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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1970

June 1970

AEC RESEARCH &
DEVELOPMENT REPORT

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Reactor Technology

PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1970

By

Staff of Battelle-Northwest
Program Leader: F. G. Dawson

June 1970

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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1970

FOREWORD

The Plutonium Utilization Program is conducted by the Pacific Northwest Laboratory for the USAEC. The objective of the Technical Activities Quarterly Report is to inform the scientific community in a timely manner of the technical progress made on the program. 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 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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December 1967, January, February 1968	BNWL-739
March, April, May 1968	BNWL-828
June, July, August 1968	BNWL-907
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PLUTONIUM UTILIZATION PROGRAM
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1.0 SUMMARY

REACTOR NEUTRONICS

The following computer codes have been documented and released to the Argonne Code Center.

- BMC-I: The Battelle Monte Carlo Code
- HRG 3: A code for Calculating the Slowing-Down Spectrum in the P1 or B1 Approximation
- BRT-I: Battelle-Revised-THERMOS-I
- DBUFIT-I: A Least Squares Analysis Code for Nuclear Burnup Data.

An abstract of each of these codes can be found in Section 6.0, Publications and Presentations.

A detailed theoretical analysis is being made of the PRCF critical experiments conducted in the cooperative USAEC-CNEN program. Theoretical methods and calculational techniques are being evaluated by comparing calculated and measured reactivities and power distributions. In general, values of k_{eff} predicted via calculation are larger than measured values. Systematic differences are noted between calculated and measured power distributions. A simple modification in calculational technique significantly improves the power distribution correlation for the twelve configurations analyzed.

A new version of the nonlinear least-squares fitting program LEARN, called LEARN-I-C, has been written. The new version increases both reliability and speed in the solution of difficult problems.

Critical experiments were completed in the PRCF in an H₂O moderated lattice (1.06 in. square pitch) of UO₂-4 wt% PuO₂ fuel rods.

The radio chemical and mass spectrometric analysis of fuel samples taken from the PRTR Batch Core has been completed.

Gulf General Atomic, Inc., has completed the reanalysis of the plutonium lattice space dependent spectral measurements. A report (GA-10116) on the results of this research is being issued.

Subcritical measurements of critical numbers of rods for Al-Pu-H₂O and UO₂-PuO₂-H₂O systems made in the CAF are completed.

Interpretation of experimental data has been completed for the determination of k_{∞} for a uniform lattice of 0.5-in. diameter UO₂-2.0 wt% PuO₂ rods at 1.0-in. square pitch in light water. The discrepancy between theoretical prediction and the value for k_{∞} inferred from PCTR measurements, apparent in the preliminary analysis, has been resolved by a detailed re-evaluation of the experimental data. The previous discrepancy resulted from an accumulation of several small systematic errors which had been either neglected or improperly evaluated.

FUELS TECHNOLOGY

Ceramographic examination of cold-press-sinter pellet UO₂-PuO₂ rods irradiated in PRTR shows that the predicted and indicated fuel behavior agree. Core-clad reaction occurs in swage-compacted UO₂-PuO₂ rods irradiated in PRTR.

Re-evaluation of the data from the internal pressure monitoring experiments irradiated in PRTR based upon more accurate ¹⁴⁸Nd burnup results shows that the fission gas release fraction for the high and lower performance rods were 34 and 12%, respectively. The fission gas release versus volumetric average fuel rod temperature relationship was revised based upon more accurate ¹⁴⁸Nd burnup data.

Fabrication of instrumented experiments to monitor internal gas pressure and fuel temperature during irradiation is in progress.

Gamma dose rate and neutron yield measurements were made on BWR type fuel pins and assemblies at General Electric Company. Measured neutron yields were less than calculated.

Dissolution studies with unirradiated cold-press-sinter UO_2 -4 wt% PuO_2 pellet fuel indicate that extensive solid solution formation occurs during fabrication such that the pellets are soluble in nitric acid.

