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**REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER, 1966**

APRIL 15, 1967

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<i>J. M. Harris</i>	<i>33881</i>	<i>328</i>	<i>NOV 8 1968</i>

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UNITED STATES ATOMIC ENERGY COMMISSION UNDER CONTRACT AT(45-1)-1830

PRINTED BY/FOR THE U. S. ATOMIC ENERGY COMMISSION

3 3679 00060 5594

BNWL-400

UC-80, Reactor Technology

REACTOR PHYSICS DEPARTMENT
TECHNICAL ACTIVITIES QUARTERLY REPORT
OCTOBER, NOVEMBER, DECEMBER, 1966

By

The Reactor Physics Department Staff

F. G. Dawson, Manager

April 15, 1967

FIRST UNRESTRICTED
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PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON

Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U.S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.65

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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. 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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* Abstract submitted for the January 30, 1967 APS meeting in New York.

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* Work performed under U. S. Atomic Energy Commission Contract AT(45-1)-1830.

** Summary of a formal report to be published.

REACTOR PHYSICS DEPARTMENT
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SUMMARY

REACTOR THEORY AND CODE DEVELOPMENT

Some integrals arising in the thermalization theory have been formulated.

Single particle wave functions which describe the elastic scattering of low energy neutrons by target nuclei of helium and oxygen have been calculated.

Program BROAD has been written which calculates temperature dependent cross sections directly from experiment skipping the intermediate step of fitting data with multilevel resonance formulas.

AI's computer code UNICORN has been modified to allow the use of different energy spacing.

A new, updated HRG data tape has been produced.

A new TEMPEST library updating program has been written for the 1107 and 7090 computers.

A family of improvements have been made on the burnup chains ZODIAC2 and ZODIAC3. A fast reactor ZODIAC version, ZODIACF is now operational.

A computer code, OP, has been written for shielding optimization studies.

An approximate model which uses the ideal gas model for energy transfer and a fictitious heavy mass isotope to provide a correct transport cross section is used to improve the RBU thermalization routine.

A computer code, CONVRT, has been developed to convert foil activation data from the paper tape output of an automatic counting system to the format required by a data analysis code.

THERMAL REACTORS

Reactivity coefficients have been calculated for a plutonium fueled light water reactor in which the fuel contained 62/22/12/4 atom percent $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu}/$ and ^{242}Pu , respectively.

Preparations have been completed which will allow effective cross section ratios to be obtained from a least squares analysis of burnup data from the Plutonium Recycle Test Reactor and the Yankee Reactor

The present design of a gamma scanning facility which is used to acquire burnup information from irradiated fuel rods is described.

A reactivity measurement technique has proven to be a very sensitive and precise method for determining the effective ^{10}B concentration in samples of borated D_2O .

A series of critical experiments has been performed in the Plutonium Recycle Facility using 1/2 in. diam UO_2 -2 wt% PuO_2 fuel rods moderated with H_2O .

The results of an extensive set of reactivity measurements of neutron absorbing rods in $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ lattices is summarized.

An experiment was conducted in the PCTR to obtain an estimate of the worth of the HTLTR control rods. It is estimated from the results that they will be worth about 54 cents when loaded in the inner most positions provided.

FAST REACTORS

An evaluation of the physics characteristics of various proposed alternate FTR concepts is partially complete. The physics characteristics of these concepts are generally quite similar.

A comparison has been made of the power distribution, flux distributions, and Doppler coefficients for an unzoned and a two-zone ceramic fueled FTR. The two-zone core has a slightly more negative Doppler coefficient.

The effect of crystalline binding on the FTR Doppler coefficient has been investigated. Results with the Dolling distribution give a reduction of the "Free Gas" Doppler coefficient of about 20% at room temperature and 2% at a temperature of 1000 °K.

A preliminary study of FTR fuel heterogeneity effects has been made. The heterogeneity effects in the FTR are estimated to be $\sim 0.0002 \Delta \rho$.

Details of the Phase A FTR Critical Experiments have been formulated and the experiments were started December 1, 1966.

The need for detailed simulation of fuel depletion and fission product accumulation in the NPTF is being investigated.

Analysis of the basic radial FTR shield concept are being carried out using the MAC code.

Sodium activation studies for the present FTR reference concept are in process.

Survey calculations have been made to determine the shielding requirements for irradiated FTR fuel.

CRITICAL MASS PHYSICS

Exploratory pulse neutron measurements have been made on small plutonium systems to investigate the $k\beta/\lambda$ reactivity technique. The sensitivity of the method to experimental parameters and procedures was touched on. The method was found to be at least internally consistent.

A computational analysis was made for the large number of available critical experiments with hydrogenous mixtures. The calculations were made using both multigroup S_4 and diffusion theory with 18 energy groups obtained from the GAMTEC-II code. Resonance capture by the isotope ^{240}Pu was treated in the narrow resonance (NR) and narrow resonance-infinite mass (NRIA) approximations. A rather good agreement was found between experiment and theory. Values for critically safe masses and dimensions are given for $\text{Pu}(\text{NO}_3)_4$ solutions and ^{239}Pu -water mixtures.

NEUTRON PHYSICS

The atomic and molecular motions of water at 268 and 299 °K has been studied using the Battelle-Northwest rotating crystal spectrometer.

Slow neutron inelastic scattering cross sections have been compiled for H_2O at five degrees below its freezing point and for room temperature H_2O and D_2O .

REACTOR THEORY AND CODE DEVELOPMENT

SOME INTEGRALS ARISING IN THERMALIZATION THEORY

A. G. Gibbs

The differential cross section for scattering of slow neutrons by a system of atoms can be expressed in terms of time-dependent correlation functions which describe the motion of atoms in the system. In the well-known incoherent, Gaussian approximation the expression for the n^{th} Legendre moment of the scattering cross section is

$$\frac{\sigma_n(E_0 \rightarrow E)}{\sigma_b} = \frac{1}{2\pi KT} \sqrt{\frac{E}{E_0}} e^{-\beta/2} \int_0^\infty \cos \beta t \left(\int_{-1}^1 e^{-\alpha w(t)} P_n(\mu) d\mu \right) dt \quad (1)$$

where α and β are the non-dimensional energy and momentum transfers and μ is the cosine of the laboratory scattering angle. The function $w(t)$ is related to the time-dependent mean square displacement of an atom of the system, and is obtained from experiment or from model calculations.

Since α is a linear function of μ , the integration with respect to μ can be carried out analytically for all values of n . However, because of the complexity of the function $w(t)$, the integration with respect to t must usually be done numerically. Since $w(t)$ often has a simple asymptotic form for

large values of t , it appears that the speed and accuracy of the computation might be enhanced by subtracting from the integrand the contribution due to the asymptotic part of $w(t)$ and evaluating this contribution analytically. Numerical methods would still be required to evaluate the remaining contribution to $\sigma_n(E_0 \rightarrow E)$ but if relatively simple expressions are obtained for the large time contribution a significant reduction in computing time could be realized.

For many applications it is sufficient to take $w(t) \sim Jt^2 + K/4$ for large t , where J and K are constants. This corresponds to the free translation of a molecule of effective mass $1/J$ times the atomic mass. When $w(t)$ in Equation (1) is replaced by $Jt^2 + K/4$ and the integral over μ evaluated we find that for each n the integral over t can be expressed as a sum of integrals of the form

$$I_k = \int_0^\infty \frac{e^{-A(Jt^2 + K/4)} \cos \beta t}{(Jt^2 + K/4)^k} dt \quad (2)$$

where A , J , K , and β are independent of t . These integrals have been successfully evaluated analytically, and we find that

$$I_k = \frac{(-1)^{k-1}}{(k-1)!} \left\{ \frac{\partial^{k-1}}{\partial x^{k-1}} \left(\hat{I}_1(x) \right) \right\}_{x=K/4} \quad (3)$$

where

$$\hat{I}_1(x) = \frac{\pi}{4\sqrt{Jx}} e^{-A(K/4-x)} \left\{ e^{-\beta(x/J)^{1/2}} \operatorname{Erfc} \left[(Ax)^{1/2} - \frac{\beta}{4}(Ax)^{-1/2} \right] + e^{\beta(x/J)^{1/2}} \operatorname{Erfc} \left[(Ax)^{1/2} + \frac{\beta}{4}(Ax)^{-1/2} \right] \right\} \quad (4)$$

Since highly efficient routines are available for evaluating error functions and exponentials, the computation of the I_k from the above expressions should result in a reduction of the computation time needed to evaluate the $\sigma_n(E_0 \rightarrow E)$. Coding is now being developed to accomplish this, and the results will be reported later.

DAMPING OF SINGLE-PARTICLE WAVEFUNCTIONS IN THE NUCLEAR INTERIOR*

R. E. Schenter

Single-particle wavefunctions which describe the elastic scattering of low energy neutrons by target nuclei of helium and oxygen 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. Emphasis has been placed on properly including all exchange terms in the

optical potential. It has been found that the use of these nonlocal potentials lead to single-particle wavefunctions for the alpha-particle greatly reduced from that given by an "equivalent" local potential ($\Psi_{NL} < 0.1 \Psi_L$) in the nuclear "interior" whereas essentially no reduction occurs for the oxygen case. The stability of the damping effect for the alpha-particle has been extensively tested for variations in energy and target wavefunction form. Calculations have been made with several types of two body correlations in the target wavefunction including short-range correlations reflecting the hard-core properties in the nucleon-nucleon interaction.

* Abstract submitted for the January 30, 1967 APS meeting in New York

DIRECT DOPPLER BROADENING OF EXPERIMENTAL DATA

R. E. Schenter

Program BROAD has been written to calculate temperature dependent cross sections values directly from experiment, skipping the tedious intermediate step of fitting the data with multilevel resonance formulas. Use has been made of the following formula which relates the effective cross section at a temperature T_2 to a lower temperature T_1 :

$$v\sigma_{T_2}(v) = 4\pi \int_0^\infty \left[v' \sigma_{T_1}(v') \right]$$

$$\left[\frac{m_A}{2\pi k(T_2 - T_1)} \right] \left[\exp \left(-\frac{m_A}{2k(T_2 - T_1)} \right) \right]$$

$$\left[v^2 + v'^2 \frac{\sinh \left(\frac{m_A v v'}{k(T_2 - T_1)} \right)}{\frac{m_A v v'}{k(T_2 - T_1)}} \right] v'^2 dv'$$

where v is the velocity of the incident neutron. Typical results showing the broadening of the fission cross section of ^{239}Pu in the unresolved resonance region are given in Figure 1.

FIGURE 1. Typical Temperature Dependence of the Fission Cross Section of ^{239}Pu

MODIFICATIONS TO UNICORN

K. B. Stewart

The Atomics International computer code UNICORN was modified to allow a different energy spacing. AI's neutron cross section Doppler broadening program DOPCRS (AI-CE-Memo-21) was

included in the UNICORN code to allow the calculation of temperature dependent cross sections. The time requirements for UNICORN doubled as a result of the increased computation requirements of the Doppler broadening.

HRG DATA TAPE

J. L. Carter

A new HRG data tape has been produced which departs in several respects from tapes previously used. The new tape is the first updating of the HRG library since the addition of large amounts of new data to the BNW Master Library in the past few months. Data for 80 nuclides have been revised or added. Multiple versions of ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu are present, to provide data compatible with several recommended self-consistent sets of thermal parameters for these nuclides. These sets are designated as the Leonard,^(1,2) Westcott,⁽³⁾ Sher-62⁽⁴⁾ and Sher-65⁽⁵⁾ sets. An extensive rearrangement of nuclide identification numbers has been made, placing the most frequently used nuclides near the front of the tape. Materials--currently H_2O , D_2O , and 304SS - appear as nuclides for the first time; these are simply convenient combinations of nuclides which also appear separately elsewhere on the tape. Also added is a synthetic

nuclide, Unit 1/v, with a 2200 m/sec absorption cross section of 1 barn and no scattering, for possible use in simulating data for nuclides not present. Data are presented for a total of 237 nuclides.

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1. B. R. Leonard, Jr. "Survey of the Status of Low Energy Cross Sections of Fissile Nuclides," *Neutron Physics, Proceedings of Symposium held at Rensselaer Polytechnic Institute, 1961*, edited by M. L. Yater, Academic Press, New York, 1962, pp. 3-33. Also HW-69342. April, 1961.
2. B. R. Leonard, Jr. "Plutonium Physics: Contribution to Plutonium Handbook," HW-72947. March, 1962.
3. C. L. Westcott et al. "A Survey of Values of the 2200 m/sec Constants for Four Fissile Nuclides," *Atomic Energy Review* 3, 2, 3-60. July, 1965.
4. R. Sher and J. Felderbaum. "Least Squares Analysis of the 2200 m/sec Parameters of U-233, U-235, and Pu-239," BNL-722. June, 1962.
5. R. Sher and J. Felderbaum. "Least Squares Analysis of the 2200 m/sec Parameters of U-233, U-235, and Pu-239, Final Report," BNL-918. March, 1965.

TEMPEST LIBRARY UPDATER

E. L. Kelley

A new TEMPEST Library updating program has been written for the 1107 and 7090 computers that allows binary libraries to be made from decimal cards or from existing Binary Library tapes. This new updating routine writes binary cross section data to a

tape in full record form for each isotope, rather than the old system of each record consisting of only one Binary card image.

Revisions have been made to TEMPEST that will enable it to read and properly store this updated Library. Decimal libraries may still be used as in the past if so desired.

