - 02		0	0	0	0	0	0	0
129	8.028 + 12	0.00	9.123 - 04		0	0	0	0	0	0	0
130	1.572 + 13	0.00	2.260 - 01		0	0	0	0	0	0	0
131*	1.365 + 11	4.194 + 00	8.631 - 03	12.0 d	6.69 - 07	0	0	0	0.163	100	0.163
131	2.275 + 13	0.00	2.199 + 01		0	0	0	0	0	0	0
132	3.044 + 13	0.00	3.109 + 01		0	0	0	0	0	0	0
133*	1.003 + 12	3.019 + 01	1.191 - 02	2.3 d	3.49 - 06	0	0	0	0.233	100	0.233
133	3.813 + 13	1.037 + 03	9.371 - 01	5.27 d	1.52 - 06	0.345	100	0.115	0.081	100	0.081
134	4.314 + 13	0.00	4.422 + 01		0	0	0	0	0	0	0
Xe 135*	1.446 + 13	3.628 + 02	6.873 - 04	16 min	7.23 - 04	0	0	0	0.52	100	0.52
135	3.132 + 13	1.242 + 03	8.163 - 02	9.19 hr	2.1 - 05	0.548	5	0.302	0.60	4	0.268
						0.910	95		0.37	1	
									0.25	96	
136	4.607 + 13	0.00	4.722 + 01	00	0	0	0	0	0	0	0
137	4.346 + 13	1.174 + 03	5.453 - 04	3.9 min	2.96 - 03	4.0	100	1.33	0	0	0
138	3.323 + 13	8.974 + 02	1.497 - 03	14 min	8.25 - 04	3.02	100	1.0	0	0	0
139	1.998 + 13	5.417 + 02	4.449 - 05	41 sec	1.69 - 02	5.15	100	1.7	0	0	0
140	9.653 + 12	2.639 + 02	8.358 - 06	16 sec	4.33 - 02	3.97	100	1.3	0	0	0
141	3.412 + 12	9.405 + 01	3.174 - 07	1.7 sec	4.07 - 01	6.07	100	2.0	0	0	0
142	7.426 + 11	2.040 + 01	6.075 - 08	1.5 sec	4.62 - 01						
143	8.697 + 10	7.151 + 00	1.419 - 08	1.0 sec	6.93 - 01	6.95	100	2.3	0	0	0
144		4.514 + 00	8.958 - 09	1.0 sec	6.93 - 01	5.78	100	1.9	0	0	0
Total Xe		5.679 + 03	1.459 + 02								

exposure of about 3×10^{23} nvt, these cross sections indicate that tag increments of about 5% would be required to avoid ambiguity due to differential burnup of the tag material.

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5.0 CRITICAL MASS PHYSICS

CRITICALITY OF $\text{Pu}(\text{NO}_3)_4$ SOLUTIONS IN SLAB GEOMETRY

R. C. Lloyd, S. R. Bierman, C. A. Rogers,
and R. D. Johnson

The critical thicknesses of plutonium-bearing slabs are needed to provide an improved basis for nuclear safety guidance and criticality calculations. In the past, criticality data have been reported for slabs of $\text{Pu}(\text{NO}_3)_4$ solutions containing 4.6, 18.4, and 23.2 wt% ^{240}Pu .^(1,2,3) These parameters were derived from experiments performed with the expandable-slab critical assembly which has been described previously.⁽⁴⁾

During the current reporting period, this assembly has been used to examine the criticality of $\text{Pu}(\text{NO}_3)_4$ solutions containing 18.4 wt% ^{240}Pu . Data were acquired for systems with plutonium concentrations of 412 and 159 g/liter in the bare configuration and reflected on both faces with either water or 1 in. of Lucite. The measured critical dimensions of the 42 in. wide vessel are listed in Table 5.1 with the results of a preliminary chemical analysis of the solutions.

A series of constant-thickness experiments was carried out in which a block of Lucite, measuring 6 × 42 × 48 in., was placed at various distances from the assembly. The effect of this large reflector on the critical height is shown in Figure 5.1. The data indicate that placing the reflector in front of the assembly is more effective than placing it the same distance behind. This is evidently due to the non-central location of the assembly within the hood. The rear wall of the hood is closer to the assembly than the front wall, and thus acts as a better reflector. The additional Lucite reflector accentuates this asymmetry by effectively moving one wall or the other closer to the assembly.

TABLE 5.1. *Criticality of $\text{Pu}(\text{NO}_3)_4$ Solution in Slab
Geometry Slab Assembly - 42 in. Width,
18.4% ^{240}Pu*

Exp. No.	Reflector Status	Pu Conc., g/l	Acid Molarity	Sp. Gr.	Total NO_3 g/l	Critical Slab Thickness	Critical Volume, l	Critical Height, in.
455	1 in. Lucite on 2 sides	412	4.2	1.7643	658	8.90 in.	134.39	22.13
456	Water					8.90	86.38	14.32
457						8.40	89.16	15.63
458						7.90	93.84	17.46
459						7.40	102.06	20.23
460						7.0	113.16	23.67
461						6.8	122.04	26.25
462		159	3.0	1.3595	357	6.8	66.63	14.45
463						5.4	109.93	29.76
464						5.3	121.62	33.49
465						5.5	100.83	26.81
466						5.8	85.28	21.56
467						6.2	74.89	17.75
468						6.7	68.05	14.97
469						7.5	58.61	11.58
471	1 in. Lucite on 2 sides					7.5	92.18	18.06
472						7.9	87.15	16.24
473						6.9	108.58	23.05
474						6.6	125.53	27.81
475						6.4	145.88	33.27
476						6.3	162.72	37.67
477						8.4	83.44	14.66
478	Bare					8.4	121.06	21.13
479						8.1	136.01	24.58
480						7.9	151.67	28.06
481						7.7	177.07	33.56
482						7.6	196.25	37.66
483	6 in. Lucite at 12 in. 6 x 42 x 48 in.					7.6	145.97	28.08
484	6 in. Lucite at 23-3/4 in. 6 x 42 x 48 in.					7.6	177.26	34.04
485	6 in. Lucite at 35-5/16 in. 6 x 42 x 48 in.					7.6	187.86	36.06
486	6 in. Lucite at 4-5/8 in. 6 x 42 x 43 in.					7.6	123.75	23.84
487	6 in. Lucite at 11-3/16 in. 6 x 42 x 48 in.					7.6	152.43	29.31
488	Bare					7.6	200.82	38.53
489						8.4	122.48	21.38
490						8.9	108.17	17.86

The results of a preliminary analysis of the data to obtain the best lateral extrapolation length and buckling for each system are given in Table 5.2. The extrapolation lengths were evaluated by minimizing the root-mean-square error in the

FIGURE 5.1. *Effect of a Large Reflector Block on Slab Criticality (6 × 42 × 48 in. Plexiglass)*

TABLE 5.2. Bucklings and Extrapolation Lengths

Concentration, g/l	Reflector	λ , cm	Buckling, m^{-2}
412	water	λ sides = 7.62	115.43
159	water	λ sides = 7.00	147.96
159	1 in. Lucite	λ sides = 5.91	144.02
159	Unreflected	λ = 4.42	140.97

critical bucklings computed for each set of experiments. As a result, the critical buckling for the 159 g Pu/liter solution varies slightly with the different reflector materials. This variation is apparently due to the spatial dependence of the energy spectrum of the neutron flux within each system. Since this dependence is effective over most of the core thicknesses and is not the same for each system, the space- and spectrum-averaged material cross sections for each system are not identical. Thus the material buckling of a given fuel may vary with the configuration of the critical assembly. This effect is being investigated.

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