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REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
JULY, AUGUST, SEPTEMBER, 1967

FEBRUARY 1968

AEC RESEARCH &
DEVELOPMENT REPORT

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REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
JULY, AUGUST, SEPTEMBER 1967

By

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February 1968

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RICHLAND, WASHINGTON

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REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
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FOREWORD

The Technical Activities Quarterly Report replaces the Physics Research Quarterly Report previously published by Battelle-Northwest's Reactor Physics Department and FFTF Reactor Physics Section. The objective of the report is to inform the scientific community in a timely manner of the technical progress made on the many phases of work within the Department. The report contains brief technical discussions of accomplishments in all areas where significant progress has been made during the quarter. The results presented should be considered preliminary, and do not constitute final publication of the work. A list of publications and papers by the Department staff is given in the report. Anyone wishing to obtain additional information on the work presented is encouraged to contact the author directly.

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1.0 SUMMARY

REACTOR THEORY AND CODE DEVELOPMENT

A space-angle synthesis approximation of the transport equation has been developed which shows improved accuracy over the P1 approximation. The problem considered is an infinite homogeneous slab with a uniform source.

The Battelle Revised Thermos code has been modified by the addition of an improved current calculation routine, fission cross section averaging both microscopically per isotope and macroscopically for the cell, and the capability to handle up to eight mixtures.

A computer program entitled HTH, comprising an overlay of codes HRG, Battelle Revised Thermos, and HFN, has been coded to provide a code package for use in reactor analysis.

Modifications have been made to the HAMMER code to make it more compatible with other codes in use at PNL.

THERMAL REACTORS

The program of experiments in the Phoenix-fueled PRCF-MTR mockup has been completed. The measurements have provided valuable information about the shutdown capability, kinetics

parameters, and power distributions to be expected in the Phoenix fuel core planned for the MTR, and have provided data with which to test the computational methods currently being applied to Phoenix fuel cores.

Calculations have been performed to determine the feasibility of fueling a maritime reactor with plutonium Phoenix fuel. The Babcock and Wilcox (B&W) Consolidated Nuclear Steam Generator (CNSG-II) design was used as the base case. The reactivity, multiplication, and core lifetime characteristics were calculated for various plutonium Phoenix loadings.

Physics calculations have been performed to determine the feasibility of a plutonium Phoenix fuel loading in the PRTR. Nineteen-rod clusters of Al-Pu with a plutonium isotopic composition of 65.5/23.3/7.7/3.5% of $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$ respectively. Core lifetimes in the range of 60,000 megawatt days were calculated.

Physics calculations have been performed to determine the feasibility of a thorium fuel loading in the PRTR. The fuel element types considered were nested metal cylinders of Th-4 wt% ^{235}U and 18 to 19 rod clusters of ThO_2 - $^{235}\text{UO}_2$. Core lifetimes of 15,000

to 20,000 megawatt days were calculated.

Measurements were made in the Physical Constants Test Reactor (PCTR) to obtain the adjoint weighted excess neutron production cross sections $\langle \Sigma_p \rangle$ for a 2.1% enriched uranium, 2.93 wt% lithium (42.8% ^6Li) in aluminum target supercell in a three-to-one ratio. Results were obtained for a test array utilizing a square shaped supercell building block, and a "T" shaped building block as the central supercell. Test array conditions were both wet (coolant water in process tube) and dry (no coolant water). Calculated values for the average supercell thermal neutron absorption cross section were used in conjunction with the values of $\langle \Sigma_p \rangle$ to obtain values of the infinite medium neutron multiplication factor, k_∞ . These values of k_∞ are compared to the corresponding values obtained by the PCTR null reactivity technique. The agreement of the values for k_∞ is very good for these systems where k_∞ is very close to 1.0.

FAST REACTORS

A model of the FTR split core in x-y geometry has been developed and extensive physics statics calculations have been carried out for this configuration.

The effect of adding BeO to the FTR reflector is being examined. Modest improvements in the Doppler coefficient seem achievable. These improvements are offset by the introduction of undesirable power gradients near the core reflector interface.

The relative neutronic characteristics of stainless steel and inconel side reflectors have been compared. The inconel reflector shows marked advantages over the steel reflector.

An analysis of typical energy deposition patterns in the FTR have been made. About 95% of the fission energy is deposited in the core, and 90% in the core fuel.

A new, maximum hypothetical accident code, MAX, has been developed. At present, this code is being used to generate parametric data for the FTR.

The analysis of the Phase A experiments is almost complete and detailed plans for the Phase B experiments have been generated.

An extensive evaluation of the functions and advantages of the NPTF has been completed and documented. Safety studies for the NPTF have been started. The possibility of using the NPTF/FTR system to detect anomalous behavior in the FTR via continuous on-line measurements of neutron noise is being examined.

Analyses of the AI shielding experiments are continuing. Comparison of MAC (a removal diffusion code) and the discrete ordinate transport code ANISN is being made.

An estimate of nuclear heating rates in the radial neutron shield and in the axial cover shield for use in the conceptual design of the shield cooling systems has been made.

Decay heat from irradiated FTR driver fuel has been estimated using the computer program RIBD. Radioactivity levels and heat generation for unirradiated FTR driver fuel has also been estimated.

Work is continuing on the calculation of one group average capture cross sections for the 450 fission product isotopes. Current investiga-

tions have been concentrated on isotopes neighboring those which have significant radiological effects.

2.0 REACTOR THEORY AND CODE DEVELOPMENT

SPACE-ANGLE SYNTHESIS

H. Zwibel and B. Bowes

A space-angle synthesis approximation of the transport equation has been developed which shows improved accuracy over the P1 approximation. The problem considered is an infinite homogeneous slab of thickness $2a$ with a uniform source.

The transport equation, if one assumes isotropic scattering and one energy group, is

$$\frac{\mu}{2} \frac{\partial}{\partial x} f(x, \mu) + f(x, \mu) = \frac{c}{2} \int_{-1}^1 d\mu' f(x, \mu') + 1,$$

with the following vacuum boundary conditions:

$$f(-a, \mu) = 0 \quad \mu > 0$$

$$f(a, \mu) = 0 \quad \mu < 0$$

Distances are measured in units of total mean-free-path, and c is the mean number of secondaries per collision ($c = \frac{\Sigma_s}{\Sigma_t}$).

The P1 solution for this problem is of the following form

$$f(x, \mu) \approx \frac{1}{1-c} + A \cosh kx + B\mu \sinh kx$$

where A and B are constants and $k = \sqrt{3(1-c)}$.

Guided by the P1 solution, we take for the synthesis approximation

$$f(x, \mu) \approx \frac{1}{1-c} + \cosh kx g_1(\mu) + \sinh kx g_2(\mu) \quad (1)$$

where $g_1(\mu)$ and $g_2(\mu)$ are functions of μ to be determined and k is a parameter to be determined.

Equations for $g_1(\mu)$ and $g_2(\mu)$ are obtained by using the reduced Lagrangian concept.⁽¹⁾ This reduced Lagrangian is obtained from the following total Lagrangian:⁽²⁾

$$\begin{aligned} L[\phi^\dagger, \phi] = & \int_{-a}^a dx \int_{-1}^1 d\mu \phi(x, \mu) \\ & - \int_{-a}^a dx \int_{-1}^1 d\mu \phi^\dagger(x, \mu) \\ & \cdot \left[\frac{\mu}{2} \frac{\partial}{\partial x} \phi(x, \mu) + \phi(x, \mu) \right. \\ & \left. - \frac{c}{2} \int_{-1}^1 d\mu' \phi(x, \mu') - 1 \right] \\ & - \int_0^1 d\mu \mu \phi^\dagger(-a, \mu) \phi(-a, \mu) \\ & + \int_{-1}^0 d\mu \mu \phi(a, \mu) \phi(a, \mu) \quad (2) \end{aligned}$$

The first term is the characteristic of interest (in this case the total flux in the slab).

The last two terms incorporate the vacuum boundary conditions into the functional. $\phi^\dagger$ and ϕ are arbitrary adjoint and flux trial functions respectively.

The flux trial function is given by Equation (1) and adjoint is obtained from the fact that $f^\dagger(x, \mu) = f(x, -\mu)$. Inserting these functions into Equation (2) and imposing stationarity of L with respect to arbitrary variation of g_1 and g_2 yields a pair of coupled integral equations for g_1 and g_2 . Due to the degenerate nature of the kernel, these are readily solved. These solutions are then substituted into Equation (2), and the extremum as a function of k is found.

The results for the absorption probability are shown in Table 2.I for several different choices of a and c . The Los Alamos transport code, DTF, was used to obtain "exact" solutions for the comparison.

TABLE 2.I. Absorption Probability

a	c	Variational	P1	DTF
0.1	0	0.2575	0.1736	0.2601
	0.2	0.2172	0.1438	0.2195
	0.5	0.1479	0.0950	0.1496
	0.8	0.0649	0.0404	0.0658
0.5	0	0.6054	0.5533	0.6097
	0.2	0.5519	0.4981	0.5561
	0.5	0.4304	0.3835	0.4409
	0.8	0.2379	0.1997	0.2410
5.0	0	0.9500	0.0464	0.9500
	0.2	0.9404	0.9365	0.9404
	0.5	0.9146	--	0.9147
	0.8	0.8355	0.8298	0.8356

Note that the synthesis approximation is much more accurate than the P1 approximation for all cases considered. Since both approximations contain the same number of terms, the angular functions generated by the space-angle synthesis approach must describe the system more adequately. A similar approach is being applied to shielding problems.

References

- (1) D. S. Selengut. *Variational Analysis of Multi-Dimensional Systems*, HW-59126. General Electric Company, Richland, Washington.
- (2) G. C. Pomraning and M. Clark, Jr. "Spatial Expansion of Transport Equation," *J. of Nuc. Energy*, vol 18, p. 191. 1964.

CURRENT CALCULATION IN THERMOS

D. R. Skeen and A. G. Gibbs

The calculation of the diffusion coefficient in the Battelle-Revised-Thermos code⁽¹⁾ is based on a neutron current. The present version of the code utilizes the diffusion theory approximation in computing the current (i.e., gradient of the flux). An improved method which uses information contained in the transport matrix has been incorporated in a revised version of the code.

In the THERMOS code, the basic neutron balance equation is written as as⁽²⁾

$$N(\underline{r}, v) = \int H'(\underline{r}', v) T(\underline{r}, \underline{r}', v) d\underline{r}' \quad , \quad (1)$$

where $N(\underline{r}, v)dv$ is the number density of neutrons at position $\underline{r}$ having speed in dv , and $H'(\underline{r}, v) = \frac{1}{v} H(\underline{r}, v)$ where $H(\underline{r}, v)dv$ is the birth rate density of neutrons at position $\underline{r}$ with speeds in dv . The kernel $T(\underline{r}, \underline{r}', v)$ represents the flux of uncollided neutrons at $\underline{r}$ produced by a unit source of neutrons of speed v located at $\underline{r}'$.

We will consider a one-dimensional system (slab, cylinder, or sphere) and denote the spatial coordinate by r .

Consider a surface $r = r_0$, which is either a plane or the surface of a cylinder or sphere. The net neutron current crossing the surface $r = r_0$ can be determined as follows.

We first assume that no neutrons leak from the system. Suppose that a unit source of neutrons of speed v is located at a point $\underline{r}'$, where $r' > r_0$. The uncollided flux at any other point $\underline{r}''$ is then $T(\underline{r}'', \underline{r}', v)$. Thus the number of first collisions which occur per unit time in the region $r'' < r_0$ is given by

$$N_1 = \int_{r' < r_0} T(\underline{r}'', \underline{r}', v) \sum_t (\underline{r}'', v) d\underline{r}'' \quad (2)$$

Since one neutron is produced per unit time, this expression also represents the fraction of neutrons born at $\underline{r}'$ which flow across the surface $r = r_0$. Since $vH'(\underline{r}', v)$ neutrons are born at $\underline{r}'$ per unit time, the total number of neutrons per unit time which flow across r_0 in the direction of decreasing r is given by

$$A(r_0)J_-(r_0, v) = \int_{r' > r_0} d\underline{r}' \int_{r'' < r_0} d\underline{r}'' \cdot vH'(\underline{r}', v) T(\underline{r}'', \underline{r}', v) \cdot \sum_t (\underline{r}'', v) \quad (3)$$

where $A(r_0)$ is the area of the surface $r = r_0$. Similarly, we deduce that

$$A(r_0)J_+(r_0, v) = \int_{r' < r_0} d\underline{r}' \int_{r'' > r_0} d\underline{r}''$$

$$\cdot vH'(\underline{r}', v) T(\underline{r}'', \underline{r}', v) \cdot \sum_t (\underline{r}'', v) \quad (4)$$

The net neutron current flowing across the surface is then

$$J(r_0, v) = J_+(r_0, v) - J_-(r_0, v) \quad (5)$$

If we consider a system with leakage out one surface, say $r = r_s$, the above expressions must be modified slightly. For a unit source of neutrons at $\underline{r}'$, the rate of first collisions in the entire system is

$$Q(\underline{r}', v) = \int_{\text{all } r''} T(\underline{r}'', \underline{r}', v) \cdot \sum_t (\underline{r}'', v) d\underline{r}'' \quad (6)$$

Since one neutron is produced per unit time, $Q(\underline{r}', v)$ is also the first-flight nonleakage probability for neutrons born at $\underline{r}'$. The corresponding leakage probability is defined by

$$L(\underline{r}', v) = 1 - Q(\underline{r}', v). \quad (7)$$

With leakage occurring through the surface $r = r_s$, the expression for $J_-(r_0, v)$, Equation (3), remains unchanged except that the $\underline{r}'$ integration now runs over $r_0 \leq r' \leq r_s$. The expression for $J_+(r_0, v)$ must be altered to account for those neutrons which cross the surface $r = r_0$ and continue on to leave the system through the surface $r = r_s$. Clearly, the appropriate expression is

$$\begin{aligned}
 A(r_0)J_+(r_0, v) = & \int_{r' < r_0} d\underline{r}' H(\underline{r}', v) \\
 & \cdot \left[L(\underline{r}', v) + \int_{r_0 < r'' < r_s} d\underline{r}'' \right. \\
 & \cdot vH'(\underline{r}', v) T(\underline{r}'', \underline{r}', v) \\
 & \left. \cdot \sum_t (\underline{r}'', v) \right] \quad (8)
 \end{aligned}$$

One other case remains, namely that of the double vacuum boundary slab. For this special case, it is necessary to have additional information about which side of the slab that the neutrons leak out. This information can be made available at the time of the $T(\underline{r}, r', v)$ matrix calculation. The left and right hand leakage is then given approximately by

$$L_-(\underline{r}', v) \approx$$

$$L(\underline{r}', v) \frac{T(\underline{r}_-, r', v)}{\left[T(\underline{r}_-, r', v) + T(\underline{r}_+, r', v) \right]} \quad (9)$$

and

$$L_+(\underline{r}', v) \approx$$

$$L(\underline{r}', v) \frac{T(\underline{r}_+, r', v)}{\left[T(\underline{r}_-, r', v) + T(\underline{r}_+, r', v) \right]} \quad (10)$$

where $T(\underline{r}_-, r', v)$ and $T(\underline{r}_+, r', v)$ are defined as the flux transport from

$\underline{r}'$ to the left hand and right hand boundary $\underline{r}_-$ and $\underline{r}_+$, respectively.