ZODIAC2

D. D. Matsumoto and R. H. Holeman

A family of improvements have been made to the 7090 burnup chain, ZODIAC2. HRG, a superior epithermal cross section averaging code, was made available as an alternative to GAM. In order to more effectively utilize HRG, reorganized, expanded libraries were prepared for HRG; for TEMPEST, the thermal cross section averaging code; and for SIGMA-3-H, the clerical code which provides cross section input for the diffusion code HFN.

Because of the reorganization and expansion of their libraries, a number of changes were made to TEMPEST and SIGMA-3-H, as well as to other

clerical sections of the chain. An improved library-accessing scheme was incorporated into the improved TEMPEST, and a number of rough edges polished in both TEMPEST and SIGMA-3-H.

Both the improved TEMPEST and the improved SIGMA-3-H were installed as optional alternatives to their predecessors, in order to maintain compatibility between ZODIAC2 and old libraries still in use, although the older versions will eventually be deleted as their related libraries become obsolete.

Testing of the augmented chain, which now includes 19 links, is still in progress.

ZODIAC3

D. D. Matsumoto

Work continues on the 1107 burnup code ZODIAC3. A versatile input routine written specifically for Z3 is working satisfactorily; some diversification for even broader utility is still planned. The generalized error routines are complete. Other service

routines are partially complete and in a state of constant change. The partially converted isotope transmutation section is receiving the bulk of current attention. Work has not begun on the cross-section averaging portion of the code.

The diffusion code HFN is now satisfactorily integrated into the ZODIAC3

system. Identical in operation to the standard HFN, the ZODIAC3 version of HFN is significantly larger, utilizing the larger core of the 1107 for additional in-core library space. The previous limits of 20 groups and 50 regions, which have proved adequate, are unchanged, but the maximum number of materials was increased from 100 to 200. The increase is most significant (and in greatest demand) for a larger

number of groups. For example, where $N_1 = N_2$ (full transfer matrix; no up-scattering):

No. of Groups	Maximum No. of Materials	
	Old HFN	HFN/Z30021
4	92	200
6	55	200
10	28	112
14	17	68
20	10	40

ZODIACF

D. D. Matsumoto and D. R. Marr

A test case has been processed on the fast reactor burnup code ZODIACF with apparent success. The code is a chain consisting of a straight LIFT (mechanical Fortran II to 1107

Fortran IV conversion) of CLERK and ALCHEMY, preceded by HFNW, and serviced by new simplified FIREUP and REFIRE routines. Input is compatible with output from FCC which generates fast reactor multigroup constants.

OPTIMUM SHIELDING PROGRAM, OP

H. S. Zwibel

The optimum shielding work reported in BNWL-315 has been extended to include multi-energy detail. A computer code, OP, has been written that is the multi-energy extension of SEARCH. The results reported in BNWL-315 have been reproduced with OP, however, the multi-

energy loops of OP have not yet been completely debugged.

A more efficient method of seeking the optimum distribution of absorber has been incorporated in OP. Initial results indicate that this reduces running time by a factor of 200 over SEARCH.

CELL CALCULATIONS USING THE RBU MONTE CARLO CODE

D. H. Thomsen

Results of thermalization calculations using the RBU Monte Carlo code

were given in the last quarterly report.⁽¹⁾ These results were based upon the use of the ideal gas model for H₂O scattering. The ideal gas model often proves inadequate for H₂O so a better

scattering kernel was needed for use in RBU. In this report an approximate model is used to improve the RBU thermalization routine.

This approximate model uses the ideal gas model for the energy transfer and a fictitious heavy mass isotope to provide a correct transport cross section. The fictitious transport cross section, σ_{TR}^f is calculated as

$$\sigma_{TR}^f(E, T) = \sigma_{TR}^{H_2O}(E, T) - \sigma_{TR}^{Ideal Gas}(E, T). \quad (1)$$

To calculate the transport cross section for light water it is first assumed that the scattering from oxygen is treated exactly by the ideal gas model. The average cosine of the scattering angle ($\bar{\mu}$) for hydrogen in light water was calculated using the Nelkin Model. The angular distribution of the scattering for the fictitious heavy isotope is treated as isotropic in the laboratory system by the RBU code. The scattering cross section

for the fictitious isotope can now be calculated as

$$\begin{aligned} \sigma_S^f(E) &= \sigma_{TR}^f(E) \\ &= \sigma_S^{Hydrogen in H_2O}(1-\bar{\mu}) \\ &\quad - \sigma_{TR}^{Mass 1 gas} \end{aligned} \quad (2)$$

A cell calculation was made using this approximate model for H_2O in the RBU code. The cell was the same considered in the last quarterly report,⁽¹⁾ which is a boron-aluminum rod in H_2O . The results of this thermalization calculation are compared in Table I to those obtained using PROGRAM S-XIII code, the THERMOS code, and the RBU code with the ideal gas model.

The comparison of RBU calculations using the ideal gas and the approximate H_2O model shows large differences in average cross sections and fluxes. The results obtained using the approximate model agree fairly well with those obtained using PROGRAM S-XIII and THERMOS. If we assume that the results

TABLE I. Comparison of RBU Monte Carlo Calculations, Using Approximate H_2O Model, with Results of Other Calculations and Models

Case	$-\phi_W \bar{\phi}_B$	$\bar{\sigma}_S^H$ (barns)	$\bar{\sigma}_a^H$ (barns)	$\bar{\sigma}_a^B$ (barns)
1. RBU Monte Carlo (Approximate H_2O Model)	2.955 $\pm$ 0.060	27.52	0.2646 $\pm$ 0.0018	2549 $\pm$ 25
2. Program S-XIII (Nelkin)*	2.952	27.47	0.2637	2543
3. THERMOS (Nelkin)**	3.032	27.45	0.2645	2551
4. RBU Monte Carlo (Ideal Gas)	2.684 $\pm$ 0.034	27.43	0.2625 $\pm$ 0.0009	2613 $\pm$ 13

* Both P_0 AND P_1 scattering kernels were used

** Transport corrected kernel

obtained from the PROGRAM S-XIII calculation are correct then we can conclude that this approximate model represents an improvement in the RBU thermalization routine for light water systems.

This approximate model can be extended to other liquid moderators by simply computing a new fictitious scatterer from Equation (1). The fictitious scattering cross section for heavy water was calculated in preparation for the study of 19 rod cluster fueled-D₂O moderated and cooled cells.

A natural uranium fueled 19 rod cluster cell was analyzed using the RBU Monte Carlo code. Lattice parameters for this cell have been measured in the PCTR. Preliminary results show the calculated thermal utilization factor is ~1% lower than the experimental value and the calculated infinite multiplication constant was ~4%

lower than the experimental value. Two errors in the formulation of the RBU resonance routine were uncovered. One was the computation of the cross section below the last resonance for each isotope and the other was the contribution of $\lambda > 0$ resonance to the cross section. These errors are being corrected and the problem will be rerun.

Acknowledgement

The author thanks Dr. C. W. Lindenmeir for suggesting the use of the approximate thermalization model.

References:

1. D. H. Thomsen. "Thermalization Calculational Study-Comparison of Results Obtained Using the RBU Monte Carlo, THERMOS, and PROGRAM S-XIII Codes," Quarterly Technical Activities Report, July, August, September, 1966, BNWL- 340.

COMPUTER CODING FOR AN AUTOMATIC FOIL COUNTING SYSTEM

G. E. Hanson

Introduction and Summary

A computer code, CONVRT, has been developed to convert foil activation data from the paper tape output of an automatic counting system to the format required by a data analysis code. CONVRT, although written for a special paper tape output format and a special analytical code input, could be used with certain modifications for any foil counting system with a paper tape output.

Use of CONVRT

The data analysis code for which CONVRT was written was APDAC-III.⁽¹⁾

The APDAC code is a rather complex code which, in its simplest form, background and decay-corrects foil activities and gives specific activities relative to some designated normalization foil. The development of CONVRT was necessitated by the fact that APDAC will not accept the data in the form in which it appears on the paper tape. Further, by having the ability to record all of the counter information (gain, bias voltage, etc.,) and all foil numbers on the paper tape by means of manually operated switches, there should be a reduction in the amount of time required to obtain output from APDAC.

The function of CONVRT is to change the data from paper tape characters to BCD

(binary coded decimal) characters which are recognized by a computer, to manipulate this data into the correct format, and to make some conversions of units so that the information will be in a form which can be read by APDAC. By the use of the appropriate control cards, both CONVRT and APDAC can be executed in sequence, resulting in a decrease in the time required for data reduction. This is due to the fact that previously counting data was sent to the computer center on special sheets of paper, then was keypunched and sent back to the experimenter, and was finally resubmitted by the experimenter to be run on APDAC.

The execution of CONVRT and the various options available is controlled by approximately 25 control cards. These are to be sent to the computer center along with the paper tape to be processed. A flow chart of the entire operation is shown below:

A few comments about the chart are in order. The Tally Reader scans the paper tape and records the paper tape characters (standard Friden 8-channel code) on a low density magnetic tape (tape A). Tape A is then read by PTCNV,* a subroutine which is called by CONVRT. The output of PTCNV is a high density magnetic tape (tape C) with BCD characters which are suitable for processing with FORTRAN coded programs on the Univac 1107.

Tape C is then read by the main program, CONVRT, and the data is manipulated to conform to the APDAC format. The data is finally punched on cards

* PTCNV is the controlling subroutine of a group of machine language subroutines which were furnished to the author by Computer Sciences Corporation for use in the CONVRT program. They control the actual character conversions.

Operational Setup

and printed in the computer output. The data is put on punched cards to allow the experimenter to make small changes and additions to the foil data for subsequent analysis. An option is provided to write the converted data on magnetic tape (tape B). Tape B can be immediately read and analyzed by APDAC.

A more detailed document is being prepared on the use of CONVRT. The

document will include a listing of the program and instructions on the preparation of the control cards and the paper tape.

References

1. J. R. Lilley. APDAC-I, A PCTR Data Analysis Code for the IBM 709, HW-63592. September 7, 1960. *APDAC-III is an up-to-date version of this code written for the UNIVAC 1107.*

THERMAL REACTORS

PuO₂-UO₂-H₂O SYSTEMS-SURVEY CALCULATIONS

U. P. Jenquin

A previous report⁽¹⁾ gave calculated results of the reactivity coefficients for a variety of plutonium fueled light water reactor systems. As an extension to this study, calculations have been performed for these reactor systems which contain a fuel differing in plutonium composition from that of the previous study. The previous study was based upon a fuel which contained 92, and 8 atom percent of ²³⁹Pu and ²⁴⁰Pu, respectively. The fuel for this study is assumed to be 62/22/12/4 atom percent ²³⁹Pu/²⁴⁰Pu/²⁴¹Pu/ and ²⁴²Pu, respectively. The fissile atom content was assumed to be equivalent to the 92/8 fuel of the previous study. The effects of fuel rod size on the physics characteristics were studied.

The results of the calculations for the 62/22/12/4 fuel are compared to the values for the 92/8/0/0 fuel in Figures 1 through 3.

The reactivity coefficients are not drastically different for the two fuels.

The reactivity coefficients variation with fuel rod size is shown in Table I. The coefficients are all smaller for the larger size fuel rod.

A report will be written summarizing the results and giving details of these calculations.

References

1. R. C. Liikala, and U. P. Jenquin. Physics Survey Calculations for a PuO₂-UO₂ Fuel in Idealized H₂O Power Reactors, BNWL-305.

FIGURE 1. Moderator Temperature Coefficient of Reactivity

TABLE I. Comparison of Reactivity Coefficients for Two Rod Sizes at the Same Moderator to Fuel Volume Ratio

	Fuel Diameter, in.	
	<u>3/8</u>	<u>1/2</u>
Average Moderator Coefficient 20 to 255 °C, $1/k_{\infty} \text{ d } k_{\infty}/dT(10^{-4}/^{\circ}\text{C})$	-0.74	-0.40
Average Doppler Coefficient 20 to 1020 °C, $1/k_{\infty} \text{ d } k_{\infty}/dT(10^{-5}/^{\circ}\text{C})$	-2.20	-1.89
Average Void Coefficient 0 to 30% Void, $1/k_{\infty} \text{ d } k_{\infty}/\% \text{ Void}(10^{-3}/\% \text{ Void})$	-1.45	-1.28

FIGURE 2. Fuel Temperature Coefficient of Reactivity

FIGURE 3. Void Coefficient of Reactivity

ANALYSIS OF BURNUP DATA

R. P. Matsen

Preparations have been completed to perform a least squares analysis of burnup data obtained from the Plutonium Recycle Test Reactor (plutonium-aluminum, natural UO_2 and $\text{UO}_2\text{-PuO}_2$ fuels) and YANKEE Reactor (enriched

UO_2). Although the burnup data are obtained from different positions in the reactor and are taken at different times, the fuel composition at any given exposure is unique under the condition that the experimental samples have been irradiated in identical environments (i.e. same moderator to

fuel-ratio, absence of nearby control shims, etc.). The experimental data listed above have been selected to meet these criteria. Therefore, the only variables that govern transmutation changes are the fuel and fission product compositions in the immediate vicinity of the experimental sample itself.