Detailed evaluation of the UO_2 -4 wt% PuO_2 fuel material being irradiated in the defect testing assembly shows that although complete homogenization of the UO_2 and PuO_2 did not occur during fabrication of the pellets, nonhomogeneous solid solution was formed. Activity release during irradiation of an intentionally defected UO_2 - PuO_2 rod operating at a peak power of 10 kW/ft is well within acceptable limits.

REACTOR SAFETY

SPERT tests to study PuO_2 particle size effects in UO_2 - PuO_2 fuels show that transient behavior is affected by the presence of PuO_2 agglomerates. TREAT tests indicate that the cladding failure threshold for pellet and vibrationally compacted fuels is approximately the same.

2.0 REACTOR NEUTRONICS

CALCULATIONS OF POWER DISTRIBUTIONS AND REACTIVITIES

V. O. Uotinen, G. L. Gelhaus, U. P. Jenquin and C. R. Gordon

Introduction

A detailed analysis is in progress of the critical experiments conducted in the PRCF under a cooperative program⁽¹⁾ between the USAEC and the Italian CNEN. The purpose of this analysis is to evaluate calculational methods and models by comparing calculated power distributions and reactivities with those measured in the USAEC-CNEN program. The experimental program comprised a large number of lattice configurations using 2.35 wt% enriched UO_2 rods and $\text{PuO}_2\text{-UO}_2$ rods of several enrichments. The configurations ranged from simple uniform lattice arrays to arrays which simulated boiling water reactor fuel bundles with rods of lower enrichments on the edges and corners to reduce power peaking.

The analysis thus far has been applied to arrays in which a single type of fuel rod was used. This study comprises a total of 12 loadings, six configurations for each of two fuel types, 2.35 wt% enriched UO_2 and 2 wt% $\text{PuO}_2\text{-UO}_2$ (8% ^{240}Pu) rods. The six configurations were (1) a regular uniform loading of rods; (2) the same loading but with a water hole in the center (i.e., the central fuel rod was removed); (3) water slab (a row of fuel rods removed); (4) water cross; (5) a 7 × 7 rod array surrounded by water slots; (6) a similar 9 × 9 rod array.

In these twelve experiments, spatial power distributions were measured by gamma-scanning selected fuel rods. The k_{eff} for an infinitely-reflected array was also determined for each case. The main interest in the power distribution measurements was in the rod-to-rod distribution, especially the effects of water slots on the power peaking.

The series of experiments, ranging from the simplest (regular) array to one simulating a 7×7 or 9×9 bundle, provides a systematic test for the evaluation of calculational methods.

Calculations

In the analysis of H_2O -moderated and reflected experiments (especially clean critical experiments) one generally assumes the reactor is composed of two regions: core and reflector. Few-group cross sections are calculated, assuming that an infinite medium spectrum applies in each region. The few-group cross sections are then used in a diffusion theory calculation of k_{eff} .

In an earlier study⁽²⁾ it was pointed out that such a two-region, infinite medium model generally does not predict the power distribution well, although it may yield a satisfactory value for k_{eff} . In general, this method shows a pronounced trend, such that if the power distribution is normalized at the center of the core, the power near the core-reflector interface is consistently under-estimated. A simple modification, which resulted in considerably improved correlations, was reported in Reference 2. This simple modification consisted in introducing an extra reflector region (one lattice unit thick, adjacent to the core) which is represented by cross sections averaged over a spectrum characteristic of the core.

The multigroup transport theory codes HRG and Battelle-Revised - THERMOS were used to generate four-group cross sections for core and reflector regions. These cross sections were used in the two-dimensional diffusion theory code 2DB in an x-y calculation of power distributions and k_{eff} .

Four mesh points per cell were used in the 2DB calculations. This mesh description was carried out two lattice units

into the reflector; then the mesh points were more widely spaced. An axial buckling of 8.9 m^{-2} was used consistently.

In our current analysis we have compared three variations of our calculational model:

Model 1. The usual two-region, infinite-medium model (described in previous section);

Model 2. A simple modification consisting of an additional reflector region whose cross sections are obtained from cell calculations performed for the core (described in the previous section). In this model all water gaps also contained these modified reflector cross sections.