The modified current equations are then given by

$$\begin{aligned}
 J_-(r_0, v) = & \int_{r' \geq r_0} d\underline{r}' H(\underline{r}', v) \left[L_-(\underline{r}', v) \right. \\
 & + \int_{r'' \leq r_0} d\underline{r}'' T(\underline{r}'', \underline{r}', v) \\
 & \left. \cdot \sum_t (\underline{r}'', v) \right] \quad (11)
 \end{aligned}$$

and

$$\begin{aligned}
 J_+(r_0, v) = & \int_{r' \leq r_0} d\underline{r}' H(\underline{r}', v) \left[L_+(\underline{r}', v) \right. \\
 & + \int_{r'' \geq r_0} d\underline{r}'' T(\underline{r}'', r', v) \\
 & \left. \cdot \sum_t (\underline{r}'', v) \right] \quad (12)
 \end{aligned}$$

The effect of the improved current calculation on the diffusion coefficient has been determined for some plutonium fueled, H_2O moderated reactor systems. Calculated diffusion coefficients are compared in Table 2.II. Included are results obtained using the HAMMER code.⁽³⁾ The first column is from HAMMER, the second shows calculations from the foregoing theory and the third column shows values calculated, using the gradient calculation which was reported in the Battelle-Revised-Thermos document.⁽¹⁾

TABLE 2.II. Calculated Diffusion Coefficients Using the HAMMER and Battelle-Revised-Thermos Codes

Case	Diffusion Coefficients		
	HAMMER	Battelle-Revised-Thermos	
		T Calc.	-D∇φ
<u>Slab Geo.</u>			
MTR Phoenix Cell ⁽⁴⁾	0.267	0.250	0.671
<u>Cylindrical Geo.</u>			
EBWR Criticals			
0.55 in. Lattice Cell ⁽⁵⁾	0.290	0.263	0.416
Saxton Criticals			
0.80 in. Lattice Cell ⁽⁶⁾	0.236	0.195	
Saxton Criticals			
0.52 in. Lattice Cell ⁽⁶⁾	0.193	0.164	

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2. H. C. Honeck. THERMOS-A Thermalization Transport Theory Code for Reactor Lattice Calculations, BNL-5826, Brookhaven National Laboratory, Upton, N.Y. 1960.
3. J. E. Suich and H. C. Honeck. The HAMMER System Heterogeneous Analysis by Multigroup Methods of Exponentials and Reactors, DP-1064, E. I. du Pont de Nemours and Company, Wilmington, Delaware. January 1967.
4. G. J. Busselman. Unpublished Data. Pacific Northwest Laboratory, Richland, Washington. 1966.
5. L. C. Schmid et al. "Critical Masses and Bucklings of PuO₂-UO₂-H₂O Systems," Trans. Am. Nucl. Soc., vol 7, p. 216. 1964.
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IMPROVEMENTS TO THE BATTELLE-REVISED-THERMOS CODE

D. R. Skeen

Several improvements have been introduced into the Battelle-Revised-Thermos⁽¹⁾ code. The improvements and input revisions are listed below.

Revisions Introduced

Revised Current Calculation

An improved current calculation routine has been incorporated in the code. This routine is based upon the information contained in the transport matrix, $T(\underline{r}, \underline{r}', v)$. The complete theory is given in another section of this report.

Fission Cross Section Averaging

Fission cross section averaging both microscopically per isotope and macroscopically for the cell has been included in the editing routines.

Mixture Expansion

The code is now equipped to handle up to 8 mixtures.

Input Revisions

The following input has been revised in Battelle-Revised-Thermos.

---

3. (815)	NX,	Number of space points ≤ 30
	IX,	Number of speed groups ≤ 30
	MX,	Number of mixtures ≤ 8
	ISOX,	Number of isotopes used in cell from library $\leq 10^*$
	ISOXE,	Number of isotopes used in edit from library $\leq 10^*$
	ICX,	Number of isotopes used in cell not from library ≤ 10 -ISOX*
	ICXE,	Number of isotopes used in edit not from library ≤ 10 -ISOXE*
	NANI	= 1 for anisotropy correction to the scattering kernel
5. (OP2F5.0,	(WSTBA(J),	First identification of J^{th} isotope used in cell.
	WSTBB(J),	Second identification of J^{th} isotope used in cell.
6E10.5)	(CONCTA(J,	Concentration of the J^{th} isotope used in the cell
	M),M=1,MX),	in the M^{th} mixture.
	AMAS(J),	Isotope mass used in mubar calculation if first
	J=1,ISOX)	moment kernel is not available.
19. (515)	LEAKT,	a) < 0, Full slab with vacuum boundaries.
		b) = 0, Half slab with symmetry and right vacuum boundary or cylinder with vacuum boundary.
		c) = 1, slab or cylinder with reflecting boundaries.
		d) = 2, Right hand albedoes will be input.
	NXA,	Number of regions.
	NGEOM,	a) = 1, Regular cylindrical geometry with Argonne corrections.
		b) = 2, Fast cosine currents; cylindrical geometry; from SRL.
		c) = 3, Regular slab geometry with Argonne corrections.
		d) = 4, Fast Cosine currents; slab geometry; from SRL.
	NSMR,	+ 0 gives punched card output of a smeared cell which can be used for region input to a later case.
	NCUR	a) = 0, Current calculation using T matrix.
		b) = 1, Current calculation utilizing flux gradient calculation.

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* The only restriction on this is that $ISOX + ICX + ISOXE + ICXE \leq 30$.

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PROGRAM HTH

D. R. Skeen and W. L. Purcell

A computer program, entitled HTH, has been coded to provide a complete package of codes for use in reactor analysis. The program consists of an overlay of the Fortran-IV codes HRG,^(1,2) Battelle-Revised-Thermos,⁽³⁾ and HFN.⁽⁴⁾ The program also produces broad group cross sections from the HRG-Battelle-Revised-Thermos overlay portion for use in various diffusion and transport theory codes. A transmutation matrix for use in Program S-XIII⁽⁵⁾ is also provided so that burnup studies can be performed. The program HTH is written in Fortran-IV for use on the UNIVAC-1108. The code presently requires separate input for HRG, THERMOS, and HFN. A routine is being developed to simplify the input and make the code easier to use.

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BATTELLE REVISED-HAMMER CODE

L. J. Page and D. R. Skeen

The recent development of the HAMMER⁽¹⁾ code for cell calculations has provided the reactor physicist with an important tool. The HAMMER code performs a spatially dependent calculation in both the thermal and the nonthermal energy ranges. The thermal and nonthermal calculations are made using the techniques of THERMOS⁽²⁾ and HAMLET,⁽¹⁾ respectively. Modifications have been made to this code to make it more compatible with reactor codes in use at the Pacific Northwest Laboratory.

The primary change is that spatially dependent energy transfer matrices and absorption cross sections are now computed microscopically for cell regions and macroscopically for the entire cell. Other changes include a variable energy mesh for output cross sections (including multiple thermal groups) and punched cards for input to other codes. This revised HAMMER code is operational on the UNIVAC 1108. A report is being written which describes all of the revisions in detail.

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3.0 THERMAL REACTORS

PRCF-PHOENIX FUEL EXPERIMENT

R. I. Smith, E. C. Davis,
L. D. Williams, W. P. Stinson

Introduction

Utilization of ^{240}Pu as convertible poison in plutonium-fueled reactors is being studied under the Phoenix Fuel Program. As a demonstration of principle, a burnup experiment is being planned for the Materials Test Reactor (MTR) which will utilize a complete core of Phoenix Fuel. A critical mockup of the MTR has been assembled in the Plutonium Recycle Critical Facility (PRCF) which duplicated as closely as possible the physical configuration planned for the MTR experiment. The mockup has been fueled with Phoenix fuel (AL-20 wt% Pu, 20 at.% ^{240}Pu), and an extensive experimental program has been carried out to determine the characteristics of such a core. This is the third quarterly progress report describing the experimental program in the PRCF.

Summary

The measurements made in the PRCF-Phoenix core have provided information on the shutdown capability, kinetics parameters, and power distributions to be expected in the Phoenix fuel core in the MTR, for use in planning the burnup experiment. In addition, data have been obtained which provide an excellent test of the computational methods currently being applied to Phoenix fuel cores.

Reactivity Drift Problems

A series of exploratory measurements was initiated in the PRCF-Phoenix fuel core to isolate the cause of a slow loss of reactivity observed during the operation of the core. There were indications that the reactivity losses might be due to the small changes in the temperature of the H_2O coolant that occurred during the course of a day's work. Temperature distributions in the core were found to be extremely sensitive to the time that the H_2O was left in the reactor tank following the periods of operation. This was believed to be due to cooling on the uninsulated top and bottom reflector surfaces. Measurements made immediately following moderator mixing displayed consistent results and this procedure was incorporated as a standard technique for reactor operations.

Power and Flux Shape Measurements

An irradiation was performed with a 3 x 9 shimmed core (Figure 3.1) and the power distributions were determined in the fuel plates by scanning the fission product activities. These measurements were obtained by utilizing the gamma scanning facility at the PRCF. The gross fission activities in the plates, with energies greater than approximately 0.3 MeV, were measured using a NaI detector. The spatial resolution was defined by a steel collimator whose aperture was

FIGURE 3.1. PRCF - Phoenix Core Configuration

0.128 x 0.52 in. and whose length was 9 in.

Power profiles were obtained from the activities in the plates of standard fuel elements and in the plates of the fuel follower of a shim rod which was withdrawn to 59%. The measurements included horizontal and axial traverses. The horizontal traverse of plates from a standard element showed the characteristic power peaking at the edge, observed previously,⁽¹⁾ toward the beryllium reflector. The measured axial power distribution at the centerline of an exterior and an interior plate from a shim rod fuel follower is illustrated in Figure 3.2. The magnitude of the power peak observed in the exterior plate of the shim fuel follower was found to be no greater than the peak observed in an exterior plate of the fixed element, as shown in Figure 3.3. A measured axial power profile from an interior plate is compared in Figure 3.4 with the profile calculated⁽²⁾ using the WHIRLAWAY⁽³⁾ code. As shown, the calculation significantly overestimated the power peaking at the bottom edge of the core and, without the measurements, would have led to severe limitations on reactor power during the MTR burnup program.

Irradiations were also made with plates containing tapered fuel cores and these results are compared with the data obtained from a standard fuel core in Figure 3.5. Six plates containing tapered fuel cores were loaded into one element in the reactor. The fuel cores had tapers of 1 1/2 and 1/2 in., respectively, at opposite ends of the plates. An

irradiation was performed with these plates and gamma scanning measurements were made on the plate having the 1 1/2 in. taper at the bottom of the core. The plates were then rotated to place the 1/2 in. taper at the bottom of the core, and the irradiation and gamma scanning were repeated. These measurements determined the effectiveness of the tapered fuel in reducing the power spikes at the bottom of the core in a standard fuel plate. A reduction of at least 30% in the power peaking at the bottom edge of the tapered plates was observed, as compared to the standard fuel plate.

An irradiation was performed in this core to determine the horizontal power distribution at 6 in. below the core midplane. The plates were removed from position X-5 and the power distributions were obtained by scanning the fission product activities in the plates. Additional gamma scans were conducted to compare a plate having uniformly distributed plutonium to a plate which has a nonhomogeneous plutonium distribution. A copper pin irradiation was also performed to obtain a horizontal flux profile across the core and reflector at 6 in. below the core midplane. Additional copper pin activities were obtained for vertical flux profiles at selected locations in the beryllium reflector. Two plates from a shim rod fuel follower in location X-2 were gamma scanned to verify the power spiking effect that was observed previously. Analysis of these data is in progress.

FIGURE 3.2. Power Profile in Shim Fuel Follower

FIGURE 3.3. Power Profile in Exterior Shim Fuel Follower and Adjacent Fixed Fuel Element

FIGURE 3.4. Measured and Calculated Power Profiles in Phoenix Core

FIGURE 3.5. Power Profiles in Standard and Tapered Fuel Plates

Shim Rod Strength and Calibration Measurements

Calibration of one interior and one peripheral shim in the PRCF-Phoenix core was completed, with the remaining six shims banked at 63.6% withdrawn. The calibration was accomplished by incrementally inserting the interior shim (Shim X-4) from 63.6 to 38.9% while incrementally withdrawing the peripheral shim (Shim Z-8) from 0 to 100%. Preliminary examination of the differential worth measurements for the peripheral shim yielded a total worth of approximately 9\$. The last 2% withdrawal, from 98 to 100%, exhibited a reversal of worth, equal to -18.1¢. This phenomenon is believed to be the result of moving the lower end of the fuel follower out of the high thermal neutron flux just below the core, while the upper part of the fuel follower was being moved into the region at the top of the core where the flux is severely depressed by the banked shims. The worth of the interior shim appears to be about twice that of the peripheral shim, or about 18¢.