The nonlinear least-square fitting program, LIKELY⁽¹⁾, is used to fit the burnup data. Several changes that improve running time and information retrieval for the burnup problem have been incorporated into both a single precision version (SINLIK) and a double precision version (DUBLIK) of LIKELY, but the basic features of the program remain unaltered. The integrated form of the burnup equations have been derived and converted into the Fortran language required by the least squares fitting routine. These equations express the relation between the uranium and/or plutonium isotopic concentrations (the raw burnup data from each sample has been cast in this form) and the ratios of flux averaged

cross sections (the parameters to be determined by the fitting analysis). Only one method has been devised for incorporating some structure into the cross section ratios which still allows the burnup equations to be integrated in closed form. An example is $\frac{\Lambda_a^{40}}{\Lambda_a^{25}} = A + B/N^{40}$ where A and B are parameters to be determined by the fitting analysis, N^{40} is the ^{240}Pu concentration, and Λ_a^{40} and Λ_a^{25} are the flux averaged cross sections of ^{240}Pu and ^{235}U respectively. The empirical self-shielding term (B/N^{40}) is limited to data which possesses a substantial amount of ^{240}Pu . Preliminary least squares fitting attempts indicate that this method of describing the cross section ratios provides a marked improvement in the description of the experimental data.

Reference

1. B. H. Duane. "Machine-Analysis Abstract--Maximum Likelihood Non-linear Vector Fields--Battelle Northwest Program Likely", *Physics Research Quarterly Report*, January, February, March, 1966, BNWL-284.

GAMMA SCAN FACILITY TO ACQUIRE BURNUP DATA BY A NONDESTRUCTIVE METHOD⁽¹⁾

D. E. Christensen

In order to acquire burnup information from irradiated fuel rods by a non-destructive method, a gamma scanning facility has been built. This work has proceeded to the point that the present design, shown in Figure 1, permits scanning of irradiated 1/2 in. diameter fuel rods under approximately 6 ft of water in the PRTR basin. The

collimator is 105 in. long from source to detector and contains 6 in. of lead near the source and 6 in. of lead next to the detector to collimate the gamma ray beam. At present a 1/4 in. diameter aperture is incorporated in the design and work is under way to permit the variation of the aperture size and shape. The detector, a 1 3/4 in. diameter by 2 in. NaI (Tl) crystal, is shielded from background irradiation by lead housing 4 1/2 in. thick. The

detector and housing are mounted on rails which makes it possible to move the collimator along a fuel rod. The rod is held horizontal in a "V" shaped trough and is raised to be positioned directly beneath the collimator head. A shutter is included near the source to close the aperture and take background readings.

The detecting system, in addition to the NaI (Tl) crystal optically coupled to an EMI 9536 B photo-multiplier contained in a light tight assembly, includes a preamplifier (white cathode follower type), a double delay line clipped nonoverloading linear pulse amplifier, a single channel pulse height analyzer which can be made to scan a given energy range, a linear count rate meter, a strip chart recorder, and a scaler. A multichannel analyzer is available to do more detailed energy analysis.

The initial purpose of the system was to obtain the longitudinal variation of the activity of fission products in a fuel rod in order to assess the relative exposure of the fuel as a function of position. Subsequent destructive analysis of 1/2 in. fuel sections would then provide the absolute measurement of the exposure. Data taken to date has been to test

the stability of the detecting system and the reproducibility of the count rate as a function of position. These data have been shown to be reproducible to $\pm 1.5\%$.

Reference

1. J. L. Stringer, "Gamma Scanner Facility," Quarterly Progress Report Research and Development Program, April, May, June, 1964, HW-83298, 7.1-7.5 (Nov. 1964).

FIGURE 1. Gamma Scanning Facility

MEASUREMENTS OF ^{10}B CONCENTRATIONS USING REACTIVITY TECHNIQUES

J. H. Lauby and J. W. Kutcher

Experiments with UO_2 - 2 wt% PuO_2 fuel in the Plutonium Recycle Critical Facility (PRCF) and in the Plutonium Recycle Test Reactor (PRTR) have used

^{10}B in the moderator for reactivity adjustment. The concentrations have varied from 0 to 25 WPPM while the corresponding reactivity worths varied from ~ 30 to 10 milli-k per WPPM. A precise knowledge of the boron concentration during a given set of

experiments is essential to the analysis of the experiments.

Preliminary measurements showed the poison sensitivity of the Thermal Test Reactor (TTR) to be high enough to give the desired accuracy with 400 ml samples. Cylindrical bottles of small diameter were selected to minimize geometric effects on the measurements. These were thin walled and made of ~ 49 g of polyethylene. Empty sample bottles were weighed and a set of six having nearly equal weights were selected to hold standard solutions. These bottles and one of the heaviest bottles were tested empty in the TTR for impurities in the polyethylene. No significant variations were detected. The empty bottle worth was determined to be - 0.128 ϵ /g of polyethylene. Two bottles most widely differing in weight were filled with 400 g of pure D_2O to determine the worth of polyethylene when filled with D_2O . This worth was found to be - 1.86 ϵ /g of polyethylene. Bottle weight differences between samples were corrected using this value.

Standard solutions of 0.995, 9.98, 15.0, 19.9, and 24.9 WPPM ^{10}B in D_2O were prepared. Reactor periods were run with alternately pure and borated samples in the TTR. Differences from the base period reactivities with pure D_2O were corrected for bottle weights

and reactor drift. A curve of ^{10}B worth versus standard concentrations was obtained. The values measured for 0.995 and 24.9 WPPM did not fit the straight line fit to the other points. These two samples were re-measured. The second values reproduced the first two within 0.03 ϵ or 0.057 WPPM. It was concluded that the worth of the 24.9 WPPM standard was low due to self-shielding. The conclusion for the 0.995 WPPM standard was that the solution was in reality ~ 0.70 WPPM.

Corrected reactivity differences between a pure D_2O base sample and PRCF and PRTR unknown samples are shown in Figure 1 on the worth curve derived from the standard solutions. A sample of 5% degraded D_2O from the PRTR coolant was also compared to pure D_2O and found to be equivalent to 3.75 WPPM ^{10}B .

The reactivity measurement technique has proven to be a very sensitive and precise method to determine the effective ^{10}B concentrations in samples of borated D_2O . The effective ^{10}B concentration of an unknown sample can be determined to ± 0.03 WPPM, which should be sufficiently precise to conduct the analysis of the present experiments.

FIGURE 1. ^{10}B - D_2O Solution Worth
400 g Samples in TTR

BATCH CORE CRITICAL TESTS IN THE PRTR

J. W. Kutcher and J. H. Lauby

Critical experiments have been performed in the Plutonium Recycle Test Reactor (PRTR) with the Batch Core loading. The fuel consists of Mark I-R 19-rod clusters of UO_2 -2 wt% PuO_2 , with a fuel length of 58 1/2 in. A maximum of fifty-five of these fuel elements have been loaded in the PRTR and sufficient boron has been added to the D_2O moderator to reduce the effective neutron multiplication to unity with the moderator level near ninety-five in.

Measurements have been made of critical loadings to determine fuel element worths and boron worth. A number of measurements of the reactivity change due to loss of coolant have been made. Flux traverses have been made using gold foils, and cell traverses have been made using lutetium foils.

A computer program has been written to calculate reactor periods, with their uncertainties, from experimental doubling time data. This program has been used in conjunction with the Period-Reactivity program to calculate

reactivities for the critical tests which have been conducted. The results have been used in the determinations of fuel element worths, boron worths, and the moderator level coefficients.

The moderator level coefficient of reactivity has been measured as a function of the moderator level, for various concentrations of boron in the moderator. The results of these measurements are shown in Figure 1. The moderator level coefficient of

reactivity is essentially independent of boron concentration for moderator levels more than 1 in. below the top of the fuel elements. Above the top of the fuel, the coefficient is inversely dependent on boron concentrations, as shown in the figure for concentrations of zero and 21 wppm Boron-10.

Axial flux traverses, using gold foils, have been made for six different core loadings. One additional

FIGURE 1. PRTR Moderator Level Coefficient of Reactivity

traverse will be made very shortly. The results of these measurements are shown in Figure 2. A cell traverse, using lutetium foils, has been performed in the 55-element core with 21 wppm Boron-10 in the moderator. The flux ratios among the three rings in the fuel element have been compared with previous THERMOS calculations; these results are tabulated below.

	Measured ϕ/ϕ_{av}	Calculated ϕ/ϕ_{av}	Calculated Power(%)
Center Rod	0.62	0.55	3.3
6-Rod Ring	0.75	0.69	23.6
12-Rod Ring	1.16	1.19	73.1

The coolant worth has been measured in three types of experiments. One set of measurements has been made with all 84 coolant channels dry and either 31 or 55 channels fueled. The results of these measurements are tabulated below.

Fuel Elements	B^{10} Concentration (wppm)	Worth of Coolant Removal (mk)
31	11	-19.6
55	15	- 9.2

These coolant removal worths are for the D_2O coolant used in normal PRTR operation ($> 97\% D_2O$).

FIGURE 2. PRTR Axial Flux Traverses with Bare Au-Normalized by Peak to Average Foil Activity Over the Core Length

In the second case the 55 channels containing fuel elements were cooled with either D_2O or H_2O while the 29 unfueled peripheral channels were filled with primary grade D_2O coolant. For 55 fuel elements, and 21 wppm Boron-10 in the moderator, the following coolant removal worths were measured:

Worth of Removal of Coolant
from Fueled Channels Only

Removal of D_2O coolant: -0.8 milli-k
Removal of H_2O coolant: +5.6 milli-k

A third set of coolant removal worth measurements has been made with all coolant channels but the central one (# 1550) in their normal D_2O -filled condition. The central channel has been dry, filled with H_2O , and filled with D_2O ; and the reactivity change determined for each configuration. In this case all of the coolant removal worths were negative. The results are tabulated below.

Fuel Elements	B^{10} Concentration (wppm)	Coolant Removal Worth (mk)	
		H_2O	D_2O
9	0.0	-6.2	-1.5
12	2.6	-5.2	-1.3
16	5.6	-4.4	-1.05
23	10.2	-3.1	-0.47
31	14.1	-2.1	-0.02
55	20.9	-1.8	-0.06

The decrease in coolant worth is due mainly to the decreased worth of the central channel as more fuel elements are added to the core, rather than to the change in boron concentration in the moderator.

Several experiments have been performed in the PRTR with the Fuel Element Rupture Test Facility (FERTF) located in the central channel (# 1550). This channel was surrounded by fifty-four standard fuel elements in D_2O coolant, with 21 wppm Boron-10 in the moderator. The FERTF element (Figure 3) consisted of six UO_2 -2 wt% PuO_2 rods in a stainless steel basket tube, and was cooled with H_2O .

With no fuel element in channel 1550, the effect of the loss of H_2O coolant was measured to be +10.4 milli-k. With H_2O in the channel, the worth of inserting the FERTF element was measured to be +8.2 milli-k. With the fuel element present, the effect of the loss of H_2O coolant was measured to be +2.0 milli-k.

A flux traverse of the FERTF cell was made with gold pins, using a separable fuel rod. Analysis of these data is in progress.

Power generation rates in fuel rods in the FERTF element and in the first ring of elements in the Batch Core have been measured by gamma scanning of

FIGURE 3. Cross Section of FERTF Basket Element

these rods after a low power PRTR irradiation. The ratio of fissions in a UO_2 -2 wt% PuO_2 rod in the FERTF element to a maximum rod in a 19-rod cluster in the first ring of elements in the Batch Core was measured to be 1.53. The ratio of fissions in a UO_2 -3 wt% PuO_2 rod in the FERTF element to the maximum UO_2 -2 wt% PuO_2 rod in a 19-rod cluster in the first ring of elements

in the Batch Core was measured to be 1.76. A similar measurement is in progress with natural UO_2 rods in the FERTF assembly. These measurements permit the normalization of a series of calculations which have been made of power generation in the FERTF element relative to the power generation in the adjacent elements in the Batch Core.

CRITICAL EXPERIMENTS IN PuO_2 - UO_2 - H_2O LATTICES

V. O. Uotinen and L. D. Williams

A series of critical experiments has been performed in the Plutonium Recycle Critical Facility (PRCF) using half-inch diameter UO_2 -2 wt% PuO_2 fuel rods moderated with H_2O . The fuel rods are 36 inches long, 0.508 inches in diameter, covered with a Zircaloy-2 sleeve 0.030 inches thick, and each rod contains an average of 1128 g of UO_2 - PuO_2 mixture. The rods were loaded in a hexagonal lattice with a pitch of 0.85 inches which corresponds to a moderator-to-oxide volume ratio of 1.83.

The fuel rods were of three types, differing only in the relative amounts of the various plutonium isotopes. The isotopic compositions of the three types of plutonium are listed in Table I.

Single-zone loadings were investigated using 8 and 24 wt% ^{240}Pu fuel rods. Two-zone loadings with various combinations of the three types of fuel rods were also investigated. All cores were surrounded by an effectively infinite H_2O reflector on all sides except the top. The thickness of the top reflector varied because the moderator height was varied to adjust reactivity.

In each of the loadings, the first experiment was an approach-to-critical to establish the number of fuel rods required for a critical loading. Measurements were then made to determine the reactivity worth of lucite templates, sleeves, and safety sheet followers, and of the control rod fuel follower rods. These results were used to correct the measured critical mass for these perturbations. The resulting corrected critical number of fuel rods represents a "clean" loading. The critical mass measurements are summarized in Table II. The last column of Table II is the "clean" critical number of rods; it includes corrections for the various perturbations that were present in the PRCF. The

TABLE I. Isotopic Composition (in wt%) of the Three Types of Plutonium

Type of Fuel Rod*	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
~8 wt% ^{240}Pu	91.62	7.65	0.70	0.03
~16 wt% ^{240}Pu	81.11	16.54	2.15	0.20
~24 wt% ^{240}Pu	71.76	23.50	4.08	0.66

* The three types of fuel rods are referred to as 8, 16, and 24% ^{240}Pu fuel rods.