Model 3. A more detailed representation of the differences in spectrum in successive rows of fuel and water. This was accomplished with THERMOS calculations in slab geometry, with appropriate homogenized regions of core, reflector and gaps; editing was done over the proper spatial points to obtain average cross sections for each "row" of fuel and water. For analysis of the UO_2 loadings, five sets of core cross sections were used to represent fuel rods in various locations, and four sets of cross sections were used to represent water. For analysis of the UO_2 - PuO_2 loadings, three sets of core cross sections and three sets of water cross sections were used.

Results

Power Distributions

Power distributions calculated using the three models described above were compared with measured distributions.

The trend that is so evident in the regular lattices when using Model 1 (i.e., calculated power consistently underestimated near core-reflector boundary) was significantly reduced when the modified models were used. The simple

modification used in Model 2 gives a better correlation than the more refined modification used in Model 3. This is evident especially in the case of the mixed oxide loading.

The lattice position that one chooses for a normalization point is rather arbitrary, yet this choice can affect the trends that one sees as well as the overall goodness or badness of the correlation. The center of the core is one likely normalization point. However, if this point is chosen, then one is basing his whole correlation on the accuracy of that one measurement (since there is only one rod in the center). A better choice might be a location away from the center and also away from water boundaries, a location which would permit four or eight symmetrical rods to be measured, the average of these measurements then providing a more reliable normalization point. However, multiple symmetrical rods were not measured in every case. Furthermore, no matter what point one chooses, there are always nonuniformities in the fuel rods, in the lattice plates, and bowing of fuel rods, etc., which introduce unknown errors into the normalization.

To eliminate this arbitrariness, and to provide a meaningful and consistent criterion for comparison of methods, we chose to represent the goodness of each correlation by a standard deviation, σ , defined by⁽¹⁾

$$\sigma = \sqrt{\sum_i^N (\bar{\delta} - \delta_i)^2 / (N - 1)}$$

where N is the number of rods measured,

$$\delta_i = \frac{P_{\text{calc}} - P_{\text{meas}}}{P_{\text{meas}}} \text{ for the } i^{\text{th}} \text{ rod,}$$

and
$$\bar{\delta} = \sum_i^N \delta_i / N.$$

The definition of σ implies an "effective" normalization such that the average fractional deviation, $\bar{\delta}$, is zero. This definition thus makes σ independent of the particular choice of normalization, and provides us a meaningful, consistent measure for purposes of comparing methods.

The σ 's for the various cases are given in Table 2.1. We can make the following general observations:

- In every case σ was significantly reduced when modifications were made to the simple two-region, infinite medium model (Model 1).
- In most cases, Model 2 gives the best correlation.

The significant improvement, and the goodness of the correlations obtained with the simple modification (Model 2), as well as its simplicity, make this model attractive for calculating power distributions in H_2O cores.

Calculation of k_{eff}

The modifications that were introduced to the two-region, infinite-medium model to improve power distribution correlations resulted in increases in calculated values of k_{eff} . This is consistent with comparisons between transport and diffusion theory results⁽³⁾ which indicate that transport theory gives higher values of k_{eff} . That is, when one represents the core-reflector boundary with a better model (be it transport theory or a modification to diffusion theory) this results in higher values of k_{eff} . The calculated values of k_{eff} are listed in Table 2.2.

For the UO_2 loadings, the k_{eff} calculated using Model 1 were consistently low, with discrepancies ranging from 0.24 to 1.7%. Best agreement between measured and calculated k_{eff} values was obtained using Model 2, with discrepancies ranging from +0.34% to -0.17%. Model 3 gave consistently high values of k_{eff} (by 1.0% to 1.7%).