Measurements of the shutdown margin of the shim rods were completed. With all interior shims (X-4, X-6, Z-4, Z-6) fully withdrawn, the reactor was estimated to be approximately 3\$ subcritical. However,

withdrawal of two peripheral and two interior shims on the same end of the core (X-6, X-8, Z-6, Z-8) produced a critical core with a small excess reactivity.

Reactor Noise Measurements

Reactor noise data were recorded in this core, with detectors in three locations: center of the core (Position Y-5), in the beryllium reflector (Hole #1), and in the H₂O top reflector. No interpretable results were obtained from the chambers in the H₂O top reflector. Satisfactory results were obtained from the chambers in the beryllium reflector and in the center of the core. Results of the analysis of the recordings are listed in Table 3.I. Analysis of the data from the chamber in the beryllium reflector, recorded at critical only, produced results not significantly different from the chamber located at the center of the core.

Loss of Coolant Measurements

Measurements were made in the shimmed core to determine the reactivity worth of voiding coolant channels at selected locations throughout the core. Aluminum strips which

TABLE 3.I. PRCF-Phoenix Fuel Reactor Noise Analysis

Excess Reactivity	0.0\$	-0.88\$	-1.65\$
Core Center ($\frac{\beta - \rho}{\lambda}$)	62 $\pm$ 3	102 $\pm$ 4	143 $\pm$ 4

occupied an effective volume fraction of 0.8 were inserted into selected coolant channels. The results of these measurements can be summarized as follows: the reactivity changes due to loss of H₂O coolant were always negative, small near core edges and larger in the core interior. Final evaluation of these data is still in progress.

Other Selected Measurements

The incremental increase in worth of a shim rod fuel follower due to increasing the number of fuel plates from 12 to 13 was determined to be approximately +27¢, with the shim withdrawn to 90%.

Borated polyethylene strips (23 1/2 x 2 1/2 x 0.005 in.) were inserted into the coolant channels of the fuel elements to simulate equilibrium xenon poisoning. Distribution of 88 strips throughout the core produced a banked shim critical height of 62.8% withdrawn, or a poisoning effect worth about 15 mk. Measurements of regulating rod worth and axial power profiles were made with the poisoned core. Copper pin flux profiles were also measured at selected locations in the beryllium reflector. Analysis of the irradiation data is continuing.

A special core loading was constructed to obtain data from a configuration more amenable to calculation. A shim rod without fuel follower was placed in location Y-1 and a symmetric array of 15 5/8 fuel elements was loaded to obtain criticality. Irradiations were performed

in this core configuration to obtain bare and cadmium covered activation profiles axially, using gold and copper pin detectors. Some exploratory measurements were made using X-ray film to determine fission product activity profiles in the fuel plates. Examination of these films is continuing.

The core has been defueled and the core structure, complete with the beryllium reflector, has been removed from the PRCF reactor cell, to be stored intact.

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PRCF-PHOENIX ANALYSIS

C. M. Heeb

Critical approach experiments on small 8% ²⁴⁰Pu cores were performed prior to undertaking the critical experiments on an MTR-Phoenix core mockup. The analysis of these approach experiments⁽²⁾ has provided

a set of nuclear data for the core and reflector which gives good agreement with experiment. In addition to validating the nuclear data, the mesh sensitivities both in space and energy which can be expected to apply to Phoenix cores were determined.

The first critical experiment consisted of a rod-free clean core loaded approximately symmetrically into the center of the water-filled, rectangular core opening in the surrounding beryllium reflector. Fourteen and a fraction standard fuel boxes were required for criticality. Twenty-seven boxes would be required to fill the entire core gap in the reflector.

The second critical experiment was a full core load complete with eight control rods and their follower fuel. The approach was made by withdrawing all eight rods by the same amount (i.e., a banked configuration).

Upon examining the two experiments in light of available codes which would be used on this analysis, it was decided to design a third critical experiment which would involve a single control rod possessing as much top and bottom symmetry as possible; i.e., without follower fuel. The core was thus amenable to two-dimensional analysis through the core midplane and would allow the development of a workable control rod model. The second experiment required a full three-dimensional representation which would have restricted the analysis to only two energy groups (a code limitation). The earlier critical approach experiments indicated

that a strong dependence of the calculated k_{eff} upon the group structure could be expected with Phoenix cores; hence, the ability to use many groups on a rodged core became an obvious necessity. The third experiment was accordingly completed in which a control rod was positioned in the end of the reflector core slot midway between the longest sides. An approach was made by loading fuel around three sides of the rod and toward the center of the core. The critical size thereby determined was symmetric through the core midplane. The plane representation also has half-core symmetry.

The initial analysis has been concerned exclusively with first experiment. Spectra used in averaging the group cross section to be used in multigroup spatial calculations were obtained from the slowing down code HRG⁽³⁾ and the thermalization code THERMOS.⁽⁴⁾ The resonance treatment methods used in the epithermal calculation were identical to those in the approach experiment analysis. For the thermal calculation, the fuel side plates and structure extraneous to the active plate region was included in the unit cell by a two-pass method which is analogous to the method used in the approach experiment analysis. First, spatial fluxes were computed over the active section only. This was a slab calculation with alternating fuel, clad, and coolant regions. Next, another slab geometry calculation was performed for the water between boxes, the side plate, a

smear region of water and aluminum representing the alternating aluminum plate edges and coolant channels, and finally, the active region. Thus, the geometry of the second calculation was essentially at right angles to the first. In this last region, the entire meat-clad coolant unit cell nuclear densities were volume and flux weighted to form the homogeneous fuel-containing region. The fuels were obtained from the first pass slab calculation. The resulting average cross sections are then correct for the entire core unit, including the structure and coolant located beyond the fuel plate ends. Indeed the spectrum is sufficiently softened by the inclusion of these regions so that the average microscopic absorption cross section of ^{239}Pu is increased by 5%.

To most economically identify the essential features of a valid model, the strategy of starting with the simplest possible model and adding detail, both in space and energy, has been followed. Accordingly, a two-dimensional theory model in four energy groups and five regions yielded a value 6% above the experiment. A change to one-dimensional slab geometry was made in order to study the individual effects of increasing detail in the model. A two-region representation of the core which broke the previously homogeneous core into active and nonpower producing regions lowered the calculated k_{eff} by 1%. The explicit inclusion of the quarter-inch aluminum wall between the core and the beryllium reflector which had been

contained in the reflector by a nuclei-preserving smear lowered the calculated k_{eff} by 2%. Small changes in the beryllium reflector composition were found to have a negligible effect on k_{eff} . The spatial mesh of roughly one centimeter interval was found to be adequate.

The effect of the number of energy groups is currently under investigation, again using one-dimensional methods. From the approach experiments analysis, a lowering of calculated k_{eff} by 1 to 2% can be expected here.

Because these separate effects do not necessarily combine additively when included in a two-dimensional calculation, it is not possible to estimate just how good an agreement will finally be obtained. The investigation of the separated effects does allow a better insight into the relative importance of the features used in final model. It also prevents the inclusion of unnecessary, costly detail, and is relatively inexpensive to conduct in one dimension.

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PHYSICS CALCULATIONS FOR PLUTONIUM PHOENIX FUEL LOADINGS IN A MARITIME REACTOR

U. P. Jenquin and T. M. Traver

Calculations have been performed to determine the feasibility of fueling a maritime reactor with plu-

tonium Phoenix fuel. The maritime reactor considered was the Babcock and Wilcox (B&W) designed Consolidated Nuclear Steam Generator (CNSG-II). The reactivity, multiplication, and core lifetime characteristics were calculated for various plutonium Phoenix loadings in the CNSG-II. The results of these calculations were utilized in an economic evaluation of the various plutonium loadings.

FIGURE 3.6. Variation of Reactivity with Enrichment for 0.4 in. diam Rods

The reactivity and multiplication calculations were performed using the HRG, THERMOS, TEMPEST, and HFN codes while the burnup calculations were done with the ZODIAC (2+2) code. Parameters varied in the study were: enrichment, plutonium composition, moderator-to-fuel volume ratio, and rod size. A brief summary of the results are given here.

The initial reactivity is nearly independent of enrichment, as shown in Figure 3.6; however, the core lifetime is essentially proportional to enrichment as shown in Figure 3.7. The effect of increasing the amount of ^{240}Pu in the plutonium is to reduce the initial values of k_{∞} and k_{eff} and to reduce the core lifetime

as shown in Figures 3.6 and 3.7. However, the reduction in core lifetime is caused by the reduction in the fissile atom content (i.e., ^{239}Pu and ^{241}Pu). The system with 10 vol% enriched fuel and a plutonium composition of 43/40/10/7 contains 21% more fissile plutonium atoms than the 6 vol% enriched system with a plutonium composition of 65.5/23.3/7.7/3.5. As seen in Figure 3.7, the high ^{240}Pu content fuel has the lowest initial reactivity value and a core lifetime which is comparable to that of the 6 vol% fuel. Thus, the plutonium composition is an important consideration in the design of these systems.

FIGURE 3.7. Variation of Effective Multiplication with Irradiation for 0.4 in. diam Rods at $V_M/V_F = 1.5$

The effect of varying the moderator-to-fuel volume ratio and the rod size is relatively small on initial reactivity and core lifetime. The variation in initial reactivity with moderator-to-fuel volume ratio and rod size is shown in Figure 3.8. The effect of moderator-to-fuel volume ratio on the core lifetime is shown in Figure 3.9. As the moderator-to-fuel volume ratio is decreased, the amount of plutonium in the system is increased; thus

accounting for the increase in core lifetime. The rod size has almost a negligible effect on core lifetime (Figure 3.10).

The results of these calculations indicate that the enrichment and the plutonium composition are key parameters to the design of plutonium Phoenix fueled compact H_2O reactors. The economic impact of these considerations in the design of these systems should be fully exploited in any future studies.

FIGURE 3.8. Variation of Reactivity with Moderator-to-Fuel Volume Ratio and Rod Diameter for 10 vol% Enrichment

FIGURE 3.9. Variation of Effective Multiplication with Irradiation for 0.4 in. diam Rods with 10 vol% Enrichment

FIGURE 3.10. Variation of Effective Multiplication with Irradiation for 10 vol% Enrichment at $V_M/V_F = 1.5$

PHYSICS CHARACTERISTICS OF PLUTONIUM FUELED LOADINGS IN THE PRTR

U. P. Jenquin, R. C. Liikala, and
W. L. Purcell

A brief calculational study was made to determine the physics characteristics of plutonium Phoenix fuel in the Plutonium Recycle Test Reactor (PRTR). The fuel was assumed to be 19 rod clusters of aluminum-plutonium (Al-Pu) alloy, 88 in. in length. The isotopic composition of the plutonium was taken as 65.5/23.3/7.7/3.5% of $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/^{242}\text{Pu}$, respectively. The operating temperatures assumed for the plutonium Phoenix fuel loadings in the PRTR were 58 °C for the moderator,

260 °C for the coolant, and 372 °C for the fuel.

The calculations were performed using the codes HRG,⁽¹⁾ THERMOS,⁽²⁾ TEMPEST,⁽³⁾ HFN,⁽⁴⁾ and ZODIAC(2 + 2).⁽⁵⁾

The effective multiplication, k_{eff} , of a 55 element loading in the PRTR has been calculated for a range of fuel enrichments at operating temperatures. The values are plotted in Figure 3.11, along with values of k_{∞} . The difference in shape of the curves indicates the variation of neutron leakage with enrichment for these loadings.

The moderator, coolant, and fuel temperature coefficients of reactivity were calculated for the Al-20

FIGURE 3.11. Variation of the Effective Multiplication with Enrichment for a 55 Element Loading in PRTR at Operating Temperatures

wt% Pu fueled cell. The values are given in Table 3.II. The coefficients are all negative in sign and relatively small. The reactivity invested in the sum of moderator, coolant, and fuel heating is less than 1% $\Delta k/k$.

A calculation was also performed to determine the amount of boron required to control the excess reactivity of a 55 element loading of Al-20 wt% Pu elements. The result of the calculation shows that about 400 atoms of natural boron per million molecules of D_2O are required to control the 400 mk of excess reactivity. Thus, the worth of the boron is roughly 1 mk/atom part per million of natural boron.

A core lifetime calculation was performed for the 55 element loading of Al-20 wt% fuel in the PRTR. The reactivity loss rate with irradiation time is shown in Figure 3.12. The calculated lifetime of this core is 100,000 MWd. Reducing the length of the elements from 88 in. to 60 in. (the length of Batch Core fuel)⁽⁶⁾ results in a 35 to 40% reduction in core lifetime.

A burnup calculation was performed for a two region core containing 19

irradiated Batch Core fuel elements (UO_2 - PuO_2)⁽⁶⁾ surrounded by 66, Al-10 wt% Pu fuel elements. The UO_2 - PuO_2 fuel was assumed to have an exposure of 13,000 MWd/tonne. The calculated reactivity loss rate for this core is shown in Figure 3.13. The core lifetime was calculated to be 28,000 MWd and an additional 7,000 MWd/tonne average exposure is obtained on the UO_2 - PuO_2 fuel. Utilizing an Al-Pu fuel containing more plutonium (i.e., > 10 wt%) for this purpose would result in a slight decrease in the power density of the UO_2 - PuO_2 zone, but the core lifetime would be increased.