TABLE II. Summary of Critical Loadings for 2 wt% PuO_2 - UO_2 Lattices*

Type of Loading	Actual Critical Loading	Effect of 8% Fuel Followers	Effect of Lucite Templates	Effect of Lucite Sheets	Effect of Lucite Sleeves	Refined Critical Loading
Uniform 8% ^{240}Pu	252.0	None	-0.8 $\pm$ 0.4	+0.6 $\pm$ 0.1	-0.2 $\pm$ 0.1	251.6 $\pm$ 0.4
2-Zone 8% - 24% in out	290.0	+5.0 $\pm$ 1	-1.5 $\pm$ 1.0	+1.8 $\pm$ 0.2	-0.3 $\pm$ 0.15	295.0 $\pm$ 1.5
Uniform 24% ^{240}Pu	386.7	+7.6 $\pm$ 0.4	-1.6 $\pm$ 0.3	+1.1 $\pm$ 0.1	-0.3 $\pm$ 0.1	393.5 $\pm$ 0.5
2-Zone 24% - 16% in out	353.35	+5.0 $\pm$ 0.4	-3.3 $\pm$ 1.0	+0.9 $\pm$ 0.2	-0.3 $\pm$ 0.1	355.7 $\pm$ 1.1
2-Zone 16% - 24% in out	345.31	+5.1 $\pm$ 0.4	-3.5 $\pm$ 1.0	+0.55 $\pm$ 0.15	-0.3 $\pm$ 0.1	347.2 $\pm$ 1.1
2-Zone 24% - 8% in out	318.8	+1.6 $\pm$ 0.1	-1.2 $\pm$ 0.6	+0.49 $\pm$ 0.08	-0.3 $\pm$ 0.1	319.4 $\pm$ 0.6
Uniform 8% ^{240}Pu	252.16	none	-0.8 $\pm$ 0.4	+0.19 $\pm$ 0.05	-0.2 $\pm$ 0.1	251.4 $\pm$ 0.4

* All quantities are in terms of number of fuel elements.

critical mass of the loading of 8 wt% ^{240}Pu rods was measured twice. Between measurements the fuel was used in other experiments for which boron was added to the moderator. The excellent agreement between measurements indicates that no boron remained on the fuel elements.

The reactivity worths of fuel elements and voids were measured as a function of radius in the two single-zone cores. The results are plotted in Figures 1 and 2. Integration of the film void data (neglecting interaction between cells) results in values of -3.0 and -2.6 mk/% void, respectively, for the void coefficients of the 8 and 24 wt% ^{240}Pu cores.

Measurements were also made of moderator level reactivity coefficients, moderator temperature reactivity coefficients, neutron distributions, kinetic

parameters, and worths of various neutron absorbing rods. The results of these experiments are being analyzed.

Calculations are being performed with the computer codes THERMOS, HRG, TEMPEST, and HFN to provide analytical results for comparison with experimental results. Cell-average cross sections are generated with the codes HRG, THERMOS and TEMPEST. These homogenized cross sections are then used in a one-dimensional diffusion-theory calculation of k_{eff} using the code HFN with 4 energy groups of neutrons. Preliminary results indicate that the calculated effective multiplication factors are ~1% smaller than measured values.

Calculated moderator void coefficients are -3.7 and -3.3 mk/% void, respectively, for the 8 and 24 wt% ^{240}Pu cores.

FIGURE 1. Fuel and Void Worth--Radial Traverse 2% PuO₂-UO₂, 8% ²⁴⁰Pu, 0.85 in. Lattice

FIGURE 2. Fuel and Void Worth--Radial Traverse 2% PuO_2 - UO_2 , 24% ^{240}Pu , 0.85 in. Lattice

EXPERIMENTS WITH NEUTRON ABSORBING RODS

V. O. Uotinen

An extensive set of experimental data has been accumulated on the worth of neutron absorbing rods in $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ lattices. The fuel rods used in all of these experiments were half-inch in diameter, 36 in. long, clad with 30 mils of Zircaloy-2. Each rod contained an average of 1128 g $\text{PuO}_2\text{-UO}_2$ mixture, of which 2 wt% was PuO_2 . The data for HfO_2 -bearing rods are

summarized in Figure 1. The HfO_2 rods H1 through H5 contain, respectively, 5.11, 3.42, 1.74, 1.03, and 0.52 g/cm³ HfO_2 in aluminum.

A paper entitled "Measurement of the Reactivity Worth of Hafnium Oxide Rods in a $\text{PuO}_2\text{-UO}_2\text{-H}_2\text{O}$ Lattice" was presented at the ANS winter meeting. The paper described critical experiments that were performed in the PRCF and compared the results with the results of subcritical experiments.

FIGURE 1. Summary of Reactivity Measurements of Hafnium Oxide Rods

HTLTR CONTROL ROD EXPERIMENT

R. M. Humes

Introduction

As part of the continuing effort to provide increased accuracy to the design calculations of the HTLTR, the test of a segment of control rod reported herein was conducted.

At the time a previous experiment designed toward the same end⁽¹⁾ was conducted only nonprototypical materials were available. Although efforts were made to estimate the effect of these materials on the results theoretically, a large uncertainty necessarily remained. Hence, with the receipt of the actual materials to be used in the control rods and in the driver rods, it was deemed desirable to repeat the measurements.

In addition to these considerations, the driver fuel has been changed to dense pellets instead of the more loosely packed material envisioned at the time of the previous experiment, the composition of the material which will be used to reduce the temperature coefficient effects in HTLTR has been finalized and the material received, and the B_4C -graphite material called for in the control rods at that time has been removed due to more recent evidence of the transport of large amounts of boron out of this material.

Description of the Experiment

The HTLTR will utilize shutter type control rods similar to those in PCTR but without the neutronic poison contained in the latter⁽²⁾. Briefly, they consist of an inner (I) section

consisting of 6 in. long by 1.365 in. OD cylinders, alternately graphite (IG) and UO_2 -graphite (IU) on a 0.500 in. OD TD-nickel rod and an outer (O) section 2.096 in. OD of alternating 6 in. long graphite (OG) and UO_2 -graphite (OU) cylinders. The UO_2 -graphite cylinders of both the inner and outer sections are made of a graphite base material containing 3 gm/cc of UO_2 , 16% enriched in U-235.

The controlling action is attained as follows. In the least reactive (closed) configuration the IU cylinders are directly under the OU cylinders and thus shielded by them. To increase reactivity, the IU cylinders are moved out of the OU cylinders so that a part of each of them is under an OG cylinder until, in the most reactive (open) position, the control rod consists for neutronic purposes of a continuous cylinder of UO_2 . The difference in the reactivity contribution between the most reactive and least reactive configuration is, by definition, the worth of the control rod.

The HTLTR control rods will have an effective length of 7 ft. Since it is not possible to place an entire control rod in the test cavity of PCTR, 3 ft and 1 ft sections were measured and compared to a 1 ft long stack of driver fuel pellets. The position of the test pieces in relation to the nearest PCTR driver rods is shown in Figure 1.

The following comparisons were made:

1. 3 ft segment of control rod - open versus closed - this and comparison 2 contain B_4C -graphite cylinders, in place of the IG cylinders,

FIGURE 1. Test Position Location

2. 1 ft segment of control rod - open versus closed - (see comment for 1),
3. 1 ft segment of control rod - open versus closed - with IG cylinders,
4. 1 ft segment of control rod - open versus closed - with Gd_2O_3 -graphite cylinders in place of the IG cylinders (see 6),
5. 1 ft segment of driver fuel pellets - fuel versus void - 1-1/16 in. diameter, UO_2 enriched to 5% U-235, total mass about 1.8 kg,
6. 1 ft segment of Gd_2O_3 -graphite rods - rods versus void - 1-1/16 in. diam, 0.65 mg/cm^3 of Gd_2O_3 (cylinders in 4 had a 1/2 in. hole drilled through axially).

Discussion

The results of the measurements are shown in Figure 2 and Table I. The zero reference in Figure 2 is the reactivity of a void in the test position. Also, shown on this figure are the values obtained in the earlier experiment.

A brief description of the differences between the present experiment and the previous one would appear to be apropos at this point.

1. The driver fuel mockup in the previous experiment had an active length of 10.25 in. (the values in Figure 2 and Table I have been scaled up to represent 1 ft) and an oxide density of 7.31 gm/cc while the present element was 12 in. long and had a density of 10.4 gm/cc .

FIGURE 2. Worth of Various Test Pieces

TABLE I. Worths of Tested Components

Description	This Experiment, ϕ	Previous Experiment, ϕ
1 ft of driver	38.5	31.3
Control Rod:		
1 ft with inner graphite	21.7	9.7
1 ft with B_4C	34.9	11.3
1 ft with Inconel	--	12.2
1 ft with Gd_2O_3	25.3	--
3 ft with B_4C	77.6	--
1 ft of Gd_2O_3 -G shim	-22.9	--

2. The IU and OU cylinders contained only 1.6 gm/cc of UO_2 previously; the present contain 3.0 gm/cc.
3. A test of control rod effectiveness using a cylinder possessing neutron absorbing properties between those of the IG and the

B_4C -graphite inner cylinders was previously made using the, at that time, most readily available material, Inconel. In this test, the material was made utilizing the Gd_2O_3 -graphite temperature coefficient shimming material.

4. The PCTR driver fuel was loaded rather differently in the two tests.

The data obtained in this experiment may be analyzed to obtain an estimate of the control rod worth when inserted into the HTLTR, as follows:

- I. From calculations of C. M. Heeb for the HTLTR with a graphite test core of varying dimensions, we can obtain estimates of the

worth of 10 ft long drivers at varying radii as shown in Table II.

Using a logarithmic extrapolation of this data to obtain the worth at the various possible positions of the control rods in HTLTR, we obtain the values shown in Table III.

- II. The values of Table III may now be used to obtain an estimate of the control rod worth. Two adjustments must be applied to the values. First, from Table I, the control rod is worth $21.7/38.5 = 0.564$ times as much as an equivalent length of driver. Second, the control rod is only 7 ft long versus the 10 ft of driver for which the calculations were made. Hence, the equivalent 7 ft driver is worth

TABLE II. Worth of 10 ft of Driver Fuel at Various Radii in HTLTR

Radius, cm	Equiv. Square, in.	Worth, ¢
67.50	47.1	170.1
87.80	61.3	122.8
132.56	92.5	58.8
136.50	95.25	46.2
137.99	96.3	42.3

TABLE III. Worth of 10 ft of Driver Fuel at the Control Rod Positions in HTLTR

Control Rod Position	Size of Square, in.	Worth, ¢
Inner	67	107
Center	83 3/4	72
Outer	100 1/2	30

$$\frac{\int_{-3.5}^{3.5} A \cos \frac{\pi}{10} x dx}{\int_{-5}^5 A \cos \frac{\pi}{10} x dx} = 0.891$$

times as much as a 10 ft driver. Thus, to obtain the worth of an HTLTR control rod we multiply the values of Table III by $0.564 \times 0.891 = 0.503$ to obtain the values in Table IV. Other information which was obtained and may be of use in the future may be summarized as follows:

CR worth with Gd_2O_3 cylinders
 CR worth with graphite cylinders = 1.166

CR worth with B_4C cylinders
 CR worth with graphite cylinders = 1.608

Reactivity addition of CR in least reactive position
 Reactivity addition of driver = 0.626

Worth of Gd_2O_3 shim rod
 Worth of driver = 0.595

TABLE IV. Worth of HTLTR Control Rods at Various Positions

Position	Worth, ¢
Inner	54
Center	36
Outer	15

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2. D. J. Donahue, D. D. Lanning, R. A. Bennett, and R. E. Heineman. "Determination of k_{∞} from Critical Experiments with the PCTR, "Nucl. Sci. and Eng., vol. 4, p. 297. 1958.

FAST REACTORSEVALUATION OF PHYSICS CHARACTERISTICS
OF ALTERNATE FTR CONCEPTS

L. L. Maas

An evaluation of the physics characteristics of various proposed alternate FTR concepts is being carried out. Alternate concepts have been proposed by four organizations: GE, Westinghouse, APDA, and INC. While some of these concepts differ markedly in their mechanical design, the physics characteristics of the concepts are generally quite similar. The reasons for this similarity are simply that all concepts employ the same fuel (viz PuO_2/UO_2) and have comparable core sizes. To make the comparisons more meaningful, all physics calculations have now

been redone using a consistent set of cross sections and consistent methods of analysis. To expedite the comparison, fairly gross core idealizations have been made.

All calculations were carried out using one-dimensional transport theory and a modified 26-group Russian cross section set.

A summary of the results available to date is shown in Table I. The APDA designs are not included since they depart very little from the PNL designs. The data contained in Table I is preliminary and is being evaluated. Some additional results are still being generated.