TABLE 2.1. Standard Deviation (%) in Power Distributions

	UO_2			PuO_2-UO_2		
	Model 1	Model 2	Model 3	Model 1	Model 2	Model 3
Regular	2.09	1.18	1.05	3.37	1.86	2.17
H ₂ O-hole	1.48	0.98	1.14	2.48	1.78	-
H ₂ O Slab	1.95	1.48	1.56	1.60	1.38	1.13
H ₂ O Cross	1.57	1.37	1.21	2.13	1.40	-
7 x 7	1.96	1.72	2.16	2.26	1.69	1.78
9 x 9	2.91	2.67	3.12	2.29	1.90	-

TABLE 2.2. Values of k_{eff}

	UO_2				PuO_2-UO_2			
	Exp.	Model 1	Model 2	Model 3	Exp.	Model 1	Model 2	Model 3
Regular	1.0032	1.0008	1.0051	1.0164	1.0006	0.9960	1.0080	1.0009
H ₂ O Hole	1.0025	1.0000	1.0046	1.0161	1.0020	0.9973	1.0096	-
H ₂ O Slab	1.0018	0.9957	1.0044	1.0162	1.0068	0.9982	1.0182	1.0063
H ₂ O Cross	1.0010	0.9924	1.0039	1.0161	1.0054	0.9956	1.0231	-
7 x 7	1.0010	0.9888	1.0044	1.0175	1.0038	0.9867	1.0220	1.0007
9 x 9	1.0027	0.9858	1.0010	1.0133	1.0078	0.9928	1.0245	-

For the $\text{PuO}_2\text{-UO}_2$ loadings, the k_{eff} calculated using Model 1 were consistently low, with the discrepancies ranging from 0.5 to 1.7%. The k_{eff} calculated using Model 2 were consistently high, with discrepancies ranging from 0.7 to 1.8%; Model 3 gave the closest agreement with measured k_{eff} values.

Conclusions

The simple modification incorporated in Model 2 of our analysis significantly improved power distribution correlations for the twelve configurations that were analyzed.

This improvement, together with the simplicity of this method, makes this method attractive for calculating power distributions in H_2O cores. The simplicity of the model comes about because cross sections for the modified water regions (water gaps and the reflector adjacent to the core) are obtained directly from cell calculations for the core. No additional calculations are necessary.

The method will next be used to calculate power distributions in more complex loadings which contain fuel rods of several enrichments.

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1. P. Loizzo, et al. Experimental and Calculated Results for UO_2 and $\text{UO}_2\text{-PuO}_2$ Fueled H_2O -Moderated Loadings, To be published, Battelle-Northwest.
2. V. O. Uotinen, G. L. Gelhaus, S. R. Dwivedi. "Analytical Correlations of Measured Power Distributions," Plutonium Utilization Program Technical Activities Report, March, April, May 1969, BNWL-1106, Battelle-Northwest. July 1969.
3. S. R. Dwivedi. "Multiplication Dependence on Energy and Angular Detail in Transport Theory Calculations," Plutonium Utilization Program Technical Activities Quarterly Report, June, July, August 1969, BNWL-1224, Battelle-Northwest October 1969.

NONLINEAR LEAST-SQUARES DATA ANALYSES

D. A. Kottwitz

A new version of the nonlinear least-squares fitting program LEARN,⁽¹⁾ called LEARN-I-C, has been written and is ready for general use. This program supercedes versions LEARN-I-A⁽²⁾ and LEARN-I-B.⁽³⁾ This version includes modifications which increase both reliability and speed for difficult problems. In addition, three new options concerning the plotting of graphical output on the CALCOMP plotter are available to the user at execution time:

- 1) Optional number of times the graph is traced,
- 2) Optional distance between the initial pen position and the edge of the paper,
- 3) Optional spacing between successive origins in case of multiple plots during a single run.

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1. G. D. Seybold. User's AID: Programs LEARN and LIKELY, BNWL-1057. Battelle-Northwest. May 1969.
2. D. A. Kottwitz. "Constraints in Nonlinear Least-Squares Program LEARN," Reactor Physics Quarterly Report, July, August, September, 1969, BNWL-1240. Battelle-Northwest. November 1969.
3. D. A. Kottwitz. "Nonlinear Least-Squares Data Analyses," Plutonium Utilization Program Technical Activities Quarterly Report, September, October and November 1969. Battelle-Northwest. January 1970.

CRITICAL EXPERIMENTS IN UO_2 -4 WT% PuO_2 FUELED- H_2O MODERATED ASSEMBLIES

W. P. Stinson

Critical experiments were conducted in the Plutonium Recycle Critical Facility (PRCF) to obtain data for verifying the accuracy of neutronic design methods. The fuel rods contained UO_2 -4 wt% PuO_2 with 18% ^{240}Pu ⁽¹⁾ and the moderator was H_2O .