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TABLE 3.II. Average Temperature Coefficients of Reactivity

Type of Coefficient	Temperature Range, °C	Avg. Temp. Coeff.
		$\frac{1}{k_{\infty}} \frac{dk_{\infty}}{dT} (10^{-3}/^{\circ}C)$
Moderator	20-58	-0.123
Coolant	20-260	-0.003
Fuel	20-377	-0.007

FIGURE 3.12. Variation of Reactivity with Exposure for 55 PRTR Elements of Al-20 wt% Pu

FIGURE 3.13. Variation of Core Reactivity and UO_2 - PuO_2 Exposure with Time Assuming 70 MW Power

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PHYSICS CHARACTERISTICS OF THORIUM FUELED LOADINGS IN THE PRTR

U. P. Jenquin and W. L. Purcell

Physics calculations were performed to determine the feasibility

of a thorium loading in the PRTR. The physics characteristics of two different types of fuel elements were investigated along with the variation in enrichment and rod size. The fuel elements considered in the study were a metallic fuel comprised of nested cylinders of Th-4 wt% ²³⁵U, and ceramic (ThO₂-²³⁵UO₂) fuels in clusters of rods. The calculations were performed using the computer codes THERMOS,⁽¹⁾ HRG,⁽²⁾ and HFN.⁽³⁾ Some of the more important physics characteristics of the various fuel elements considered in the study are shown in Table 3.III.

The PRTR loadings assumed were 55 elements, 5 ft long, surrounded fully by D₂O. Survey calculations for these loadings were made with the ALTHAEA code. Results indicate that the 4 wt% metallic elements and the 5 wt% ceramic elements, 3/8 in. rods, could attain a total core exposure of 20,000 MWd. Since this is

TABLE 3.III. Physics Characteristics of Various Fuel Elements

	U Enrichment, wt%	k _∞	Conversion Ratio	Resonance Escape Probability	Thermal Power Density Ratio ^(a)
					Inner Ring Outer Ring
Metallic Nested Cylinders ^(b)	4.0	1.424	0.346	0.801	0.71
Oxide Clusters of Rods ^(c)					
19 rods, 1/2 in. OD	5.0	1.460	0.310	0.725	0.49
18 rods, 1/2 in OD	5.0	1.466	0.306	0.737	0.54
19 rods, 3/8 in. OD	5.0	1.499	0.278	0.833	0.66
18 rods, 3/8 in. OD	5.0	1.510	0.274	0.836	0.69
18 rods, 3/8 in. OD	7.0	1.597	0.204	0.808	0.63

(a) For the 19 rod clusters: Thermal Power Density in the 6 Rods adjacent to the Center Rod/Thermal Power Density in the Outer 12 Rods

(b) Density of 11.7 g/cm³

(c) Density of 8.5 g/cm³

about the exposure of interest, further reactivity calculations for these two systems were made. In reality the most enriched uranium that can be produced is about 93% ^{235}U . The fuel compositions were changed to reflect this for the additional calculations. Values of the multiplication, k_{eff} , are summarized in Table 3.IV. The leakage for a loading of 55 fuel elements in the PRTR is worth about -15% $\Delta k/k$ in reactivity.

The boron necessary to control the excess reactivity is in the range which has been previously used in the PRTR. The overall temperature coefficient of reactivity (Doppler, moderator, and coolant) is large and negative. The loss of coolant results in a loss in reactivity if D_2O is the coolant, but a gain in reactivity if H_2O is the coolant.

A refined burnup calculation on the metallic loading was performed using Program ZODIAC (2+2).⁽⁴⁾ The lifetime of a 55 element core was

calculated to be 15,000 MWd at which point the average exposure would be 14,000 MWd/tonne; peak exposure would be 27,000 MWd/tonne; and 5.0 kg of ^{233}U would exist in the core. At this exposure the ^{235}U will have been depleted to 55% of its original density.

Interest has been expressed in obtaining additional exposure on some of the UO_2 -2 wt% PuO_2 Batch Core elements. Calculations were made for various PRTR loadings consisting of thorium-uranium and uranium-plutonium fuels. The UO_2 - PuO_2 elements were assumed to have an average exposure of 13,000 MWd/tonne of oxide. Static calculations of reactivity and power distribution are summarized in Table 3.V at room temperature conditions, 100 ppm boron, and no exposure on the Th-U Fuel elements.

A burnup calculation with ZODIAC was performed for the arrangement of 19 UO_2 - PuO_2 fuel elements (13,000 MWd/tonne) surrounded by 66 Th-U

TABLE 3.IV. Multiplication Calculations for a Loading of 55 Fuel Elements in the PRTR

Reactor Conditions	Multiplication, k_{eff}	
	Th-4 wt% U Tubes	ThO_2 -5 wt% Cluster
Room Temperature	1.207	1.271
Room Temperature, 100 ppm Boron ^(a)	1.046	
Room Temperature, 120 ppm Boron ^(a)		1.049
Operating Temperatures	1.178	1.239
Operating Temperatures with H_2O Coolant	1.153	1.181
Operating Temperatures with no coolant		1.217

(a) Atoms of natural boron per million molecules of D_2O in the moderator.

TABLE 3.V. Multiplication and Power Distributions for Various Arrangements of Th-U Metal and UO_2 - PuO_2 Elements

Arrangement (a)	k_{eff}	Power Density Relative to 1.0 for 85 Elements		Exposure Ratios $\frac{\text{MWd/tonne on } \text{UO}_2\text{-PuO}_2}{\text{MWd/tonne on Th-U}}$ (b)
		Avg in Th-U	Avg in UO_2 - PuO_2	
55 Th-U, 30 void shroud tubes	1.046	1.55		
55 Th-U, 30 UO_2 - PuO_2	1.068	1.29	0.48	0.20
30 UO_2 - PuO_2 , 55 Th-U	1.001	0.93	1.12	0.66
19 UO_2 - PuO_2 , 66 Th-U	1.024	0.96	1.12	0.64
85 Th-U	1.079	1.00		

(a) Cold, 100 ppm boron

(b) The volume and mass of fuel is not the same for both fuel elements; the factor to calculate this number is 0.55.

fuel elements. The lifetime of this core is calculated to be 21,000 MWd. The Th-U metal would attain an average exposure of 12,000 MWd/tonne while the UO_2 - PuO_2 would attain an additional average exposure of 6,500 MWd/tonne.

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DETERMINATION OF k_{∞} FOR A URANIUM-LITHIUM SUPERCELL BY THE UNPOISONED TECHNIQUE IN THE PCTR

D. F. Newman

Introduction

The infinite medium neutron multiplication factor (k_{∞}) for the 2.1% enriched uranium-lithium target supercell has been measured in the Physical Constants Test Reactor (PCTR) by use of the null reactivity technique.⁽¹⁾ This method requires that the amount of thermal neutron absorber necessary to absorb the excess neutrons of the supercell lattice sample be accurately determined by a null reactivity condition of the test sample.^(2,3)

Additional measurements were made on the supercell to obtain data from which k_{∞} could be inferred by an unpoisoned technique. This technique gives information about the multiplication properties of the test lattice without requiring that absorber be added to the test cells.^(4,5) An

experimentally determined quantity, $\langle \Sigma_\rho \rangle$, defined as the adjoint weighted excess neutron production cross section was obtained that is directly comparable to multigroup calculations. Values for $(k_\infty - 1)$ were inferred from the quantity $\langle \Sigma_\rho \rangle$ by dividing by the average supercell thermal neutron absorption cross section and including a calculated leakage dependent term.

Results

A comparison of the values of k_∞ determined by the null reactivity technique and the unpoisoned technique is presented for each lattice condition and supercell configuration. Values of $\langle \Sigma_\rho \rangle$ are reported for all these measurements. The experimental results are presented in Table 3.VI.

Discussion

The method of determination of k_∞ from experimental data by the null reactivity technique has previously been reported in detail for the uranium-lithium supercell.⁽¹⁾ Development of the expressions utilized in the measurement of the adjoint

weighted excess neutron production cross section $\langle \Sigma_\rho \rangle$ has been shown⁽⁵⁾ to be given by:

$$\langle \Sigma_\rho \rangle_o = - \left(\frac{\Delta \rho}{\Delta \rho^a} \right) \frac{V_{a\ o a}}{V_{cell}} \frac{(A_{2a} + p A_{1a})}{A_{2a}^S} \\ = \left(\frac{S}{V} \right) \frac{1}{\phi_{th}^+ \phi_{th_o}} \sum_{i=1}^m \phi_i^+ J_i \quad (1)$$

The first expression is composed of quantities that can be determined experimentally, and the second expression can be obtained from multigroup calculations. The subscript on $\langle \Sigma_\rho \rangle_o$ means that this cross section is normalized to the 2200 m/sec cross section of the absorber, Σ_{oa} . The same normalization must be used in multigroup calculations to compare values of $\langle \Sigma_\rho \rangle$. The quantity p was obtained from computer code MOFDA:⁽⁶⁾ however, the values for $\langle \Sigma_\rho \rangle$ and k_∞ are not sensitive to this number since A_{1a} is small.

Values for k_∞ using the unpoisoned technique were obtained from the relationship:

$$k_\infty = 1 + \frac{\langle \Sigma_\rho \rangle \phi_{th} V_{cell}}{(\Sigma \phi V)_{th}} \frac{1}{cell} \\ + g(\Delta \phi, \Delta \phi^+) + \tau^2 B^4, \quad (2)$$

TABLE 3.VI. Summary of Results

Lattice Condition	k_∞ Null Reactivity Technique	k_∞ Unpoisoned Technique	$\langle \Sigma_\rho \rangle$ cm ⁻¹
6x6 Wet (Square supercell)	1.0035 $\pm$ 0.0008	1.0047 $\pm$ 0.0015	31 $\pm$ 3x10 ⁻⁶
5x5 Wet ("T" shaped supercell)	1.0001 $\pm$ 0.0033 - 0.0020	1.0001 $\pm$ 0.0030	0.67 $\pm$ 3x10 ⁻⁶
6x6 Dry	0.9984 $\pm$ 0.0018	0.9981 $\pm$ 0.0013	-13.3 $\pm$ 9x10 ⁻⁶
5x5 Dry	0.9850 $\pm$ 0.0008	0.9840 $\pm$ 0.0013	-114 $\pm$ 9x10 ⁻⁶

that has been extended to supercell measurements. The relative thermal neutron absorption rate in each cell of the supercell was determined from relative reaction rate measurements throughout the supercell. The relative absorption rate in the i th cell of the supercell is given by

$$(\text{Abs})_i = (\Sigma\phi V)_i \quad (3)$$

Thermal neutron spectrum averaged cross sections for each cell were calculated with computer code THERMOS⁽⁷⁾ using the experimentally determined equivalent cell boundary conditions.⁽¹⁾ The cell flux-volume product is obtained by rearranging Equation (3):

$$(\phi V)_i = \frac{(\text{Abs})_i}{\Sigma_i} \quad (4)$$

The supercell flux weighted thermal neutron absorption cross section is obtained from summation of the quantities in Equations (3) and (4):

$$\begin{aligned} \bar{\Sigma}_{\text{supercell}} &= \frac{(\Sigma\phi V)_{\text{th}}^{\text{supercell}}}{\phi_{\text{th}} V_{\text{supercell}}} \\ &= \frac{\sum_{i=1}^4 (\text{Abs})_i}{\sum_{i=1}^4 (\phi V)_i} \quad (5) \end{aligned}$$

In order to use the value of $\langle \Sigma_p \rangle_0$ in Equation (2), the supercell flux weighted cross section, $\bar{\Sigma}_{\text{supercell}}$, must have the average cross section of a $1/v$ absorber at the surface of

the cell boundary normalized to the 2200 m/sec value. This normalized supercell flux weighted cross section is given by

$$\bar{\Sigma}_0^{\text{supercell}} = \bar{\Sigma}_{\text{supercell}} \cdot \left(\frac{\sigma_{0a}}{\sigma_{\text{oa}}} \right)_{\text{boundary}} \quad (6)$$

so that Equation (2) can be simplified as follows:

$$\begin{aligned} k_{\infty} &= 1 + \frac{\langle \Sigma_p \rangle_0}{\bar{\Sigma}_0^{\text{supercell}}} \\ &\quad + g(\Delta\phi, \Delta\phi^+) + \tau^2 B^4 \quad (7) \end{aligned}$$

The function $g(\Delta\phi, \Delta\phi^+)$ is related to the flux mismatch between the fluxes in the test cavity and the equilibrium fluxes characteristic of the medium. This correction was ignored because analysis of the spectral measurements indicated that the mismatch was small. The leakage correction term, $\tau^2 B^4$, was evaluated using MOFDA⁽⁶⁾ to calculate L^2 and τ , and using age diffusion theory to obtain B^2 .

The quantities used in the unpoisoned technique k_{∞} determination are shown in Table 3.VII for the wet lattice (coolant water in the process tubes), and Table 3.VIII for the dry lattice (no coolant water). Thermal neutron flux profiles around the boundary of the central supercell are shown in Figure 3.14 for the square shaped supercell (6x6 lattice), and Figure 3.15 for the

TABLE 3.VII. Wet Lattice

Quantities Used in Unpoisoned Technique k_{∞} Determination

	6 x 6	5 x 5
$\Delta\rho_{\text{cell}}$	1.677¢	0.039¢
$-\Delta\rho^a$	2.063¢	2.187¢
$V_a \int_{oa}$	2.905 cm ²	2.905 cm ²
V_{cell}	8.0655 x 10 ⁺⁴ cm ³	8.0655 x 10 ⁺⁴ cm ³
A_{2a}	5.660	5.226
A_{1a}	0.417	0.355
ρ	0.9081	0.9081
$(\int \phi V)_{\text{supercell}}$		
ϕV	6.447 x 10 ⁻³ cm ⁻¹	6.447 x 10 ⁻³ cm ⁻¹
M^2	409.8 cm ²	409.8 cm ²
L^2	137.2 cm ²	137.2 cm ²
τ	272.6 cm ²	272.6 cm ²
B^2	8.53 x 10 ⁻⁶ cm ⁻²	8.53 x 10 ⁻⁶ cm ⁻²
A_{2a}^S	5.788	5.299

TABLE 3.VIII. Dry Lattice

Quantities Used in Unpoisoned Technique k_{∞} Determination

	6 x 6	5 x 5
$\Delta\rho_{\text{cell}}$	-0.654¢	-6.153¢
$\Delta\rho^a$	1.899¢	1.943¢
$V_a \int_{oa}$	2.905 cm ²	2.905 cm ²
V_{cell}	8.0655 x 10 ⁴ cm ³	8.0655 x 10 ⁴ cm ³
A_{2a}	5.2531	4.6986
A_{1a}	0.5333	0.4273
P	0.8985	0.8985
$(\int \phi V)_{\text{supercell}}$		
ϕV	7.1475 x 10 ⁻³ cm ⁻¹	7.1475 x 10 ⁻³ cm ⁻¹
M^2	474.6 cm ²	474.6 cm ²
L^2	125.8 cm ²	125.8 cm ²
τ	348.8 cm ²	348.8 cm ²
B^2	-3.37 x 10 ⁻⁶ cm ⁻²	-3.37 x 10 ⁻⁶ cm ⁻²
A_{2a}^S	5.3383	5.0693

FIGURE 3.15. Thermal Neutron Flux Around the Boundary of the Central Supercell Wet 5 x 5 Lattice

"T" shaped supercell (5x5 lattice). The PCTR external thermal column was used to normalize the relative magnitudes of the flux between the cell in and the cell out measurements. A 3% decrease in the average surface flux is observed when the cell was removed from the lattice in both supercell geometries. The values of k_{∞} using the unpoisoned technique agree with the values from the null reactivity technique to within 1.2 mk for all lattice conditions.