TABLE I. Comparison of Neutronics Characteristics of Alternate FTR Designs (1-D Models)

	FTR	GE	INC I-L	INC II-L	INC V	Westinghouse	
Core Radius, cm	52.11	54.08	61.99	58.36	59.27	47.14	
Core Length, cm	93.80	91.44	86.36	86.36	86.36	99.06	
Volume, liters	800.0	840.11	858.59 + 184 = 1042.59	924.15	822.57 + 130.61=953.18	691.50	
Core, vol%							
PuO ₂ (0.85 _p TD)	6.88	6.65	7.97	8.14	7.85	7.26	
²³⁸ UO ₂ (0.85 _p TD)	23.12	22.45	23.93	28.16	25.91	21.64	
SS	25.0	5.96	18.31	21.56	19.07	16.53	
Inconel	-	15.31 (Inc-800)	8.22	5.91	8.72	-	
Na	45.0	42.37	41.56	36.23	38.45	54.57	
BeO	-	7.27	-	-	-	-	
Reflector vol%		Region I	II	III		Region I	II
Inconel	80.0	12.06	-	-	80.0	80.0	34.29 80.0
Na	20.0	21.76	50.0	30.0	20.0	20.0	37.14 20.0
SS	-	5.88	50.0	10.0			28.57
Ni	-	60.30	-	-			
B ₄ C	-	-	-	60.0			
Flux/Power Ratio	1.06 x 10 ¹⁶	1.01 x 10 ¹⁶	0.91 x 10 ¹⁶	1.08 x 10 ¹⁶	0.93 x 10 ¹⁶		1.01 x 10 ¹⁶
φ ₀ (P = 400 MW)	0.99 x 10 ¹⁶	0.95 x 10 ¹⁶	0.58 x 10 ¹⁶	0.90 x 10 ¹⁶	0.66 x 10 ¹⁶		1.05 x 10 ¹⁶
φ (> 0.1 MeV)	0.669	0.611	0.670	0.671	0.686		0.689
φ (< 10 keV)	0.064	0.111	0.058	0.066	0.062		0.057
²³⁹ Pu Worth							
δk/(δN ²³⁹ Pu/N ²³⁹ Pu)	0.582	0.541	0.593	0.578	0.585		0.594
B ^{Nat} Worth							
δk/(δN ^{BNat} /N ²³⁹ Pu)	-0.0514	-0.0542	-0.0490	-0.0517	-0.0510		-0.0487
T(dk/dT)	-3.53 x 10 ⁻³	-6.54 x 10 ⁻³	-2.38 x 10 ⁻³	-3.29 x 10 ⁻³	-3.02 x 10 ⁻³		-2.76 x 10 ⁻³
Na Void (core)	-0.0128	-0.0161	-0.0003	-0.0066	-0.0009		-0.0223
Na Coeff (δk/°K)	-3.41 x 10 ⁻⁶	-4.29 x 10 ⁻⁶	-0.069 x 10 ⁻⁶	-1.75 x 10 ⁻⁶	-0.231 x 10 ⁻⁶		-5.96 x 10 ⁻⁶
Central Na Void (R<10 cm)	+0.0003	+0.0001	+0.0001	+0.0002	+0.0001		+0.0003
β	~0.0030	~0.0030	~0.0029	~0.0030	~0.0030		~0.0029
Λ	3.8 x 10 ⁻⁷	5.1 x 10 ⁻⁷	3.1 X 10 ⁻⁷	3.9 x 10 ⁻⁷	3.2 x 10 ⁻⁷		3.4 x 10 ⁻⁷
Mass ²³⁹ Pu (kg)	425.	432.	562.	484.	525.		388.
²³⁸ U/ ²³⁹ Pu	3.59	3.60	3.20	3.69	3.52		3.18
Breeding Ratio	0.48	0.51	0.40	0.49	0.46		0.42

EFFECT OF FUEL ZONING ON POWER DISTRIBUTIONS AND DOPPLER COEFFICIENTS

L. L. Mass

A comparison has been made of the power distribution, flux distribution, and Doppler coefficients for an unzoned

and a two-zoned ceramic fueled FTR core. The core characteristics and results are shown in Table I. Figures 1 and 2 are the power distributions for the unzoned and two-zone cores, respectively.

TABLE I. Comparison of One and Two-Zone Cores

	<u>Unzoned</u>	<u>Zone 1</u>	<u>Zoned</u>	<u>Zone 2</u>
vol% PuO ₂	5.657	5.0		6.85
²³⁸ UO ₂	19.343	20.0		18.15
SS	25.0	25.0		25.0
Na	45.0	45.0		45.0
Void	5.0	5.0		5.0
Outer Core Radius, cm	54.2	38.32		54.2
Outer Reflector Radius, cm	84.2			84.2
Core Length, cm	86.7	86.7		86.7
Total Reflector Savings, cm	22.0		22.0	
Volume, liters	800.0	400.0		400.0
L/D	0.8			
kg ²³⁹ Pu	411.5	181.86	0.8	249.15
k _{eff}	1.026		1.018	
Fertile-Fissile Mass Ratio	3.648	4.267		2.827
$\frac{dk}{dt}$	-4.15×10^{-3}		-4.24×10^{-3}	
P _{pk} /P _{avg} , core	1.439	1.161		1.137
ϕ_{pk}/ϕ_{avg} , core	1.472	1.352		0.993

FIGURE 1. Power Distribution-Unzoned Core

FIGURE 2. Power Distribution-Two Zone Core

DOPPLER COEFFICIENT CALCULATIONS

R. E. Schenter

Doppler broadening is the primary mechanism for reactivity feedback in the ceramic fueled reactor cores currently being investigated for the FTR. Consequently, for safety considerations, it is important to be able to calculate precisely the Doppler coefficient in these cores. This problem is being studied and calculations are underway using the highly detailed fission and capture cross section data obtained from the recent analysis of the "Petrel" bomb test (Nevada Test Site - June, 1965).

Temperature Dependent Self-Shielding Factors

Temperature dependent self-shielding factors, which are used in Doppler coefficient calculations, have been calculated for ^{239}Pu . This has been done using Program FFAC, which calculates

the temperature dependent factors in the group structure of the "Russian" set. Program FFAC uses as input, Program BROAD which calculates temperature dependent cross sections values directly from experimental data. Typical results for ^{239}Pu are given and compared with the Russian set in Table I, where f_f , f_c , f_t , f_e are the fission, capture, total, and elastic scattering self-shielding factors of ^{239}Pu , respectively, and σ_o represents the contribution to the total cross section from all the other isotopes in the reactor calculation.

Crystalline Binding Effects

The effect of crystalline binding on the calculation of the Doppler coefficient is currently being investigated for typical FTR ceramic driver elements (4:1 UO_2/PuO_2). In order to get a preliminary estimate, a calculation has been made using the

TABLE I. Self Shielding Factors - Group 14 (1.0-215 keV)

TABLE I. Self Shielding Factors - Group 14 (1.0-215 keV)															
Temp., °K	f_f at σ_0 equal to				f_c at σ_0				f_t at σ_0			f_e at σ_0			
	10^3	10^2	10	0	10^3	10^2	10	0	10^2	10	0	10^2	10	0	
"Petrel"	300	0.994	0.955	0.863	0.813	0.994	0.954	0.859	0.808	0.953	0.867	0.826	0.983	0.949	0.931
	900	0.994	0.957	0.866	0.817	0.994	0.956	0.863	0.813	0.955	0.871	0.829	0.984	0.951	0.933
	2100	0.995	0.959	0.872	0.824	0.995	0.958	0.869	0.820	0.957	0.875	0.834	0.985	0.953	0.935
"Russian"	300	0.98	0.88	0.67	0.56	0.98	0.87	0.62	0.51	0.89	0.79	0.69	0.87	0.85	0.84
	900	0.99	0.92	0.78	0.69	0.99	0.91	0.76	0.66	0.95	0.90	0.82	0.91	0.87	0.85
	2100	1.00	0.95	0.86	0.82	1.00	0.95	0.84	0.79	0.97	0.94	0.90	0.93	0.90	0.87

Dolling⁽¹⁾ phonon distribution curve for a UO₂ lattice as given by Adkins et al.⁽²⁾ With Dolling's curve, the "effective" -temperature versus temperature dependence has been found where we have used the following equation:

$$kT_{\text{eff}} = \int_0^{\infty} d\omega g(\omega) (\hbar \omega / 2) \coth (\hbar \omega / 2kT),$$

where $g(\omega)$ is the Dolling distribution. Assuming that U₂₃₈ in a UO₂ lattice can be well represented as a free gas with an effective temperature (SCNL-approximation⁽³⁾) and that

$$T \frac{dK}{dT} \Big|_{\text{Free Gas}}$$

is constant, the Doppler coefficient, taking into account binding, becomes

$$T \frac{dK}{dT} \Big|_{\text{Binding}} = \frac{T}{T_{\text{eff}}} \frac{dT_{\text{eff}}}{dT} T \frac{dK}{dT} \Big|_{\text{Free Gas}}$$

Results with the Dolling distribution give a reduction of the "Free Gas"

Doppler coefficient of about 20% at room temperature (300 °K) and 2% at a temperature of 1000 °K. It was also found that for temperatures greater than 200 °K, to a good approximation, the "effective" temperature is given by a Debye distribution ($g(\omega) = \omega^2 / 3\omega_D^3$) with a Debye temperature of 600 °K. The 600 °K value was also consistent with specific heat data ($T > 200$ °K)².

References

1. G. Dolling, R. A. Cowley, and A.D.B. Woods. *Can. J. of Phys.* vol. 43, no. 8, p. 1397. 1965.
2. C. R. Adkins, P. J. Persiani, R. N. Hwang, J. J. Kaganove. *Conf. of Neutron Cross Section Technology*, March, 1966.
3. This is the "short compound nucleus lifetime" or "weak binding approximation" and has been discussed by Adkins et al. (Reference 2). The approximation should be good for the predominant flux in fast reactors where the total level cross section width plus Doppler width is large compared to the maximum binding energy within the crystal.

FTR HETEROGENEITY EFFECTS

R. A. HARRIS

A preliminary study of the heterogeneity effects which may be encountered in the analysis of the FTR and the FTR critical mockup experiments has been made. Typically, FTR design calculations assume that the core materials are homogeneously distributed. The heterogeneity effects in the final FTR design are not expected to be large, since the fuel is in the form of ~ 0.2 in. diameter pins. The critical experiments to be conducted in conjunction with the design of the FTR may, however, contain noticeable heterogeneity effects. It is expected that these experiments will be carried out in the ZPR-III facility. The core materials in this facility are in the form of plates which may extend the entire width and height of the core.

An estimate of the magnitude of such effects was obtained by investigating two cores. These cores were representative of a typical fast reactor critical experiment and the final FTR design.

The results indicate that the heterogeneity effect on reactivity in the ZPR-III, assembly 48, and the FTR are 0.0114 and $\sim 0.0002 \Delta\rho$ respectively. While these effects are negligible in the FTR, the assembly 48 heterogeneity effects account for a critical mass increment of $6.3\% \left(\delta\rho = 0.18 \frac{\delta \text{ vol}}{\text{vol}} \right)$.

The primary mechanism producing the reactivity changes in the ZPR-III assembly 48 was found to be the fast effect in the fissile fuel ($\sim 70\%$); the reduction of the ^{238}U capture accounts for 27% of the total change. If, however, only the low energy depressions of the flux in the fuels are considered, a reactivity change of -28% of the total is observed. This occurs because the fissile fuel is sandwiched between two ^{238}U plates. Thus, any reduction in ^{238}U capture is accompanied by an even greater reduction in the fission in the fissile fuel. The importance of the ^{238}U capture reduction to the total heterogeneity effect due to spatial shielding is, therefore, system dependent.

The results of this study can be used to evaluate the adequacy of future heterogeneity calculations in FTR and ZPR-III cores. The present study was limited to evaluating the heterogeneity effect on reactivity. The methods and models employed, however, yield the local spatial variation in the flux, which can be used to evaluate the heterogeneity effects on other reactor parameters, i.e., Doppler and coolant loss coefficients. A report discussing the details of the calculations has been prepared.

FTR PHASE A CRITICAL EXPERIMENTS

R. A. Bennett, S. Engstrom, and Q. Baird

Details of the Phase-A Critical Experiments were formulated in cooperation with ANL personnel at NRTS, on November 29, 1966.

The main objective of the Phase-A experiments is to provide "clean" control rod worth measurements to supply a basis for experiment-theory correlation.

An outline of the Phase-A, Assembly-48 experiments is given below:

A. Central Control Rod Experiments

- Measurements of spatial worth and reaction rates
- Central sodium column, to be used as base point for subsequent control rod experiments
- Stepwise insertion of a highly enriched boron rod

- Stepwise insertion of a natural boron rod

• Rod heterogeneity test

Reflector Control Rod Experiments

- Addition of inconel reflector sections
- Stepwise insertion of a highly enriched boron rod
- Boron enrichment comparison
- Rod interaction experiments
- Symmetric two rod experiment
- Measurement of spatial distributions of worth and reaction rates for the symmetric two rod experiments

The experiments started on December 1, 1966, and are on schedule to date. Control rod worths have been measured and some preliminary rod calculations have been carried out.

NPTF DESIGN

R. A. Bennett and G. J. Busselman

In order that the measurements in the NPTF be directly applicable to the FTR, it has been conjectured that the NPTF must be an accurate nuclear mock-up of the FTR. In fact, it has sometimes been proposed that an attempt should be made to simulate the fuel depletion and fission product accumulation in the FTR by suitable compositional changes in the NPTF. While such a procedure seems quite feasible, the degree to which the burnup simulations may have to be carried remains to be established.