The core consisted of the fuel (1/2 in. diameter rods) in a 1.06 in. square lattice pitch loaded to form a cylinder with a 36 in. height and a radius equivalent to 126 fuel rods when critical. The measurements conducted on this core were:

- Critical size
- Kinetic parameters (β/ℓ)
- Flux distributions
- Void coefficient
- Approach-to-critical.

Approach-to-critical data were taken as a check on the techniques used in similar type measurements made in the Critical Approach Facility (CAF). The critical loading inferred from the subcritical approach was 126.70 ± 0.04 rods which agrees well with the actual critical loading of 126.44 ± 0.07 rods.

The kinetics measurements were made by recording and analyzing the reactor noise. The preliminary value of β/ℓ obtained is $93 \pm 2 \text{ sec}^{-1}$. The flux distribution was measured by scanning the fuel rods for fission product decay gamma rays. Empty thin-wall aluminum tubes were placed in interstitial positions in the central part of the core to obtain data from which the void coefficient can be determined. Analysis of the data is continuing to obtain final values.

Reference

1. L. C. Schmid et al. "Fuel Description," Reactor Physics Data for the Utilization of Plutonium in Thermal Reactors, BNWL-801, p. A2. Battelle-Northwest. May 1968.

BURNUP DATA FROM THE PRTR BATCH CORE EXPERIMENT

D. E. Christensen and R. P. Matsen

Fourteen rods of the PRTR Batch Core Experiment⁽¹⁾ were selected for destructive analysis. Thirty five samples were

cut from these fourteen rods. A gross gamma scan was made of each rod to measure the fission profile of the rod in order to select appropriate samples. Gamma ray spectra were then taken at the positions of the individual samples. The location and irradiation history of the samples are given in Table 2.3. The initial composition of the various plutonium fuel lots are shown in Table 2.4. Within an element, positions j, n, and r are from the 12-rod or exterior ring, positions c and f are from the 6-rod or middle ring, and position a is the center rod of the nineteen rod cluster. The location of each sample with respect to the top of the fuel (including the UO_2 pellet) is also indicated in Table 2.3.

The destructive analysis data have been processed through the data analysis code, ISODIL. Preliminary results indicates the maximum exposure is 9630 MWd/MTM for the sample from the exterior rod in Element 6063 which was located in the first ring of the core.

Additional analysis of the data will be made to derive ratios of effective cross sections. The isotopic data will be used in conjunction with the gamma ray spectral data in studies which seek to determine fuel burnup nondestructively. All the data will be used as a basis for verifying the accuracy of neutron design methods in predicting reactor burnup.

Reference

1. J. R. Worden, W. L. Purcell and L. C. Schmid. Physics Experiment-High Power Density Core of the PRTR, BNWL-221, Battelle-Northwest. 1966.

GULF GENERAL ATOMIC SPECTRAL DATA

The reanalysis of the plutonium lattice space dependent spectral measurements conducted by Gulf General Atomic, Inc.

TABLE 2.3. Batch Core Samples

Fuel Element No.	Position in Reactor (Ring)	Rod No.	Position in Element	Pu Fuel Lot	Irradiation Terminated on	Removal During	Sample	Position* in Rod, in.	Approximate Date that Samples Analysed	Exposure, Mwd/MTM
6065	3	FS12	r	62-23	9-10-67	Interim Test 1	1	6.5	9-10-68	1540
							2A	10.4	9-10-68	1770
							3A	29.4	9-10-68	2900
							4	41.3	9-10-68	2490
							5	47.5	9-10-68	2140
→										
6066	3	FS16	r	62-23	1-28-68	Interim Test 2	1	6.5	6-15-69	3360
							2	9.6	6-15-69	3960
							3	28.9	6-15-69	5970
							4	43.6	6-15-69	5290
							5	48.1	6-15-69	4650
F079			c	62-22	1-28-68		1	6.5	6-15-69	2120
							2	10.4	6-15-69	2640
							3	30.9	6-15-69	3850
							5	47.3	6-15-69	3050
							1	6.5	6-15-69	1810
FE77			a	62-23	1-28-68		3	30.3	6-15-69	3270
							5	47.6	6-15-69	2700

* Length of Fuel = 58.675 ± 0.125 in. which includes 57.837 ± 0.125 in. of UO_2 - PuO_2 and a UO_2 pellet on each end.