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4.0 FAST REACTORSPHYSICS STATICS CALCULATIONS FOR THE
FTR SPLIT CORE

L. L. Maas

A model of the FTR split core in x-y geometry has been developed for use in physics statics calculations. (See Figure 4.1). The model is a representation of a transverse cut at the core midplane. The 2DF transport theory computer code using this x-y

model was used to calculate critical mass, control rod worths, fluxes, and fission rates in the FTR. A four energy group cross section set derived from the PNL 26-group fast reactor cross section library was used for all calculations. The cross sections include an axial leakage correction.

Table 4.1 lists the characteristics of the FTR for a core with a uniform driver fuel and for a core with two

FIGURE 4.1. Schematic of FTR Split Core in X-Y Geometry

TABLE 4.I. Comparison of One and Two-Zone FTR Split Cores

	Unzoned	Zoned	
		Zone 1	Zone 2
Volume Fraction PuO ₂ ($\rho = 11.46$) (90/10, ²³⁹ Pu/ ²⁴⁰ Pu)	0.0743	0.0695	0.0834
²³⁸ UO ₂ ($\rho = 10.96$)	0.2687	0.2645	0.2686
Void	0.0610	0.0590	0.0620
Stainless Steel	0.2510	0.2560	0.2460
Sodium	0.3450	0.3510	0.3400
Split Volume, liters (includes test volume)	154.0	154.0	
Test Volume, liters	65.0	65.0	
Core Volume, liters	1000.0	500.0	500.0
Driver + 4 Open Loops	912.8	447.7	465.1
Control Rods	87.2	52.3	34.9
Core Length, cm	85.0	85.0	
Total Reflector Savings, cm	22.0	22.0	
k_{eff}	1.04	1.04	
²³⁹ Pu Critical Mass, kg	616.7	282.9	352.7
²³⁹ Pu Test Mass, kg	15.9	15.9	
Total ²³⁹ Pu Mass, kg	632.6	651.5	
²³⁸ U/ ²³⁹ Pu Ratio	3.859	4.061	3.436
Center Point Flux, $\frac{n}{cm^2 \cdot sec}$	5.920x10 ¹⁵	5.461x10 ¹⁵	

equal-volume driver fuel zones. In the zoned case, the fuel loading in each zone is adjusted to flatten the power profile.

For fully inserted rods, worths and peak-to-average values for the power and flux are shown in Table 4.II. The control rods are composed of 28.6 vol% B₄C, 31.4 vol% 304 SS, and 40.0 vol% Na; the follower composition is 40.0 vol% 304 SS and 60.0 vol% Na.

The worth of ²³⁹Pu for various positions at the core midplane is shown in Figures 4.2 and 4.3. The total flux and fission rate distributions at various positions in the core are shown in Figures 4.4 through 4.7

for unrodded and rodded configurations. Reference should be made to Figure 4.1 when using these plots. It should be noted that the flux is significantly dipped and the power distribution is tilted in the neighborhood of the control rods. Thus, any tests in a position adjacent to a control position will be influenced by the movement of control rods.

EFFECT OF BeO IN THE FTR

R. W. Hardie

The feasibility of adding BeO to the FTR to enhance the Doppler coefficient was surveyed. BeO was added

TABLE 4.II. Rods Worths and Peak-to-Average Power and Flux for FTR Split Core with Two-Zone Driver Fuel

	Rod Worth, $\delta k/\text{rod}$	Peak-to-Average ^(a)	
		Power	Flux
No Rods In	0.0	1.548	1.671
B_4C Rods in 4 ^(b)	-0.0131	1.483	1.604
B_4C Rods in 5	-0.0092	1.606	1.736
B_4C Rods in 9	-0.0021	1.560	1.681

(a) Includes 1.25 axial peak-to-average factor

(b) See Figure 4.1 for rod positions

FIGURE 4.2. ^{239}Pu Worth Versus Position in FTR

FIGURE 4.3. ^{239}Pu Worth Versus Position in FTR

FIGURE 4.4. Total Flux Versus Position in FTR Split Core

FIGURE 4.5. Total Fission Rate Versus Position in FTR Split Core

FIGURE 4.6. Total Flux Versus Position in FTR Split Core

FIGURE 4.7. Total Fission Rate
Versus Position in FTR Split Core

either uniformly to the driver zone (replacing $^{238}\text{UO}_2$) or to the reflector (replacing Inconel). Power peaks near the reflector were reduced using an Inconel sleeve. Calculations were carried out using the one-dimensional DTF-IV code for a 1150 liter core (1000 liter driver zone and 150 liter test zone). The reactor composition is given in Table 4.III.

The change in the Doppler coefficient for a $\text{BeO}-^{238}\text{UO}_2$ interchange in the driver zone is shown in Figure 4.8. Figure 4.9 illustrates the corresponding changes in k_{eff} for temperatures of 300 and 2100 °K.

The neutron source rate ($v\Sigma_f\phi$) in the FTR with a BeO reflector is shown in Figure 4.10 for various Inconel sleeve thicknesses. In Figure 4.11 the Doppler coefficient is plotted versus the Inconel sleeve thickness.

TABLE 4.III. Reactor Composition for
1150 Liter Core

	Test Zone	Driver Zone	Reflector	BeO Reflector
PuO_2	1.1	6.8	0.0	0.0
$^{238}\text{UO}_2$	4.3	24.5	0.0	0.0
Void	1.8	5.7	0.0	0.0
SS	17.5	26.4	0.0	0.0
Inconel	0.0	0.0	80.0	25.0
Na	75.3	36.6	20.0	20.0
BeO	0.0	0.0	0.0	55.0

Finally, Figures 4.12 and 4.13 give the Doppler coefficient and k_{eff} as a function of reflector thicknesses for a 2 cm Inconel sleeve.

The above analysis indicates that introducing BeO either into the driver zone or the reflector results in a moderate increase in the Doppler coefficient (≈ 10 to 20%). Since BeO in the reflector causes power peaking near the interface, Inconel

FIGURE 4.8. Doppler Coefficient for
BeO-²³⁸UO₂ Interchange in Driver Zone

FIGURE 4.9. Dependence of k_{eff} on
Volume Percent BeO for Temperatures
of 300 and 2100 $^{\circ}\text{K}$

FIGURE 4.10. Radial Neutron Source Rate for Various Inconel Sleeve Thicknesses

FIGURE 4.11. Doppler Coefficient for Various Inconel Sleeve Thickness

FIGURE 4.12. Doppler Coefficient as a Function of Reflector Thickness

FIGURE 4.13. Dependence of k_{eff} on Reflector Thickness

sleeves may be necessary to remedy this condition. However, such sleeves also adversely affect the Doppler coefficients.

RADIAL REFLECTOR CALCULATIONS

L. L. Maas

Because stainless steel is less expensive than Inconel, a construction cost reduction might be effected by replacing the Inconel with stainless steel or by decreasing the thickness of the proposed FTR Inconel/sodium radial reflector. The effect on k_{eff} of changes in both composition and thickness of the radial reflector has been investigated.

Transport theory calculations in cylindrical geometry with 26 energy groups have been performed. The radial reflector was considered to be either 15 or 30 cm thick and composed of 80 vol% of either Inconel or stainless steel with 20 vol% sodium. The effects on k_{eff} of variations in the radial reflector are shown in Table 4.IV.

TABLE 4.IV. Effect on k_{eff} of FTR Radial Reflector

Thickness, cm	Composition Material, vol%	δk_{eff}
30	Inc/Na, 80/20	0.0
30	SS/Na, 80/20	-0.0114
15	Inc/Na, 80/20	-0.0254
15	SS/Na, 80/20	-0.0430

As can be seen, the difference between a 30 cm Inconel reflector and a 15 cm Stainless Steel reflector is quite significant. A 4% loss in reactivity corresponds to about an 8% increase in critical mass and an 8% decrease in flux.

ENERGY DEPOSITION IN FTR

W. L. Bunch

A preliminary analysis was made to estimate the distribution of the energy deposited in the FTR and environs as a result of the fission process to provide a basis for the conceptual design of fuel. It was assumed the 1000 liter core will consist of mixed oxide fuel encased in stainless steel with sodium as the coolant. The actual deposition distribution will be dependent on the fuel pin geometry; therefore, this preliminary estimate can only define the gross features of the deposition distribution with later refinements to be made as the fuel geometry becomes established.

Composition of the reference core is given in Table 4.V, while results of the analyses are presented in Table 4.VI. The results are based on a total of 215.5 MeV per fission event and are normalized to a reactor power of 400 MW. It is estimated that about 90.1% of the fission energy will be deposited directly in the fuel, with the bulk of this energy coming from the heavy fission fragments. A total of about 95.3% of the energy will be deposited within

the core, with the other 4.7% being deposited in the reflectors and shields external to the core. No calculations were made to estimate heat generation within poison control elements or the perturbations caused by their presence.

On the basis of the uncertainty in available nuclear data together with the effect of simplifying approximations used in the analyses, it is estimated that the energy deposition within the fuel has an uncertainty of about $\pm 2\%$; whereas, the uncertainty associated with the other components is about $\pm 10\%$. These uncertainties can be reduced as the fuel concept becomes more firm and as more rigorous calculations are made; however, significant uncertainty will remain because of the available applicable nuclear data.

Details of this study are presented in Energy Deposition Distribution in the FTR, BNWL-563, by W. L. Bunch, dated September 1967.

TABLE 4.V. *Composition of Reference FTR Core*

<u>Element or Isotope</u>	<u>Atom Density, per barn-cm</u>	<u>Weight, kg</u>
^{239}Pu	0.001489	591.04
^{240}Pu	0.000165	65.95
^{238}U	0.005964	2357.4
Oxygen	0.015238	404.77
Chromium	0.004557	393.47
Iron	0.017444	1617.5
Nickel	0.001794	174.87
Sodium	0.008247	314.79

TABLE 4.VI. Energy Deposition
Distribution in the FTR

Source of Energy	Fuel	Steel	Sodium	Core Total	Outside Core	System Total
Fission Fragments	174.0	0	0	174.0	0	174.0
Decay Betas	6.3	0.7	0	7.0	0	7.0
Neutron Kinetic Energy	2.4	1.05	1.05	4.5	1.0	5.5
Gammas (Prompt, Decay)	7.8	5.0	0.7	13.5	1.5	15.0
Neutron Capture Energy	3.6	2.4	0.3	6.3	7.7	14.0
MeV/fission	194.1	9.15	2.05	205.3	10.2	215.5
Percent of Energy	90.1	4.25	0.95	95.3	4.7	100.0
Total MW	360.4	17.0	3.8	381.2	18.8	400.0

PARAMETRIC ANALYSES OF MAXIMUM HYPOTHETICAL ACCIDENTS

W. W. Little

The maximum possible nuclear accident in a fast reactor is extremely difficult to estimate. Perhaps the greatest uncertainty lies in predicting the failure mechanisms and initial conditions.

To get a preliminary estimate of the maximum nuclear accident, the reactor response to a wide range of reactivity ramp inputs and initial conditions has been computed. All calculations were performed by using the MAX⁽¹⁾ code. Briefly, MAX is a multizone, one-dimensional maximum accident code which employs a point kinetics model with fixed power and worth profiles. Linear and Doppler feedback is included. The hydrodynamics model consists of numerous compressible and movable mesh points.

All calculations were performed in spherical geometry for a one-zone reactor similar in size and composition to the FTR. The principal input parameters are given in Table 4.VII.

TABLE 4.VII. Input Parameters for
Maximum Accident Calculations

Core Radius	65 cm
Core Volume	1150 liters
Delay Fraction	0.0030
Neutron Generation Time	3.5×10^{-7} sec
Number of Precursor Groups	6
Inertial Density	5.4 g/cm ³
Fuel Density	3.4 g/cm ³
Melting Point	3000 °K
Heat of Fusion	280 J/g
Eq. of State	Taken from Menzies ⁽²⁾
Power and Worth Profiles	$1 - c(r/r_0)^2$, with a peak- average of 2
Hot Core:	
Initial Power	540 MW
Initial Core Temperature	3000 °K (solid)
Cold Core:	
Initial Power	0.54 MW
Initial Core Temperature	300 °K

Figure 4.14 shows the total energy as a function of the Doppler coefficient for various reactivity ramps in a cold reactor. The total energy, available work, and maximum pressure as a function of the Doppler coefficient in a hot core are given in Figures 4.15, 4.16, and 4.17. Cross plots of these graphs--that is,

FIGURE 4.14. Total Energy as a Function of Doppler Coefficient

FIGURE 4.15. Total Energy as a Function of the Doppler Coefficient

FIGURE 4.16. Available Work to 20 Atmospheres

FIGURE 4.17. Maximum Pressure Versus Doppler

total energy, available work, and maximum pressure as a function of ramp rate for various Doppler coefficients--are given in Figures 4.18 through 4.23.