A series of investigations has been started to obtain some quantitative answers to this question. Reactivity worth calculations have been made and compared in a clean, undepleted core and in a core subjected to a higher de-

gree of burnup. Two exchange worths have been calculated. First a single, exposed FTR element was replaced with a fresh element in a 40,000 MWD burned FTR core. Using a convergence criterion of $10^{-5} \frac{\Delta k}{k}$ for the calculations this exchange was found to be worth 0.820 ± 0.014 mk. In the second replacement a fresh element from an unburned FTR core was replaced by a burned element, from a 40,000 MWD burned FTR core. The exchange worth was found to be 0.80 ± 0.14 mk. This agreement illustrates that one may very well be able to infer worths of small changes in FTR, at various stages of burnup, from measurements in an NPTF which only simulates a single reactor condition.

This conclusion will have to be tested further for other test element configurations.

FTR SHIELDING STUDIES

D. R. Marr and W. L. Bunch

The removal diffusion code MAC was used to estimate attenuation characteristics of a basic radial shield concept for the FTR. The arrangement used in the calculations homogenizes and somewhat idealizes a typical concept for a cylindrical core. The reactor is assumed to consist of an 800 liter, oxide fueled core operating at 1 MW/liter. Composition and thickness of the regions used in the calculation are given in Table I.

Results of the calculation are presented in Figures 1, 2, and 3. Important considerations in evaluating a radial shield are: radiation environment for sensors for the reactor instrument system, heat generation in the shield components, radiation damage to structural components, and activation rates in the secondary sodium. The radial shield concept used in these calculations does not adequately reduce the activation flux entering the secondary sodium; therefore, alternate

materials and shield arrangements will be considered in future calculations to optimize the shield design.

FIGURE 1. Neutron Intensity as a Function of Distance into Shield

TABLE I. Assumed Basic Radial Shield Arrangement

Region Number	Name	Thickness, cm	Composition, vol%
1	Core 1	27.1	PuO ₂ = 5.657, UO ₂ = 19.343, 304 SS = 25.0, Na = 45.0, voids = 5.0
2	Core 2	27.1	
3	Skewed Reflector	17.3	304 SS = 55.7, Ni = 26.2, Na = 18.1
4	Sodium Pool 1	10.0	Na = 100
5	Sodium Pool 1a	35.0	Na = 100
6	Tube Reflector	60.0	Na = 19.4, 304 SS = 17.6, Inconel = 62.9
7	Sodium Pool 2	5.0	Na = 100
8	Vessel	7.0	304 SS = 100
9	Pipe Space	62.0	Void
10	Neutron Shield	152.0	Steel = 25.7, graphite = 54.1, void = 20.2
11	Heat Exchanger Wall	1.0	Steel = 100.0
12	Secondary Sodium	60.0	Na = 100.0

FIGURE 2. Gamma Intensity as a Function of Distance into Shield

FIGURE 3. Fast, Intermediate, and Thermal Neutron Fluxes as a Function of Distance into Shield

SODIUM ACTIVATION IN FTR

W. L. Bunch

Activation of the sodium serving as the FTR coolant has been estimated using results of analyses for an 800 liter, oxide fueled cylindrical core (radius 54 cm, length 87 cm) operating at 400 MW. Radioactivity of the sodium will influence the design of the shield and the location of sensors for the reactor instrumentation system. These preliminary results indicate the approximate

level of activity that will be encountered and the design parameters that influence the activation level.

Two radioactive isotopes of sodium will be formed; ^{24}Na as a result of neutron capture, and ^{22}Na as a result of the (n,2n) process. ^{24}Na has a 15 hour half life; thus, the activation level of this isotope will reach an equilibrium value after a few days of operation. On the other hand, it will decay and become negligible within a

few days following shut down of the reactor. ^{22}Na has a 2.58 year half life; thus, the concentration of this isotope will increase slowly for many years and the activity will not decrease appreciably during reactor shut downs. Both of these sodium isotopes emit relatively high energy gamma radiation that is quite penetrating.

Generation of radioactive sodium within the core was calculated by the relation

$$R_c = \phi N \sigma V$$

where ϕ is the volume average flux capable of causing the activation event, N is the volume average sodium (atom) density in the core, σ is the activation cross section appropriately weighted for the flux distribution, and V is the volume of the core. Values for the calculation of activation of ^{24}Na were taken directly from the results of the core analyses; however, the threshold energy for formation of ^{22}Na is 12.4 MeV and spectral detail at this energy is not obtained directly. In order to estimate the flux above 12.4 MeV, it was assumed the high energy tail of the neutron spectrum could be approximated by a fission spectrum.

Activation of the sodium external to the core was estimated by:

$$R_1 = n_1 \Sigma_{na} / \Sigma_t$$

where n_1 is the rate at which neutrons are leaking from the core, Σ_{na} is the appropriate macroscopic activation cross section considering the sodium content of the regions adjacent to the core, and Σ_t is an appropriate total macroscopic cross section for these same regions. When considering activation of ^{24}Na , total neutron leakage was used and Σ_t was the total spectrum weighted absorption cross section of the reflec-

tor regions. For activation of ^{22}Na , n_1 was estimated for neutrons with energy in excess of 12.4 MeV and Σ_t was based on the macroscopic "removal" cross section for these high energy neutrons.

Results of the calculations are presented in Table I, in which it has been assumed that the axial reflectors would contain 70 vol% sodium and 30 vol% stainless steel and the radial reflector would contain 55 vol% stainless steel, 25 vol% nickel, and 20 vol% sodium. It can be seen that for the composition considered activation of ^{24}Na occurs predominantly in the axial reflectors. Thus, for a given power level, an appreciable decrease in the ^{24}Na activation rate could be effected only by materially reducing the sodium content in these regions. Activation of ^{22}Na takes place predominantly in the core, making it difficult to reduce the production rate of this isotope.

The saturated specific activity of the sodium in the system can be estimated (assuming the radioactive nuclei become homogeneously distributed) by dividing activation rates by the total volume of sodium. For example, for a 5800 ft³ system ($1.6 \times 10^8 \text{ cm}^3$), the above results correspond to 0.085 Ci of ^{24}Na per cm³ and 25 μCi of ^{22}Na per cm³. Dilution is one method of reducing the specific activity of a system, though not of reducing the number of radioactive atoms.

TABLE I. Activation Rates for ^{22}Na and ^{24}Na in FTR

	^{24}Na	^{22}Na
In Core	0.7×10^{17}	1.1×10^{14}
In axial reflectors	3.8	0.4
In radial reflector	0.5	0.1
	5.0×10^{17}	1.5×10^{14}

IRRADIATED FUEL SHIELD STUDIES

W. L. Bunch

Survey calculations were made to determine the magnitude of the neutron sources that might be associated with irradiated FTR fuel and test specimens to determine if neutron shielding might be required. The four potential neutron sources that were considered are: delayed neutrons, spontaneous fission neutrons, neutrons from the (α, n) reaction, and neutrons from the (γ, n) reaction. It is concluded that significant neutron shielding will be required only if beryllium is present in the irradiated fuel.

Delayed neutrons will not be important under the present concept that fuel will be discharged after the reactor has been shut down. If charge-discharge during operation were to be carried out, it would be necessary to remove the fuel slowly to permit the delayed neutrons to decay. The neutron generation rates due to spontaneous fission and the (α, n) reaction are summarized in Tables I and II, respectively, based on fuel composition for a typical 800 liter oxide reactor. If a fuel assembly were to contain about 40,000 g of the assumed mixture of uranium and plutonium, these two

TABLE I. Neutrons from Spontaneous Fission in FTR Fuel

Isotope	Spontaneous Fission Rate neutrons/ g_i sec	Neutrons per Fission	Assumed Composition g_i/g_m	neutrons/ g_m sec
^{238}U	7.0×10^{-3}	2.0	0.7630	0.011
^{238}Pu	1.1×10^3	2.2	0.0002	0.48
^{239}Pu	1.0×10^{-2}	2.2	0.2100	0.0046
^{240}Pu	4.6×10^2	2.2	0.0230	23
^{242}Pu	8.2×10^2	2.2	0.0002	0.36

Total = 24.

TABLE II. Neutrons from (α, n) Reaction in FTR Fuel

Isotope	Reaction Rate neutrons/ g_i sec	Assumed Composition g_i/g_m	neutrons/ g_m sec
^{238}Pu	1.4×10^4	2.0×10^{-4}	2.8
^{239}Pu	4.5×10^1	2.1×10^{-1}	9.5
^{240}Pu	1.7×10^2	2.3×10^{-2}	3.9

Total = 16.

Note: g_i = grams of isotope, g_m = grams
of plutonium and uranium.

neutron sources combined would generate about 1.6×10^6 n/sec. This generation rate would exist whether or not the fuel had been irradiated; therefore, special handling techniques will probably be required to protect operating personnel from the unirradiated fuel. It is unlikely that this source of neutrons will require special shielding for handling the irradiated fuel.

The (γ, n) reaction will generate a significant number of neutrons in irradiated fuel that contains beryllium. The threshold energy for this reaction is approximately 1.67 MeV; it is estimated that the photon flux above this

energy at the center of a fuel assembly might exceed 10^{13} photons/cm²sec 8 hr after shutdown. Since the photoneutron cross section for these photons is about 5×10^{-28} cm², a neutron production rate of about 10^7 n/cm³sec will result per volume percent of contained beryllium. For volume of fuel contained in a typical FTR element, the (γ, n) reaction would generate about 10^{11} neutrons/sec per volume percent of beryllium. A neutron source of this magnitude would require that neutron shielding be incorporated in some types of fuel handling systems.

CRITICAL MASS PHYSICS

THEORETICAL ANALYSES OF HOMOGENEOUS PLUTONIUM CRITICAL EXPERIMENTS

C. R. Richey

Introduction

It has been the practice, at least in the United States, to base criticality safety guidance for handling and processing fissile materials primarily on data obtained experimentally. It is not always feasible, however, to determine all the desired data experimentally and a need exists for reliable calculations to supplement the existing data and reduce the cost of experimental programs. For calculational techniques to meet the necessary requirements for nuclear safety criteria, they must have 1) realistic input, 2) convincing method, and 3) a wide enough range of checks against experiment to establish error patterns.

Many critical experiments have been conducted to date in support of nuclear criticality safety, including a large number of critical experiments with hydrogenous plutonium mixtures having simplicity in geometry, reflection, and composition, making them amenable to theoretical study.^(1,2,3,4) This report presents a computational analysis of representative data from the accumulation of "clean" experiments with plutonium fueled assemblies. Various approximations for simplifying the transport equation and cross-section data are evaluated, with

expected errors established. Values for critically safe masses and dimensions are recommended for aqueous $\text{Pu}(\text{NO}_3)_4$ solutions and ^{239}Pu -water mixtures.

Method of Calculation

Two sources of error are obviously present in any theoretical analysis of neutron multiplying media, one from uncertainties in microscopic cross-section data and the other in the approximations used for simplifying the transport equation. In the subsequent study, criticality calculations were made using both the S_n approximation of Carlson^(5,6) and multigroup diffusion theory. First order allowance for anisotropic neutron scattering was provided in the S_n calculations.

Multigroup constants, generally 17 fast groups and one thermal group, were obtained with the GAMTEC-II code.⁽⁷⁾ The fast group constants, $E > 0.683$ eV, were averaged over a 64-group slowing down spectrum computed with the B_1 approximation to the Boltzman equation and the GAM-I cross-section library.⁽⁸⁾ The one thermal group constants were averaged over a Wigner-Wilkins spectrum utilizing the SOFOCATE library.⁽⁹⁾ Thermal neutron data for ^{239}Pu were normalized to the 2200 m/sec data of Sher-Felberbaum.⁽¹⁰⁾

Resonance absorption by ^{240}Pu was computed from resonance parameters

in the GAM-I library by the narrow resonance (NR) and narrow resonance-infinite mass (NRIM) approximations.⁽¹¹⁾ In particular, the effective absorption cross section for a resonance line within the n^{th} broad group was defined as

$$\sigma_n = \frac{\int_{u_i}^{u_{i+1}} \sigma(u) \phi(u) du}{\int_{u_i}^{u_{i+1}} \phi(u) du} = \frac{\bar{\phi}_{as.} \sum_i RI_i}{\sum_i \bar{\phi}_i \Delta u}$$

where $\bar{\phi}_i$ is the lethargy flux and RI_i the resonance integral for the i^{th} fine group. $\bar{\phi}_{as.}$ is the asymptotic lethargy flux just above the resonance line.

Reference Experiments

In support of Nuclear Criticality Safety, a large number of critical experiments have been performed with hydrogenous plutonium solutions having simplicity in geometry, reflection, and composition. The first experiments were conducted in 1950 to 1951 by Kruesi, et al., for dilute solutions, ~20 to 80 g Pu/liter, in spherical and cylindrical geometries.⁽¹⁾ Isotopic ^{240}Pu content ranged from ~0.5 to 4%. More recent experiments conducted by Lloyd, et al., extended the criticality data to higher concentrations of plutonium, up to ~300 g Pu/liter.⁽²⁾ These experiments were in the form of bare and water-reflected spheres with the plutonium fuel containing 4.6% ^{240}Pu . Other experiments involving dilute solutions 20 to 60 g Pu/liter, in water reflected cylinders have been

performed at Saclay.⁽³⁾ The primary difference between these experiments and the cylindrical experiments of Kruesi, et al., is that the former vessels were reflected on the lateral surfaces only, thus more amenable to theoretical study. The cylinders utilized in the experiments of Kruesi were reflected on all surfaces.