A number of comments should be made on these results. First, note that the cold core excursions are quite sensitive to the Doppler coefficient; whereas, the hot core excursions are rather insensitive to the Doppler coefficient. This difference is caused, for the most part, by the inverse dependence of the Doppler coefficient on temperature. Also, observe that zero Doppler excursions are more severe in a cold core; whereas, moderate Doppler excursions are more severe in a hot core.

The oscillations in the energy release versus Doppler curves are caused by phase differences in the power and reactivity profiles. The oscillations can be more completely understood by observing the temperature, reactivity, and power as a function of time. These variables are shown in Figures 4.24, 4.25, and 4.26, respectively, for Doppler coefficients of -0.0045 and -0.005 and a reactivity ramp of 100\$/sec. Observe that the high Doppler case yields a higher energy release because it suffers one extra power pulse. In other words, n pulses are sufficient to trigger significant core expansion in the low Doppler case, whereas $n + 1$ pulses are necessary in the high Doppler case.

FIGURE 4.18. Total Energy as a Function of Ramp Rate

FIGURE 4.19. Available Work to
20 Atmospheres

FIGURE 4.20. Maximum Pressure Versus Ramp Rate

FIGURE 4.21. Total Energy as a Function of Ramp Rate

FIGURE 4.22. Available Work to 20 Atmospheres

FIGURE 4.23. Maximum Pressure Versus Ramp Rate

FIGURE 4.24. Average Core Temperature as a Function of Time

FIGURE 4.25. Total and Expansion Reactivity

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FIGURE 4.26. Relative Power as a Function of Time

ANALYSIS OF PHASE A EXPERIMENTS

S. L. Engstrom

Comparisons of calculated and experimental reactivity worths and reaction rates provide integral tests of cross section sets and theoretical models used in reactor design. Initial comparisons of this nature have been made for ^{239}Pu , enriched uranium, depleted uranium, ^{10}B , and stainless steel measurements made in the FTR Phase A Critical Experiments performed in ZPR-III Assemblies 48 and 48A. These are shown in Figures 4.27 through 4.30.

The calculated worths and reaction rates were computed using the small sample perturbation code, PERT-IV.⁽¹⁾ Flux and adjoint inputs for PERT-IV were calculated using a 2DF quarter core, x-y geometry model.

In summary, these comparisons illustrate that an unresolved difficulty exists in obtaining agreement between measured and calculated worths of small fissile samples. Further, more detailed spatial dependence of cross section sets is required to describe spatial worths and reaction rates near core-reflector boundaries and in reflector media.

FIGURE 4.27. ^{239}Pu Worth Core 48

FIGURE 4.28. Depleted Uranium Worth
Core 48A with Control Rods

Specific discussion and illustrative curves are presented below.

Calculated plutonium sample worths are higher than experiment (see Figure 4.27). This same discrepancy has been reported by several other laboratories.⁽²⁾

Calculated depleted uranium worths are also higher than the experimental worths as shown in Figure 4.28. This is thought to be a result of substantial uncertainties in the fission and absorption cross sections for ^{238}U .

FIGURE 4.29. B^{10} Absorption Rate
Core 48A with Control Rods

These cross sections are being re-vised. Where worths or reaction rates have a large thermal component, there tend to be large discrepancies between calculation and experiments.

It appears that neither core nor Inconel averaged cross sections are appropriate for these reflector calculations (see Figure 4.29). This is in part due to the geometry of

FIGURE 4.30. ^{238}U Fission Rate
Core 48A with Control Rods

the Inconel sector, since it is sandwiched between regions of ^{239}Pu and ^{238}U , both of which propagate hard spectra into the reflector. At no point in the Inconel sector is the transport neutron spectrum as soft as a fundamental mode calculation, assuming $B^2 = 0$. The actual spectra in the Inconel sector vary in softness between the core spectrum and the fundamental mode Inconel spectrum, but never closely approach either.

Low energy components of the neutron spectrum, which are nearly negligible in the core and blanket materials, are quite significant in

the reflector and are apparently not well reproduced by calculation. This is illustrated by comparing disagreement between experiment and calculation for ^{10}B capture as shown in Figure 4.29 with agreement for ^{238}U fission rate as shown in Figure 4.30.

In the rodded cores the low energy components of the spectrum are highly depressed in the rods and do not contribute strongly to the integrated reaction rates; hence, the reaction rates are predicted well by cross section sets averaged over the core neutron spectrum.

References

1. R. W. Hardie and W. W. Little, Jr. *PERT-IV, A Two-Dimensional Perturbation Code in FORTRAN-IV*, BNWL-409. Pacific Northwest Laboratory, Richland, Washington.
2. *Proceedings of the International Conference on Fast Critical Experiments and Their Analysis*, ANL-7320.

FTR PHASE B CRITICAL EXPERIMENTS

R. A. Bennett

Phase B experiments were conceived to provide experimental verification of nuclear design information for Title II design and for use in the Preliminary Safeguards Report on the FTR. It is expected that portions of the experimental results can also be used in the design of the Nuclear Proof Test Facility (NPTF). Detailed plans for Assembly I (a uniform cylindrical configuration) of these Phase B experiments are being formulated jointly with Argonne National Laboratory. Present plans call for the following configurations and measurements.

1. Core: right circular cylinder core; $H \approx 86$ cm, $R \approx 35$ cm
2. Side and end reflectors:
 $\Delta t = 30$ cm
3. Core Volume: ~ 322 liters
4. Core composition:
Fuel ≈ 36 vol%
Stainless Steel ≈ 25 vol%
Sodium ≈ 37 vol%
Void ≈ 3 vol%
5. Critical Mass ^{239}Pu : ~ 220 kg
6. Radial Reflector Composition:
Nickel ≈ 80 vol%
Sodium ≈ 20 vol%
7. Experiments:
  - a) Spectrum measurements at core center, core reflector interface, midway between the latter two, midway in the reflector.
  - b) Radial and axial measurements of the spatial distribution of the fission rates of ^{238}U and ^{239}Pu and capture rate of ^{10}B .
  - c) Radial and axial measurements of the spatial distribution of reactivity worths of small samples of ^{239}Pu , ^{238}U , and ^{10}B .
  - d) Distributed worths of depleted uranium, depleted U_3O_8 , and sodium will be measured in whole core experiments.
  - e) Large sample worths of core materials will be measured in the central drawers (P-16, 1, & 2) by the following series of reactivity measurements:
    - Core fully loaded
    - Central drawers (P-16, 1, & 2) filled with sodium. The Na will extend through both core and reflector.
    - Starting from critical remove the fuel and fertile material from the P-16 drawers and measure the reactivity change.
    - A series of three reactivity measurements is to be made as two identical $2 \times 2 \times 5$ in. (or a 7 in.) sections of core material are moved along the core drawers (P-16, 1, & 2) which have been filled with sodium. The positioning of the segments is to be symmetrical with respect to the ZPR-III split core interface.
  - f) Fuel compaction experiment.
  - g) Doppler coefficient of small samples at the core locations as indicated for the spectrum measurements.
  - h) Central fission ratios; ^{239}Pu : ^{235}U and ^{238}U : ^{235}U .

NPTF EVALUATION

Q. L. Baird

In response to an AEC request, additional quantitative information has been generated for justification of the NPTF. These data and additional information have been transmitted to the AEC in BNWL-559, Supplementary Information for FTR-NPTF Evaluation. This report contains two reference designs that are being evaluated at present for final selection and inclusion in the NPTF CFDD.

NPTF ACCIDENT ANALYSES

A. E. Waltar and R. A. Harris

A computer code (currently designated as MELT-I) has been developed to analyze possible meltdown patterns in the NPTF. Because core nodal temperatures are determined by a projection of temperatures accurately computed in only a very small region of the reactor, some comparisons of the MELT-I code were made with selected NUTIGER results.

Figure 4.31 illustrates the comparison of temperatures calculated by the two codes for the axial center of the average radial channel fuel pin in an unzoned NPTF. In the NUTIGER case, all temperatures in the entire pin were originally assumed to be at room temperature; whereas, in the MELT-I case, only the core volume averaged temperature was at room temperature. Other initial temperatures assumed axial and radial distribution according to the peaking factors used in the projection scheme. Hence,

there is a rather large temperature discrepancy between the two codes during the initial stage of the transient. The important region to compare, however, is in the vicinity of melting, since the purpose of the meltdown program is to determine when and how fuel might slump to cause a more reactive core configuration. In this melting region, the two codes seem to agree quite well. Other comparisons between the two codes, namely for center and outer channels of an unzoned NPTF and for ramp rates of 2 and 20 \$/sec, were made; and, again, the agreement near the vicinity of melting was favorable. It should be pointed out that the solid lines of Figure 4.31 were obtained from the MELT-I code by assuming zero reactivity feedback due to core collapse. This was necessary in order that a comparison could be made with NUTIGER results, since the latter code does not account for geometric changes in the core. In the actual analysis performed using the MELT-I code, temperatures beyond the melting point increased much more drastically due to dramatic power level increases resulting from the positive core collapse reactivity feedback.

Based upon the rather favorable comparisons discussed above, several transient conditions in the zoned NPTF were studied. External ramp rates of 1, 2, 4, 8, 20, and 40 \$/sec were investigated, usually assuming a Doppler coefficient of -0.004. For some of the lower ramp rates, a Doppler of -0.002 was used since for these cases it might be argued that enough heat transfer could take place

FIGURE 4.31. Comparison of MELT-I With NUTIGER for Temperatures at the Axial Center of an Average Radial Channel Fuel Pin (Ramp Rate = 4 $^{\circ}\text{F}/\text{sec}$)

in the transient to flash the sodium. It should be clearly emphasized, however, that the studies thus far are primarily of a parametric nature. They are not in any way to be interpreted as a credible flow of events.

The radial power and temperature distribution assumed for the zoned NPTF corresponds to that of Figure 4.32. The dip in the center is due to the relatively cold test loop and the peak at about two-thirds core radius is due to fuel zoning. Axial power and temperature distributions are assumed to follow a chopped cosine curve with a peak to average value of 1.25.

Figure 4.33 illustrates the reactivity components which are effective throughout the transient caused by an infinite external ramp of 4 $\$/\text{sec}$ (Doppler = -0.004). As usual, the negative Doppler term becomes effective in holding the net reactivity to slightly less than 1\$ after prompt critical is reached. At about 0.85 sec, some melting in the center of the hottest fuel pin begins; and, at 0.940 sec, melting begins in the outer fuel node (adjacent to the clad) in the hottest fuel pin. The latter event is assumed to cause local clad failure and the entire fuel column above that point is allowed to fall

FIGURE 4.32. Assumed Relative Radial Power and Temperature Distribution in the Zoned NPTF

FIGURE 4.33. Reactivity Components of 4\$/sec Ramp Imposed on Zoned NPTF (Doppler = -0.004)

freely under gravity. Radial and axial propagation of fuel melting causes further collapse, and the net feedback due to collapse is seen to quickly approach a positive ramp rate substantially larger than the externally applied ramp rate. Since perturbation theory is used to determine the collapse feedback, once a substantial amount of fuel has slumped, the model is no longer valid. Hence, beyond about 1.00 sec, Figure 4.33 is totally invalid. Figure 4.34 is a blowup of the collapse feedback during the time of interest. By taking the slope of this curve at various times after

collapse begins (the number in parenthesis indicating this time in milliseconds), one can determine the positive ramp rate due to core collapse as a function of time. These numbers are given in Figure 4.34 at 5 msec intervals.

Also shown on Figure 4.34 is an indication of peak fuel centerline temperatures in the reactor. Prior to the time when no fuel is above about 4000 °K, fuel vapor pressure is so low that an additional reactivity term (negative) due to core disassembly is not possible. After that point, however, such a feedback term

FIGURE 4.34. Reactivity Feedback
Due to Collapse for 4\$/sec Ramp
Rate in Zoned NPTF

should be included. Initial work indicates that the disassembly term becomes important once peak fuel temperatures are in the vicinity of 5000 °K. Hence, the question of when to transfer from the melting phase to the explosion phase of the analysis can be somewhat bracketed by starting the disassembly phase within the time intervals corresponding to peak fuel temperature between, say, 4000 and 7000 °K. For the 4 \$/sec case shown, the feedback term due to collapse is very close to a pure ramp of about 25 \$/sec in this region. Hence, although a second order term is used to fit the total reactivity insertion

curve within this region, the total energy release during the excursion is not too sensitively affected by the exact starting time of the disassembly code. It is quite important, however, to properly match core temperatures in going to the disassembly code because the Doppler term is so large. This may be difficult to do in the WEAKE explosion code since it does not allow for the heat of fusion in fuel. Therefore, another disassembly code, being developed by W. W. Little, may prove to be more useful than WEAKE since it does allow for the heat of fusion in fuel and,

thus, may allow for a better temperature (Doppler) match in going from the melting phase to the explosion phase.

Figure 4.35 is a pictorial representation of the core during the

melting phase. For the particular case considered, the core was divided into 18 axial segments and 10 radial channels. Each fuel pin was divided into 7 radial fuel nodes plus 1 node each for the clad and the coolant.

FIGURE 4.35. Melting Pattern in Zoned NPTF for Externally Applied Ramp Rate of 4\$/sec

Figure 4.35 then shows an R-Z cross section of the core (where the vertical skew is not considered), and the region within the heavy black lines represents the portion of the core melted (actually, it represents the boundary corresponding to outer fuel node temperatures of 3000 °K). Times given correspond to those enclosed in parenthesis in Figure 4.34, i.e. they are the times after melting and collapse begin. The two hot spots on either side of the core center correspond to the two hot spots of Figure 4.32.