Particularly lacking are critical experiments with clean systems of homogeneous hydrogen moderated plutonium in the undermoderated range. One series of experiments with PuO_2 -polystyrene compacts having a H/Pu atomic ratio of 15 was reported by Richey, et al.⁽⁴⁾ Experiments were conducted with rectangular prisms ranging from near cube, to long columns, to thin slabs. Criticality data was obtained for bare and Plexiglas ($\rho = 1.2 \text{ g/cm}^3$) reflected arrays. Two sets of compacts were tested, one containing plutonium with 2.2% ^{240}Pu and the other having 8.0% ^{240}Pu .

Comparison of Theory and Experiment

Spheres and Infinite Slabs

For purposes of comparing calculational methods and cross-section data, critical eigenvalues were computed for several $\text{Pu}(\text{NO}_3)_4$ solutions and the experiments with PuO_2 -polystyrene compacts (H/Pu = 15). The critical solutions selected for the comparison are representative of the data available in spherical geometry. Summarized in Table I are critical eigenvalues computed by S_4 theory with the Hansen-Roach cross-section set and the 18-group constants

TABLE I. Computed Eigenvalues for Critical Assemblies

Sphere Diameter, in.	Reflector	Pu, g/liter	NO ₃ , g/liter	H ₂ O, g/liter	k _{eff} (Diffusion Theory)		k _{eff} (Transport Theory)	
					GAMTEC-II, Sher Data	GAMTEC-II, Leonard Data	GAMTEC-II, Sher Data	Hansen- Roach Data
Aqueous Pu(NO ₃) ₄ Solutions (4.6% ²⁴⁰ Pu)								
14	Full Water	33.0	162	924.9	0.999	1.025	1.015	1.027
14	Full Water	47.9	465	774.4	1.005	1.031	1.019	1.027
11.5	Full Water	73.0	86	970.9	1.001	1.028	1.021	1.025
11.5	Full Water	268.7	346	868.0	1.001	1.030	1.018	1.042
15.2	Full Water	24.4	58	976.9	1.000	1.023	1.014	1.030
15.2	Full Water	38.7	517	736.6	1.007	1.034	1.013	1.029
15.2	0.048 in. SS	39.0	64	977.8	0.997	1.024	1.006	1.011
15.2	0.048 in. SS	172.8	483	765.8	0.993	1.029	1.010	1.017
Mean Value for k _{eff} *					1.000 ± 0.004	1.028 ± 0.004	1.015 ± 0.005	1.026 ± 0.009

PuO₂-Polystyrene Assemblies (H:Pu = 15)

Infinite Slab Thickness, cm	Reflector	²⁴⁰ Pu				
16.09±0.41	Bare	2.2	0.994	---	1.004	1.043
5.99±0.10	Full Plexiglas	2.2	0.993	---	0.991	1.069
18.48±0.41	Bare	8.0	0.984	---	0.988	1.011
7.38±0.09	Full Plexiglas	8.0	0.982	---	0.979	1.046

* Uncertainties denote standard deviation from the mean value.

provided by the GAMTEC-II code with the Sher-Felberbaum normalized ²³⁹Pu data for the thermal group. Also given are eigenvalues computed with 18-group diffusion theory utilizing the GAMTEC-II group constants with both the Sher-Felberbaum ²³⁹Pu data and cross-section data recommended by Leonard.⁽¹²⁾

It is easily seen that the best comparison between experiment and theory was obtained using the GAMTEC-II group constants with the Sher-Felberbaum normalized ²³⁹Pu data. The S₄ computed eigenvalues were consistently greater than unity for the Pu(NO₃)₄ solutions with the Hansen-Roach cross sections giving the larger values having a mean eigenvalue of 1.026 ± 0.009. An eigenvalue of 1.026 underestimates the critical volume of a 14 in. sphere by approximately 10%. Also, the Hansen-Roach set shows a larger

discrepancy at the higher Pu concentrations (H/Pu = 15) where the GAMTEC-II group constants give results that agree rather well with experiment. On the other hand, and somewhat surprising is that diffusion theory provides better overall agreement than does S₄ theory, even for the higher plutonium concentrations. This certainly is no indication that the lower order diffusion theory is more accurate, but is indicative that compensating errors exist in the cross-section data and diffusion theory approximations.

What improvements, if any, can be gained between experiment and theory by going to higher order S_n approximations and higher than first order allowance for anisotropic scattering by hydrogen? As easily seen from Table II where eigenvalues are given through S₁₆ for the 47.9 g Pu/liter solution in the

water reflected 14 in. sphere, a converged eigenvalue is obtained with the S_4 approximation. Table III summarizes for a bare 15.2 in. sphere with a 39.0 g Pu/liter solution, eigenvalues computed with anisotropic scattering defined through the P_3 term in the Legendre expansion. The resulting eigenvalue obtained with P_1 scattering was within 3 mk of the converged value using the higher order P_3 expansion. It was concluded that little improvement in the computed eigenvalue is obtained by utilizing higher order, and more costly, calculations than the S_4 approximation with P_1 scattering.

TABLE II. Eigenvalues as a Function of S_n Order (47.9 g Pu/liter, 14 in. Reflected Sphere)

S_n Order	k_{eff}
4	1.019
8	1.020
16	1.020

TABLE III. Eigenvalues as a Function of Anisotropic Scattering Order

Order Anisotropic Scattering	k_{eff}
P_0	1.1853
P_0^*	0.9634
P_1	1.0064
P_2	1.0092
P_3	1.0090

* σ_{tr} modified cross section

Diffusion theory utilizing GAMTEC-II group constants and the Sher-Felberbaum data for ^{239}Pu provided the best overall agreement with the selected experiments. Although compensating errors exist in the method, if similar agreement is obtained for the large amount of existing experimental data for aqueous plutonium solutions, reliable survey calculations can be made in support of Nuclear Criticality Safety. Recorded in Table IV are computed eigenvalues for a large number of critical $\text{Pu}(\text{NO}_3)_4$ solutions in spherical geometry. As easily noted, the agreement between experiment and theory is excellent.

Cylinders

Summarized in Table V are eigenvalues computed for aqueous $\text{Pu}(\text{NO}_3)_4$ solutions critical in cylinders of finite height. Each cylinder was fully reflected on the lateral surface by water. The eigenvalues were computed with GAMTEC-II cross-section data using one-dimensional diffusion theory. Axial leakage was provided by modifying the absorption cross sections in the usual diffusion theory approximation

$$\Sigma_a(\text{eff}) = \Sigma_a(\text{actual}) + DB_A^2$$

where D is the diffusion coefficient and B_A^2 the axial buckling.

TABLE IV. Diffusion Theory Calculated Eigenvalues for $\text{Pu}(\text{NO}_3)_4$ Solutions

Pu, g/liter	% ^{240}Pu *	NO_3 , g/liter	H_2O , g/liter	Calc., k_{eff}	Pu, g/liter	% ^{240}Pu	NO_3 , g/liter	H_2O , g/liter	Calc., k_{eff}
<u>11.5 in. Reflected Sphere**</u>					<u>14 in. Reflected Sphere</u>				
73.0	4.6	86.0	970.9	1.001	26.33	0.54	77.3	964.7	0.998
96.0	4.6	203.0	918.3	1.002	26.77	0.54	107.0	951.7	0.996
119.0	4.6	245.0	904.0	1.006	27.91	0.54	138.0	935.2	1.000
132.0	4.6	281.0	877.7	0.998	28.63	0.54	188.0	912.8	0.997
140.0	4.6	284.0	880.8	1.001	27.95	1.76	109.0	950.1	0.997
268.0	4.6	346.0	868.0	1.001	28.78	3.12	87.4	959.7	0.997
<u>12.0 in. Reflected Sphere</u>					30.15	3.12	109.0	951.3	1.004
50.39	3.12	138.0	941.8	0.998	30.33	3.12	143.0	933.5	0.998
51.80	3.12	163.0	931.2	0.999	31.79	3.12	208.0	903.3	0.997
56.20	3.12	207.0	902.5	0.997	35.82	3.12	308.0	851.1	1.001
59.93	3.12	237.0	889.7	1.001	39.62	3.12	408.0	797.4	0.998
63.75	3.12	270.0	877.8	1.004	29.94	4.05	87.5	959.3	0.998
70.22	3.12	322.0	844.9	0.999	30.75	4.05	119.3	945.2	0.999
77.22	3.12	359.0	828.6	1.002	34.07	4.05	210.7	901.9	1.003
<u>13.0 in. Reflected Sphere</u>					36.38	4.05	272.4	870.1	1.003
33.17	1.76	86.2	964.4	0.994	38.72	4.05	335.2	837.8	1.006
34.18	1.76	116.0	948.3	0.998	41.12	4.05	384.9	809.1	1.001
37.99	3.12	132.0	939.0	1.000	31.04	4.40	127.0	942.4	0.997
38.83	3.12	156.0	931.1	1.001	32.41	4.40	158.0	927.9	1.001
41.13	3.12	205.0	905.8	1.001	33.0	4.6	162.0	924.9	0.999
44.63	3.12	269.0	873.2	1.002	38.4	4.6	292.0	861.5	1.004
36.27	3.12	93.1	960.5	0.999	39.2	4.6	313.0	841.6	0.998
<u>15.0 in. Reflected Sphere</u>					47.9	4.6	465.0	774.4	1.005
25.02	3.12	116.0	945.3	0.998	<u>15.2 in. Bare Sphere</u>				
25.83	3.12	147.0	931.9	1.000	39.0	4.6	63.8	977.8	0.997
27.05	3.12	212.0	898.8	0.998	172.8	4.6	483.0	765.8	0.993
<u>15.2 in. Reflected Sphere</u>					<u>16.0 in. Bare Sphere</u>				
24.4	4.6	58.2	976.9	1.000	34.8	4.17	104.4	955.8	1.001
38.7	4.6	517.0	736.6	1.007	38.3	4.17	180.0	918.9	1.003
					43.2	4.17	281.8	866.6	1.001

* All experiments conducted by Kruesi, et al.,⁽¹⁾ except 4.6% ^{240}Pu which were performed by Lloyd, et al.⁽²⁾

** Fully water reflected

The computed eigenvalues, Table V, are all less than unity but exhibit a trend toward unity with increasing critical height. This indicates that the DB_A^2 approximation overestimated the axial leakage. A least squares analysis of the data, Figure 1, gives a corresponding eigenvalue for an infinitely long cylinder of 1.009 ± 0.005 . Thus it is concluded that 18-group diffusion theory calculations for infinitely long cylinders give

eigenvalues within the same accuracy obtained for spherical geometries.

Limiting Critical ^{239}Pu Concentration

The limiting critical concentration for ^{239}Pu (the concentration for which k_{∞} is unity) was determined experimentally to be $8.0 \pm 0.3 \text{ g } ^{239}\text{Pu/liter}$ by Masterson, et al.⁽¹³⁾ A re-examination of the experiment revealed an error in converting the H:Pu atomic ratio for which $k_{\infty} = 1$ to the equivalent

TABLE V. Calculated Eigenvalues for Water Reflected Cylinders* (^{240}Pu content $\sim 1.5\%$, stainless steel wall thickness of 0.12 in.)

Pu, g/liter	Total NO_3 , g/liter	Sp. Gr. g/cm ³	Critical Height, cm	Calc., k_{eff}
Cylinder Diam, 11.6 in.				
44.7	174	1.132	31.19 $\pm$ 0.05	0.9846
41.0	170	1.126	33.15 $\pm$ 0.05	0.9870
38.0	162	1.118	35.36 $\pm$ 0.05	0.9901
35.25	161	1.115	37.95 $\pm$ 0.05	0.9914
32.6	157	1.111	41.15 $\pm$ 0.05	0.9922
30.7	157	1.109	44.81 $\pm$ 0.05	0.9936
28.9	153	1.106	49.56 $\pm$ 0.05	0.9958
Cylinder Diam, 12.8 in.				
45.45	177	1.133	26.26 $\pm$ 0.05	0.9739
38.17	176	1.113	29.01 $\pm$ 0.05	0.9714
33.28	168	1.118	32.72 $\pm$ 0.05	0.9882
30.07	161	1.111	35.59 $\pm$ 0.05	0.9886
27.97	152	1.105	38.46 $\pm$ 0.05	0.9909
25.67	146	1.110	43.74 $\pm$ 0.05	0.9989
23.45	145	1.094	51.56 $\pm$ 0.05	0.9949

* Reflected on lateral surface only.

concentration of ^{239}Pu in water. The corrected value for the limiting critical concentration is 7.8 ± 0.3 g ^{239}Pu /liter. A corresponding value of 7.5 ± 0.1 g ^{239}Pu /liter was calculated utilizing S_4 theory with the GAMTEC-II group constants and the Sher-Felberbaum ^{239}Pu data. This result is based on the hypothesis that neutron leakage is correctly computed by S_4 theory and the limiting critical concentration occurs when the computed $k_{\infty} = 1.015 \pm 0.005$ (the corresponding mean eigenvalue for critical spheres recorded in Table I).

Parametric Survey Calculations

Evident from the previous section is that diffusion theory utilizing GAMTEC-II group constants and Sher-Felberbaum data for ^{239}Pu provide a

FIGURE 1. One-Dimensional Diffusion Theory Eigenvalues for $\text{Pu}(\text{NO}_3)_4$ Solutions in Water-Reflected Cylinders

means for making reliable survey calculations for aqueous plutonium solutions up to ~ 1000 g Pu/liter and 5% ^{240}Pu . Survey calculations made for $\text{Pu}(\text{NO}_3)_4$ solutions and ^{239}Pu -water mixtures are summarized in Tables VI to VIII.

Recommended values of critically safe masses [grams ^{239}Pu for $\text{Pu}(\text{NO}_3)_4$ solutions] are given in Table VI. Maximum safe dimensions (diameters and slab thicknesses) are summarized in Table VII for spheres, infinitely long cylinders, and infinite slabs. Also given in Tables VI and VII for comparison are corresponding values for criticality; i.e., $k_{\text{eff}} = 1.0$. Corresponding values are given in Table VIII for water reflected ^{239}Pu -water mixtures. Since minimum dimensions for ^{239}Pu atoms in water are

nonexistent (dimensions become progressively smaller with increasing Pu density), the "safe" and minimum critical dimensions given in Table VIII are for solutions ≤ 250 g ^{239}Pu /liter.

The recommended values given in Tables VI to VIII were based on the following criteria: (1) acid molarity is zero; (2) plutonium is totally ^{239}Pu ; (3) for spherical geometry the computed eigenvalue for the critically safe data was 0.986, an eigenvalue reduced from unity by twice the uncertainty observed between experiment and theory; (4) for infinite slabs and infinitely long cylinders the critically safe eigenvalue was taken as 0.98; (5) a 1/8 in. stainless steel shell enclosed the nonreflected vessels. The lower eigenvalue was chosen for the slabs and cylinders, since the theoretical checks were less numerous than those for spherical geometries.

Summary

Multigroup diffusion theory with GAMTEC-II group constants and the Sher-Felberbaum data for ^{239}Pu provided the best overall agreement between experiment and theory and is an accurate means of computing criticality for

TABLE VI. Minimum Critical and Safe Masses for $\text{Pu}(\text{NO}_3)_4$ Solutions

Solution	Reflector	Safe Mass, g ^{239}Pu	Minimum Critical Mass, g ^{239}Pu
$\text{Pu}(\text{NO}_3)_4$	Full H_2O	510	547
$\text{Pu}(\text{NO}_3)_4$	1/8 in. Stainless Steel	850	905

TABLE VII. Minimum Critical and Safe Dimensions for Slabs, Cylinders, and Spheres of $\text{Pu}(\text{NO}_3)_4$ Solution

Vessel Geometry	Full H_2O Reflected		1/8 in. SS Shell	
	Safe Dimension, cm	Critical Dimension, cm	Safe Dimension, cm	Critical Dimension, cm
Infinite Slab Thickness	5.79	6.21	13.07	13.48
Infinite Long Cylinder Diam	15.68	16.30	22.62	23.26
Sphere Diam	24.56	25.14	31.30	31.88

TABLE VIII. Minimum Critical and Safe Dimensions for Slabs, Cylinders, and Spheres of Water Reflected ^{239}Pu - H_2O Mixtures: $\rho \leq 250$ g Pu/liter

Vessel Geometry	Safe Dimensions, cm	Critical Dimensions, cm
Infinite Slab Thickness	5.46	5.83
Infinite Long Cylinder Diam	15.10	15.56
Sphere Diam	23.60	24.14
Minimum Critical Mass = 531 g ^{239}Pu		
Maximum Safe Mass = 495 g ^{239}Pu		

hydrogeneous plutonium mixtures up to approximately 1.0 g Pu/cm^3 . Using the more accurate S_4 theory gave eigenvalues approximately 1.5% in excess of unity for critical $\text{Pu}(\text{NO}_3)_4$ solutions. Equivalent S_4 calculations for the epithermal PuO_2 -polystyrene experiments (H:Pu atom ratio = 15), however, were in consistent agreement with both experiment and diffusion theory. It is also expected that quite satisfactory results may be obtained with S_4 theory over the whole range of Pu densities if some adjustments are made to the ^{239}Pu data below 0.7 eV. The well-known Hansen-Roach cross-section sets gave eigenvalues that consistently exceeded unity by approximately 2.6%.

The effects on criticality of ^{240}Pu were accurately predicted using the NR and NRIA approximations for isotopic contents up to about 5%. Some difficulty was experienced in computing the PuO_2 -polystyrene compacts containing

8% ^{240}Pu . Further investigations of higher ^{240}Pu content were prohibited by the scarcity of experimental data.

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PULSE NEUTRON STUDIES - G. M Hess

Some exploratory pulse measurements were performed on a multiple slab Pu-Lucite system to investigate the $k\beta/\lambda$ reactivity technique.

The system consisted of three slabs of Pu-polystyrene blocks ($H/Pu = 15$, 2% ^{240}Pu) nominally 12 in. long, 3 in. thick, and approximately 12 in. high, separated by 1 in. slabs of Lucite. There was also a 1/2-in. Lucite slab on the outside of the outer fuel slabs. The overall H/Pu ratio was 21.7.

The system was pulsed at a nominal 20 pulses/sec for various loadings during the approach to critical. Various data collection schemes were used to investigate their effect on the results.

The resulting decay curves were least squares fit for the minimal root-mean-square error in the decay constant, and the prompt and delayed neutron contributions separated. These fractions of the total decay curve were then used to iterate for $k\beta/\lambda$.

It was found, as expected, that the value of $k\beta/\lambda$ was very sensitive to the background estimate. For example, a

0.2% background variation changed the value of $k\beta/\lambda$ by 4%. For this reason, the raw data was also fit for the minimal root-mean-square error in the background. This value of the background, along with the minimal rms decay constant were then used to iterate for $k\beta/\lambda$. Table I is the experimental data.

In Figure 1 the reactivity in dollars, obtained from the experimental decay constant and $k\beta/\lambda$, is plotted versus the ratio of the decay constant to that at critical. The critical decay constant was obtained by extrapolating a plot of inverse count rate and the decay constant at various loadings, and is believed to be accurate to <5%.

It is interesting to note the deviation of the experimental points from the line marked $\Lambda/\Lambda_c = 1$ which shows the variation of the generation time with reactivity.

All the experimental Points except 4 and 6 were taken by pulsing after the total delayed precursor yield had died off to less than 95% of its equilibrium value. The pulsing was

TABLE I. Experimental Decay Constant and $k\beta/\lambda$

Pulse	Height in Blocks	α , sec ⁻¹	$k\beta/\lambda$
1	6.438	748 $\pm$ 2	489.8
2	6.438	762 $\pm$ 1	499.6
3	6.375	1277 $\pm$ 3	597.3
4	6.375	1275 $\pm$ 2	634.9
5	6.375	1291 $\pm$ 3	613.0
6	6.375	1310 $\pm$ 3	643.3
7	6.312	1396 $\pm$ 6	661.4
8	6.312	1404 $\pm$ 3	622.9
9	6.250	2030 $\pm$ 5	713.0
10	6.250	2059 $\pm$ 5	806.0

* Fitting errors shown.

continued for 400 or 600 seconds, more than ample time for the delayed precursors to build up to equilibrium.

Point 4 was pulsed, starting with dead precursors, for 400 seconds, and then swept without pulsing for an additional 230 seconds while the precursors died off. For this Point, which was the same physical system as Points 3 and 5, the value of $k\beta/\lambda$ was higher, and thus the reactivity lower, than for Points 3 and 5. This is reasonable, since sweeping during the decay of the precursors after the end of the pulsing essentially increases the contribution from the delayed neutrons, which in turn requires a larger $k\beta/\lambda$ to iterate correctly.

FIGURE 1. Reactivity Versus Decay Constant Ratio

Point 6 was taken by first building up the precursors to equilibrium and then recording data until the pulsing was stopped. Again, $k\beta/\ell$ was larger than for Points 3 and 5.

Ideally, one would like to have a constant delayed source and not have to worry about the rise and fall of this quantity at the start and end of a pulsing experiment. However, it is time-consuming and costly in terms of equipment life to build up these delays before taking data. It appears that starting with the precursors dead and sweeping after the pulsing for a length of time required for the precursors to die off accounts for all the delayed neutrons produced by the pulsing.

In systems with a large background, however, sweeping after the pulsing tends to mask the prompt portion of the decay curve. For such systems it may be worthwhile to obtain a constant delayed background before accumulating data. These questions are being looked into further.

If a value of the effective delayed neutron fraction is known, or assumed, the multiplication constant and life-time may be calculated from the $k\beta/\ell$ value and the decay constant. If the decay constant is written as

$$\alpha = \frac{\rho - \beta}{\Lambda}$$

and the ratio to that at critical taken

$$\frac{\alpha}{\alpha_c} = \frac{\rho - \beta}{\Lambda} / \frac{-\beta}{\Lambda_c} = (1 - \beta) \frac{\Lambda_c}{\Lambda}$$

it can be seen that the experimental reactivity in dollars along with the generation time (ℓ/k) calculated by assuming a β can be used to compare with the experimental ratio of the decay constant.

Figure 2 is just such a plot for three assumed values of β . It can be seen that for $\beta = 0.0025$, the experimental points are the same as those back-calculated from the value of $k\beta/\ell$. This does not insure that the $k\beta/\ell$ calculation is correct, but rather indicates a consistency. The calculation of the actual β for the system would shed light on the usefulness of this method of testing the $k\beta/\ell$ calculation. Such a calculation is pending.⁽¹⁾

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FIGURE 2. Comparison of Experimental and Backcalculated Decay Constant Ratios

NEUTRON PHYSICS

HIGH RESOLUTION NEUTRON INELASTIC SCATTERING STUDY OF WATER AND ICE*

O. K. HARLING

Abstract of paper presented to the American Crystallographic Association, January 26, 1967.

The Battelle rotating crystal spectrometer⁽¹⁾ has been used to study the atomic and molecular motions of water at 268 and 299 °K. Initial energies of 0.1515 ± 0.0015 , 0.304 ± 0.0073 and 0.608 ± 0.021 eV were used. Energy spectra of the scattered neutrons were measured by time-of-flight at scattering angles from 15 to 155 degrees. These conditions correspond to studying the dynamics of the samples over a range of energy and momentum wave vector transfers corresponding to 0.6 eV and $32 \text{ \AA}^{-1}$ respectively. The double differential cross-section obtained from the scattered neutron spectra exhibits clearly resolved phonon peaks corresponding to the hindered rotation and the high energy bending and stretching vibrations of the H₂O molecule. These results constitute the first known observation of well resolved high energy vibrations in H₂O,

using the neutron scattering technique. Resolution on scattered neutron energy was good enough to permit determinations of the widths associated with the observed phonon peaks. The energies and reduced width determinations of the hindered rotation, bending mode and the average energy of the stretching modes of H₂O are given below, with all energies in meV.

Temperature	E_r	ΔE_r	E_1	ΔE_1	E_2	ΔE_2
299 °K	~64		206	23	474	184
268 °K	93	~63	204	47	485	215

Measurements made at 15° scattering angle.

The angular distribution of inelastic scattering in the rotation band of ice was compared with the expected distribution for a spherical harmonic oscillator potential.⁽²⁾ To a good first approximation the dynamical unit for hindered rotation in ice appears to be bound in a parabolic potential well.

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* Work performed under U.S. Atomic Energy Commission Contract AT(45-1)-1830

COMPILATION OF DOUBLY DIFFERENTIAL
CROSS SECTIONS AND THE SCATTERING
LAW FOR H₂O AND D₂O AT 299 °K AND
FOR H₂O AT 268 °K*

O. K. Harling

This report contains a compilation of slow neutron inelastic scattering cross sections for room temperature H₂O and D₂O and for H₂O at five degrees below its freezing point. The double differential cross section and from these the Egelstaff scattering law,⁽¹⁾ have been obtained by using the Battelle Rotating Crystal time-of-flight spectrometer.⁽²⁾ Experimental conditions were designed to provide cross sections over a wide range of energy and momentum transfer space. Energy transfers, by downscattering, of up to 0.6, 0.21 and 0.6 eV with wave vector ($\vec{k} = \vec{P}/\hbar$) transfers of up to 32, 9.5 and 32 Å⁻¹ were used, respectively with H₂O, D₂O and light ice. Such an extensive grid is required to accurately represent the scattering kernel for reactor thermalization calculations and for the derivation of a generalized frequency distribution,⁽³⁾ which is useful in understanding the dynamics and structure of the scatterer.

Inelastic differential scattering studies have been reported previously for room temperature water^(4,5,6,7,8) and D₂O.^(5,9,10) Ice has been studied⁽¹¹⁾ previously by upscattering of cold neutrons. The energy and momentum transfer range of the measurements reported here overlaps and extends to considerably higher transfer values

than all previously reported work except the room temperature water scattering reported in Reference 6 and 8, which was done with an electron linac source. These linac results however were not done with comparable energy resolution and do not show as much of the structure associated with the high energy intramolecular vibrations in the H₂O molecule. The present results were made with adequate energy resolution to resolve both the bending and stretching modes of the water molecule in the liquid and solid phases, as well as the hindered rotation band in the solid. High energy molecular vibrations in water have not been previously resolved in neutron scattering experiments.⁽¹²⁾ Interpretations of these compiled cross sections including the derivation of generalized frequency spectra will be published elsewhere.

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