The area within the single cross-hatched boundary (starting at $t = 20$ msec) indicates the portion of the core where fuel temperatures in excess of 4000 °K exist. Finally, the double crosshatched area represents the core region which contains fuel temperatures in excess of 5000 °K. As discussed earlier, the negative feedback due to core disassembly begins to overwhelm all other feedback effects when fuel temperatures reach approximately 5000 °K so that the core pictures beyond $t = 30$ msec are certainly not valid. The core will

have disassembled prior to those times.

Preliminary results from the disassembly phase have been generated; but since the sensitivity analysis of various equations-of-state, etc. is not yet complete, the overall total energy generation numbers corresponding to the various accidents considered will be reported at a later date.

NEUTRON NOISE THEORY INCLUDING FEEDBACK

J. R. Sheff

As a basic part of an attempt to detect anomalous behavior in the FFTF through continuous on-line measurements of the neutron noise, the basic theory of neutron fluctuations in a system involving reactivity feedback is being studied.

The linearized neutron noise problem including linear feedback has been solved via the Langevin approach. The particular case picked, which is really a nonessential part of the solution, is that of a linear reactivity temperature feedback. The solution is

$$\psi_n(\omega) = \frac{F_n^*(\omega) F_q(\omega) + \frac{\psi F_q(\omega)}{\omega^2 + \alpha_q^2}}{\left(\alpha_n(\omega) - \frac{\alpha_q N k^2 \gamma}{\bar{v} l^2 (\omega^2 + \alpha_q^2)} \right)^2 + \omega^2 \left(\gamma_n(\omega) + \frac{N k^2 \gamma}{\bar{v} l^2 (\omega^2 + \alpha_q^2)} \right)^2} \quad (1)$$

where the usual no-feedback spectrum is

$$\psi_n'(\omega) = \frac{F_n(\omega)}{\alpha_n^2(\omega) + [\omega \gamma_n(\omega)]^2} \quad (2)$$

and N = mean neutron density

λ = prompt neutron lifetime

γ = reactivity temperature coefficient

α_q = ratio of the heat transfer coefficient to the heat capacity of the material changing temperature

q = "temperature" in energy units of a single fission

$\Psi^F_q(\omega)$ is the spectral density of the underlying temperature fluctuation processes and $\langle F_n^*(\omega) F_q(\omega) \rangle$ is the mixed spectrum between the underlying processes for neutron fluctuations and temperature fluctuations. The latter quantity will almost certainly

be zero, while $\Psi^F_q(\omega)$ will undoubtedly be nonzero and will probably

dominate over the $\Psi^F_n(\omega)$, the spectral density of the underlying neutron fluctuation processes, in Equation (1). No further speculations

have as yet been made about $\Psi^F_q(\omega)$.

The first term of Equation (1),

that containing $\Psi^F_n(\omega)$, is the feedback-noise spectral density resulting from only the fluctuations which have a basis in the characteristic stochastic behavior of the neutron portion of the problem. The feedback is deterministic in nature. This term has been included into an existing space dependent neutron noise program NOISY1, in a space independent fundamental mode sense. The resulting feedback spectrum approximately representative of the FTR is shown

in Figure 4.36 along with the no-feedback spectrum. Critical and three dollar subcritical curves are shown. The effect is a depression producing here a minimum in the spectrum below the frequency α_q which equals 0.6 in the calculation. The magnitude of this depression is proportional to the mean neutron density and the temperature coefficient, and inversely proportional to the quantity α_q .

These results are in at least qualitative agreement with those obtained by McGrath using the noise analysis approach of Rice to the problem. The advantages of the present method are (1) it results in an analytical expression for the result as opposed to an expression which required a numerical integration because it appeared to be beyond simplification, (2) it gives a much clearer understanding of the basic processes involved and, hence, enables intuitive estimates to be made about aspects of the problem beyond our current experimental and theoretical reach, and (3) it points to a clear-cut method for attempting the full nonlinear problem for the case of linear feedback.

Work is proceeding on a numerical comparison with the results of McGrath and on the development and/or guessing of at least some functional shapes for the feedback reactivity noise sources. Some work has been done on the nonlinear problem, and its solution appears feasible after better information on at least the third neutron density moment is developed.

FIGURE 4.36. Comparison of the Auto Spectral Density Including and Excluding Feedback Effects for Critical and 3% Subcritical Cases

A second key part of the determination of the feasibility of anomaly detection through neutron noise measurement concerns characterizing the neutron detector. Historically, one key parameter which has been used to characterize a necessary detector property has been

$$\epsilon = \frac{\text{counts per second in the detector}}{\text{fissions per second in the reactor}}$$

Such a definition is based upon a space independent model, and the predictions of the range of necessary values has not exactly overlapped the range of experimentally determined values in successful experiments. One has found that values in the range of 10^{-5} to 10^{-6} seem necessary though for zero power experiments. For high power experiments the situation is very much less clear. A direct application quickly leads one to necessary counting rates which are unattainable. It is felt that this is primarily a problem of the model, however; but the question is what must one actually know to specify the detector requirements? Some work is proceeding along the line of looking at equivalent quantities based upon space dependent analysis, and there is work being done elsewhere along the lines of a variational principle reduction of the space dependent problem to the space independent form. Hopefully, one or both of these approaches will yield a workable expression for the ratio of correlated-to-uncorrelated events in a measurement which is the fundamental quantity needed.

Before one can conclude whether anomalous behavior can be predicted, there still remains considerable effort upon the items discussed above as well as upon identification of expected types of mechanical and other reactivity noise perturbations. One then must also postulate various anomalous situations and evaluate whether they could be observed.

ANALYSIS OF SODIUM SHIELD EXPERIMENTS

D. R. Marr

Experimental measurements have been made of the penetration of fission radiations through sodium to provide a basis for the incorporation of the effects of this material in the design of the FTR shield.⁽¹⁾ In the experimental arrangement, a uranium disc 24 in. in diameter and 60 mils thick provides a source of fission neutron and photons. One-foot thick and five-foot square sodium slabs, which were cast in 1/4 in. thick aluminum cans, were placed adjacent to the disc source to form a 10 ft thick sodium shield. Radioactivants placed between the slabs provide a measure of the spectral and spatial distribution of the neutron flux. The following reactions were used:

$^{115}\text{In}(n,\gamma)^{116}\text{In}$	(1.46 eV resonance)
$^{197}\text{Au}(n,\gamma)^{198}\text{Au}$	(4.91 eV resonance)
$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	(132 eV resonance)
$^{23}\text{Na}(n,\gamma)^{24}\text{Na}$	(2.85 keV resonance)
$^{115}\text{In}(n,n')^{115\text{m}}\text{In}$	(about 1.2 MeV threshold)
$^{32}\text{S}(n,p)^{32}\text{P}$	(about 3.0 MeV threshold)
$^{27}\text{Al}(n,\alpha)^{24}\text{Na}$	(about 8.1 MeV threshold)

The transport code ANISN was used to calculate the spectral and spatial distribution of the neutron flux from a point fission source using spherical geometry. The same atomic composition as employed in the experiment was used in the calculations, both as a homogeneous mixture of aluminum and sodium and also in a layer arrangement. It was found that the effect of the thin aluminum container walls was small, so that this material can be included adequately using the simpler homogeneous method.

For comparison of the ANISN calculations to the experimental data, a computer program was written to integrate the point source results over a disc source. The resultant flux distribution was then used to calculate activation rates for the seven reactions listed above. Results of the calculations are presented in Figures 4.37, 4.38, and 4.39 for those materials in which it was possible to measure absolute, rather than relative, activity.

At the higher energies measured by the threshold reactions, there is good agreement between the calculated values and the experimental results. At the higher resonance energies, the calculations realistically predict the experimental shape; but there is significant disagreement in the absolute activity for the sodium samples. This disagreement may be due to the large contribution to the activation from lower energies for which the slowing down treatment may be inadequate or possibly to the impor-

tance of the sodium resonance in establishing the spectral distribution. Studies are in progress to refine the group structure in the sodium resonance and lower energies to determine the effect on the calculated distribution and activation. At the lower energies the agreement is less satisfactory for another reason; the shapes of the calculated curves are not in agreement with the experimental results. In addition to determining the effect of using a more detailed energy group structure, calculations are also being made using a two-dimensional transport code (2DF) to more rigorously incorporate the boundary conditions at the fission plate source.

The calculations to date were made using 27 energy groups. Anisotropic scattering was considered using both the first and third Legendre coefficients (P_1 and P_3); whereas, angular distribution calculations were made using both the S4 and S16 approximation. Cross section information for sodium is being reviewed to incorporate latest results. The purpose of this work is to establish a valid calculational technique for evaluating FTR shield concepts and for use in establishing the FTR shield design.

Reference

1. A. W. Thiele. *Neutron and Gamma Ray Attenuation in Sodium Shields*, NAA-SR-MEMO-12467. North American Aviation, Inc., Downey, California, June 1967.

FIGURE 4.37. Threshold Activations Through Sodium Shield

FIGURE 4.38. Resonance Activations Through Sodium Shield

FIGURE 4.39. Resonance Activations Through Sodium Shield

ANISN-MAC SHIELDING CALCULATIONS

C. A. Mansius

We compared the two one-dimensional shielding codes MAC, a removal diffusion calculational code, and ANISN, a discrete ordinate transport code, by calculating the relative fast epithermal lower axial neutron flux emanating from the reactor core through the tube sheet for a conical FFTF reactor core con-

cept. The depth of the shield penetration calculations is 150 cm or about 5 ft. We made an intercomparison of ANISN by using anisotropic scattering through the first and third Legendre coefficients (P_L) along with S_4 and S_{16} approximations in the solution of the transport equation.

For calculational purposes all shield regions are taken as homogeneous mixtures consisting of sodium

nickel, iron, and chromium in varying percentages. Similar 26 neutron energy group structures were used in both methods; however, the cross section sets for MAC and ANISN were derived independently.

Figures 4.40 and 4.41 show plots of relative fast flux (0.1 MeV) and epithermal flux (0.4 eV to 0.1 MeV) through 150 cm of shield for the MAC and ANISN cases. No significant differences in flux distributions

FIGURE 4.40. Fast Flux

were calculated with ANISN by varying the order of anisotropic scattering terms (P_L) or the number of approximations (S_n) in the solution of the transport equation. The maximum P_L and S_n used was P_3 and S_{16} . (See Curve 1, Figures 4.40 and 4.41).

After a 150 cm shield penetration for infinite slab geometry, the MAC calculation predicts both lower fast and epithermal flux values than ANISN. However, a finite geometry ANISN calculation (to correct for traverse leakage) predicts lower

FIGURE 4.41. Epithermal Flux

fast flux in the tube sheet than the MAC infinite slab calculation (Figure 4.40, Curves 2 and 3).

A comparison of calculational methods and cross section sets with experimental data is required to establish the validity of the results. A mockup experiment of this approximate shield arrangement is planned this fiscal year in the Shield Test and Irradiation Reactor.

HEAT GENERATION IN SHIELD REGIONS

C. A. Mansius

An estimate was made of nuclear heating rates in the radial neutron shield and in the axial cover shield for use in the conceptual design of shield cooling systems. The estimates are based on calculations using the one-dimensional removal-diffusion code MAC. This code first calculates the neutron flux distribution throughout the shield regions, uses this distribution to provide the gamma source distribution, then calculates the gamma distribution using the Taylor buildup approximation. The gamma distribution provides the basis for nuclear heating in the shield materials.

Calculations were made for a typical split conical core arrangement operating at a power level of 400 MW. Radial shielding between the core and the neutron shield was assumed to consist of a 2 ft thick reflector shield composed of stainless steel and nickel and cooled by sodium, a 2 ft thick annular ring of sodium plus a total of about 5 in. of steel in the shroud and reactor

vessel walls. Axial shielding between the core and the cover shield was assumed to consist of a diverging array of stainless steel tubes through the sodium pool for a total thickness of about 18 ft.

The maximum (midplane) heat deposition rate in the neutron shield was calculated to be 0.035 W/cm^2 , indicating that nuclear heating in the entire shield would be of the order of 10 kW. Nuclear heating in the cover shield was calculated to be less than $30 \text{ } \mu\text{W/cm}^2$ for a total rate of less than 10 W. These values do not indicate the required capacity of the cooling systems for these shield regions, since sources of heat other than direct nuclear heat will exist.

DECAY HEAT FROM IRRADIATED FTR FUEL

W. L. Bunch

Decay heat from irradiated FTR driver fuel was estimated using the computer program RIBD. For this study, the following FTR characteristics were assumed:

1. Reactor power level of 400 MW
2. Probably goal exposure for the fuel of 45 MWd/kg with a possible range from 35 to 80 MWd/kg
3. A total of 78 driver fuel channels with a possible range between 70 and 88
4. Approximately 2.95 tonnes of metal (uranium and plutonium) in the core with a possible range from 1.8 to 3.3 tonnes.

Using these assumed characteristics, input parameters were defined for

three RIBD calculations to estimate decay heat from a driver fuel element:

1. "Best Estimate"--5.31 MW/channel for 332 operating days
2. "Upper Limit"--5.72 MW/channel for 600 operating days
3. "Lower Limit"--4.55 MW/channel for 158 operating days.

These values apply to average fuel for the specific assumed conditions. In addition, a peak-to-average value of 1.25 has been used to define the hottest element for each case and a factor of 0.8 has been applied to define the coldest. Results of the calculations are presented in Figures 4.42, 4.43, and 4.44 as a function of time after shutdown.

The most important factor in establishing the level of the decay heat will obviously be the operating power per channel, which is set by the total reactor power and the number of channels. Based on the design operating efficiency of 75%, goal exposure in the range considered will not be an important factor in sizing heat removal requirements for the fuel handling equipment, nor will the actual length of each operating cycle be important.

RADIOACTIVITY ASSOCIATED WITH FTR FUEL

W. L. Bunch

Radioactivity associated with unirradiated fuel for the FTR can influence the design of facilities in which this material will be fabricated, handled, stored, or utilized. In addition to radiation exposure received by operations, heat gener-

ation is also an important consideration, especially in the design of NPTF. Half lives and decay energies are known with sufficient accuracy to estimate heat generation as a function of the isotopic composition of the fuel; however, data for estimating the radiation environment have not been so well established.

Heat generation for each of the isotopes that might be found in FTR fuel is shown in Table 4.VIII. These values are all associated with the decay process; the contribution from spontaneous fission is negligible in comparison. Both ^{237}U and ^{241}Am result from the decay of ^{241}Pu . Because of the short half life of ^{237}U (6.75 days), the equilibrium decay rate that is tabulated will be attained within a few weeks following chemical processing of the material. Because of the long half life of ^{241}Am (458 years), the contribution from this isotope will continue to increase for several years after

TABLE 4.VIII. Heat Generation of Fuel Isotopes

Isotope	W/g
^{237}U (a)	$1.4 \cdot 10^{-5}$
^{238}U	$8.2 \cdot 10^{-10}$
^{238}Pu	$5.5 \cdot 10^{-1}$
^{239}Pu	$1.8 \cdot 10^{-3}$
^{240}Pu	$6.9 \cdot 10^{-3}$
^{241}Pu	$2.7 \cdot 10^{-2}$
^{242}Pu	$1.1 \cdot 10^{-4}$
^{241}Am (a)	$1.06 \cdot 10^{-1}\text{f(t)}$

(a) Heat generation per gram of ^{241}Pu .

FIGURE 4.42. Decay Heat from FTR
Fuel Element as a Function of Time--
"Est Estimate"

FIGURE 4.43. Decay Heat from ETR
Fuel Element as a Function of Time--
"Upper Limit Estimate"

FIGURE 4.44. Decay Heat from FTR
Fuel Element as a Function of Time--
"Lower Limit Estimate"

processing. For the first three years, the buildup function for this isotope can be approximated by:

$$f(t) = 1.0 - \exp(0.053t)$$

where t is the number of years since processing.

For the prototype fuel being considered for FTR and for the NPTF, heat generation from the decay process in the unirradiated fuel will be about 2.2 kW per core loading. Heat generation for any quantity and isotopic composition of fuel can be estimated from the values in Table 4.VIII.

For an oxide fuel, the neutron production by spontaneous fission and (α, n) reactions is given in Table 4.IX for the isotopes of interest in FTR fuel. The neutron flux that might be incident on an operator in the vicinity of a relatively small fuel sample of known composition and weight can be estimated readily from the source strength and geometric considerations. For large arrays of material, attenuation and multiplication effects would need to be considered.

Gamma radiation is emitted from the fuel isotopes as a result of both alpha decay and spontaneous fission. The low energy photons associated with the decay process contribute significantly to the surface exposure rate associated with the fuel, but are relatively easy to shield. The higher energy photons associated with spontaneous fission are much less abundant and become important only for large, shielded arrays of fuel. The decay products of ^{241}Pu can play

TABLE 4.IX. Neutron Production in Oxide Fuel

Isotope	W/g
$^{237}\text{U}^{(a)}$	0
^{238}U	$1.4 \cdot 10^{-2}$
^{238}Pu	$1.64 \cdot 10^4$
^{239}Pu	$4.6 \cdot 10^1$
^{240}Pu	$1.17 \cdot 10^3$
^{241}Pu	3.4
^{242}Pu	$1.8 \cdot 10^3$
$^{241}\text{Am}^{(a)}$	$2.5 \cdot 10^3 f(t)$

(a) Neutron generation rate per gram of ^{241}Pu , where
 $f(t) = [1.0 - \exp(-0.053t)]$
 for $\leq t \leq 3$ years.

a very important role in establishing the gamma intensity in the vicinity of plutonium fuel; therefore, if significant ^{241}Pu is present in the FTR fuel, the gamma intensity associated with the unirradiated fuel will continue to increase for several years.

CROSS SECTION ANALYSES

R. E. Schenter

Fission Product Capture Cross Sections

Work is continuing on the calculation of one group (FTR spectrum weighted) average capture cross sections for the 450 fission product isotopes used as input in program RIBD. RIBD calculates the buildup of fission products within the reactor fuel. Current investigations have concentrated on isotopes neighboring those which have significant radiological effects. These include the isotopes of krypton, strontium, zirconium, iodine, xenon, cerium, barium, and

lanthanum. Figure 4.45 shows typical results for the FTR spectrum weighted capture cross section of the stable isotope ^{139}La which produces the important, from a radiological standpoint, isotope ^{140}La .

Optical Model Potential Studies

Theoretical calculations of the capture cross section for fission product isotopes involve the use of optical model potentials and their resulting single particle wavefunctions. Consequently, investigations have been continued on the construction of the optical potential in terms of the nucleon-nucleon

potential and the target nucleus wavefunction. Initial studies have shown important differences arise in single particle wavefunctions generated from the theoretically constructed optical potential in comparison with those given by the usual phenomenological forms. Figures 4.46, 4.47, and 4.48 show these results for targets of ^{12}C , ^{16}O , and ^{40}Ca , where V_{op} corresponds to the theoretical potential and Perey-Buck⁽¹⁾ the phenomenological local and nonlocal potentials.

Reference

1. F. Perey and B. Buck. Nuclear Physics, vol 32, p. 353. 1962.

FIGURE 4.45. FTR Spectrum Weighted
Capture Cross Section of ^{139}La

FIGURE 4.46. Single-Particle Wave-
 functions for Neutron-Nucleus
 S-Wave Scattering, Carbon-12
 Target-1 ($E = 3.8$ MeV)

FIGURE 4.47. Single-Particle Wave-
 functions for Neutron-Nucleus S-Wave
 Scattering, Oxygen-16 Target-1 (E
 = 3.7 MeV)

FIGURE 4.48. Single-Particle Wave-
 functions for Neutron-Nucleus
 S-Wave Scattering Calcium-40
 Target-1 ($E = 3.6$ Mev)

5.0 NEUTRON PHYSICSLEAST-STRUCTURE COMPUTER PROGRAM

D. G. Foster, Jr.

The least-structure technique is a method for obtaining the inverse convolution of a set of data which exhibit appreciable statistical scatter. For a data-vector y taken with resolution matrix R , we seek a solution x with the properties:

- 1) x is "as smooth as possible."
- 2) The resolution-broadened function Rx differs from y in a manner consistent with the known covariance matrix of the elements of y .

Condition (1) is fulfilled by minimizing the integral, over the interval in t which is covered by the data, of the function $(d^n x/dt^n)^2 w$, in which w is a weighting function. The original development by B. C. Cook⁽¹⁾ employed $w = 1$ and $n = 1$ or $n = 2$. He demonstrated that the method works remarkably well when the resolution function is the pathologically-broad bremsstrahlung spectrum, and that it makes little difference which value of n is chosen.

The BNW computer program for performing least-structure calculations is now operating well enough to permit further development of the method. The application under consideration is quite different from that considered by Cook, since it is applied to a narrow resolution function and pathologically-sharp data (specifically, a set of sharp peaks superimposed on a very low background which contains one very weak peak).

Preliminary results include the following:

- 1) The original method does not conserve counts. This deficiency has been removed by modifying the theory, which fortunately can be done in closed form without significantly increasing the time required for solution. Unfortunately, making the method conservative does not significantly improve the solution in other respects.
- 2) Cook's original prescription deals very adequately with the background region by itself. However, a case including both peaks and background breaks into severe oscillations in the background region, with many points exhibiting a negative number of counts. These oscillations permit Rx to follow y too closely in the background region, where the random errors in the data are large relative to the data themselves. This is compensated by disproportionately large values of $Rx - y$ in the region of the peaks, which are therefore smoothed too much.
- 3) This difficulty could be mitigated if a weighting function such as $w = 1/x^m$ were used. In this case, $m = 0$ corresponds to Cook's unweighted method, and $m = 2$ yields a definition of "smoothness" based on the relative slope or relative

curvature. Unfortunately, such a function cannot be used if the current method of solution is to be retained. However, a similar result might be expected if the first approximation $w = 1/y^m$ is used instead. The function $w = 1/y^2$ produces dramatically improved solutions for both $n = 1$ and $n = 2$. The relative-slope case ($n = 1$) is definitely superior, but both are excellent. The required matrix inversion becomes progressively less stable as m increases, and no compensating improvement in the solution results from using $m = 3$. A value of $m = 1$ has not been tried. The best solution to date still has systematically large errors near the bases of the large peaks, where x is oversmoothed at the expense of other regions. In an attempt to remove this fault, a second approximation to $w = 1/x^m$ has been worked out but not yet incorporated into the program.

- 4) Particularly for $m \leq 2$, a good starting guess for the Lagrange multiplier (usually called the "smoothing parameter") can be calculated directly from the data. A relatively sophisticated extrapolation scheme has been worked out which will find the critically-smoothed solution within 1% in about three iterations. At present the solution

(on the Univac 1108) requires approximately $30 \times N^3$ μ sec per iteration for a set of N data points.

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SHORT RANGE CORRELATION EFFECTS ON THE DAMPING OF SINGLE-PARTICLE WAVE-FUNCTIONS IN THE NUCLEAR INTERIOR*

R. E. Schenter

Single-particle wavefunctions which describe the elastic scattering of low energy neutrons by target nuclei of ^{12}C , ^{16}O , and ^{40}Ca have been calculated. Use has been made of nonlocal optical model potentials formally derived within the framework of the "no-polarization" approximation in terms of the nucleon-nucleon interaction and the ground-state wavefunction of the target. Previous calculations⁽¹⁾ have shown that the nonlocality of the "no-polarization" optical potential, which is a consequence of the Pauli exclusion principle, gives rise to a reduction of the single-particle wavefunction in the nuclear "interior" in comparison with that given by an "equivalent" local potential. Several forms of short-range correlation in the target wavefunction have been tried in an investigation of their effect on this

* Abstract submitted for the August 31, 1967 APS meeting in Seattle.

reduction. It has been found that very large reduction occurs for cases in which hard-core correlations are used with less extreme damping for the softer Amado and intermediate forms.

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NONLOCAL OPTICAL MODEL ANALYSIS OF NEUTRON TOTAL CROSS SECTIONS

D. W. Glasgow

Introduction

Historically, the local optical model of nuclear reactions replaced the intricate details of the multiple body interactions with an effective interaction between the incident particle and the nucleus as a whole. As time progressed the local-complex potential, which was used to describe the effective interaction, grew in complexity. A square well potential was replaced by a diffuse edged potential and a spin orbit interaction. It was eventually shown from first principles that the true optical potential was nonlocal due to exchange and target polarization effects. This nature produces energy dependent parameters in a local optical model and energy independent parameters in a nonlocal model. The important consequence of this fact is that once the appropriate non-local potential is found, it applies not only to all nuclei but also to

all energies within the range of validity of the model. The present program utilizes the high quality Battelle-Northwest total cross section data as the basis for study and improvement of the nonlocal optical model.

Status of Program

A computer program for calculating the effects of the interaction of neutrons with a nonlocal optical potential has been placed into operation on the IBM-7090.

The reaction, elastic scattering, and total cross sections for neutrons incident upon 32 nuclides have been calculated for neutron energies of 3, 5, 7, 9, 11, 13, and 15 MeV. The nuclides were [^1H , ^2H , ^6Li], [^{19}F , ^{27}Al , ^{45}Sc , ^{51}V , ^{55}Mn , ^{59}Co , ^{75}As , ^{89}Y , ^{93}Nb , ^{99}Tc , ^{103}Rh], [^{115}In , ^{127}I , ^{133}Cs , ^{139}La , ^{141}Pr], [^{159}Tb , ^{165}Ho , ^{169}Tm , ^{175}Lu , ^{181}Ta], [^{197}Au , ^{206}Pb , ^{209}Bi], [^{232}Th , ^{233}U , ^{235}U , ^{238}U , and ^{239}Pu]. The theoretical neutron total cross sections were compared with the neutron cross sections measured at Battelle-Northwest.

The experimental results for ^1H , and the odd-odd nuclei ^2H , and ^6Li disagree with theory by 10 to 25% over the energy range 3 to 15 MeV. This disagreement is not surprising since light nuclei with a few nucleons fail to satisfy the requirements of the optical model, i.e., inadequately replacing the intricate details of the multiple body interactions with an effective interaction between the incident neutron and the nucleus as a whole.

The agreement between experiment and theory is quite impressive for the even-odd spherical nuclei-- ^{19}F , ^{27}Al , ^{45}Sc , ^{51}V ($N=28$), ^{55}Mn , ^{59}Co , ^{75}As , ^{89}Y ($N=50$), ^{93}Nb , ^{99}Tc , and ^{103}Rh . The theoretical and experimental results agree to within 1 to 3% over the entire energy range 3 to 15 MeV.

The next five even-odd nuclei-- ^{115}In , ^{127}I , ^{133}Cs , ^{139}La ($N=82$), and ^{141}Pr --lie in a transition region between spherical and nonspherical nuclei. With the exception of ^{115}In , which agrees with theory to within 1 to 2%, the disagreement increases from 3 to 5%; however, the agreement is still not too bad. The next five even-odd nuclei ^{159}Tb , ^{165}Ho , ^{169}Tm , ^{175}Lu , and ^{181}Ta have large nuclear quadrupole moments, are highly deformed, and are nonspherical. The disagreement between experiment and theory is as much as 15% over the

energy range 7 to 13 MeV. The next three nuclei, even-odd ^{197}Au and ^{209}Bi ($N=126$), and even-even ^{206}Pb lie in a transition region between nonspherical and spherical nuclei. The disagreement between experimental and theoretical results over the energy range 7 to 13 MeV is now improved to 4 to 7%. However, there is a trend toward more and more disagreement at lower energies. This amounts to as much as 8% over the energy range 3 to 6 MeV. The last five nuclei, even-even ^{232}Th and ^{238}U , and odd-even ^{233}U , ^{235}U , and ^{239}Pu are all heavily deformed and nonspherical. Again the disagreement over the energy range 9 to 15 MeV and 3 to 6 MeV is typically 11%.

This systematic survey shows in a clear manner that the phenomenological nonlocal optical model, in its present form is not any adequate description of nonspherical nuclei.

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