TABLE 2.3 (contd)

Fuel Element No.	Position in Reactor (Ring)	Rod No.	Position on Element	Pu Fuel Lot	Irradiation Terminated on	Removal During	Sample	Position in Rod, in.	Approximate Date that Samples Analyzed	Exposure, MTD/MTM
6063	1	FS01	n	62-23	6-1-68	Interim Test 3	3	28.9	6-15-69	8920
6067	3	FS23	r	62-23	6-1-68		1	6.5	6-15-69	4520
							2	9.7	6-15-69	5340
							3	29.1	6-15-69	8050
							4	43.5	6-15-69	7250
							5	48.5	6-15-69	6590
6058	5	FS08	j	62-23	6-1-68		3	29.9	6-15-69	7130
6516	7	FR64	j	62-6	6-1-68		3	30.5	6-15-69	5800
6520	1	FR78	n	62-18	7-14-68		1	30.4	4-1-70	9630
6065	3	FS07	j	62-23	7-14-68		1	30.4	4-1-70	9180
		F017	c	62-22	7-14-68		2	10.0	4-1-70	3920
							3	30.4	4-1-70	5840
		FE74	a	62-23	7-14-68		2	10.0	4-1-70	3290
							3	30.4	4-1-70	4860
6519	7	FN86	f	62-18	7-14-68		1	7.8	4-1-70	1940
							2	30.4	4-1-70	3300
		FE69	a	62-10	7-14-68		1	7.8	4-1-70	1710
							2	30.4	4-1-70	2880

TABLE 2.4. Initial Composition of Plutonium Fuel Lots

Fuel Lot	^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
62-23	0.026	90.526	8.229	1.139	0.080
62-22	0.016	91.119	8.085	0.739	0.041
62-6 } 62-10 } 62-18 }	Unknown	91.045	8.086	0.817	0.052

has been completed. A final report entitled Reanalysis of Plutonium Lattice Spectral Data, GA-10116, has been written and is being issued. The Introduction and Conclusions sections of this report are repeated here.

Introduction

"This report describes the research performed for Pacific Northwest Laboratory (PNL) through USAEC Contract AT(45-1)-1830 during the period November 1, 1969 through April 30, 1970. The research covers the reanalysis and documentation of the data measured at Gulf General Atomic for the U.S. Atomic Energy Commission under Contract AT(04-3)-167, Project Agreement No. 32. These data were time-of-flight thermal neutron spectra measured at various positions and a different temperatures in H_2O -moderated lattices of Pu-Al fuel. In this program significant distortions were found in the spectral data that were believed to be due to complex neutron emission time effects which could not be treated at the time with existing codes. In the program conducted for PNL, these effects have been corrected for as far as possible and the revised measured and calculated data have been listed for reference use.

"The experiments performed are described in detail in Reference 1; however, for the sake of completeness, a brief

description is given in Section 2 of this report. Section 3 describes the problems encountered in the experimental data and the treatment that has been given to them. We discuss the trends in the revised experimental data and their comparison to theory in Section 4. The measured and calculated values are listed in the appendices."

Conclusions

"From the present analysis of the thermal neutron spectra in Pu-Al lattices, several conclusions can be drawn:

1. A fairly good agreement (within 10%) exists between the calculated and the measured power distribution in the cell.
2. Relative temperature effects are well taken care of by the Haywood-II scattering kernel for H₂O.
3. Slowing down and thermalization processes are reasonably well understood and in the framework of the present analysis, no significant neutron emission time problems remain.

"However, this work leaves unanswered two questions:

1. The mechanism for the distinct shift of the 0.3 eV resonance flux dip towards low energies.
2. The lower effective temperature of the measured spectra with respect to calculations.

On the basis of the present information, it can be said that the second question casts some doubts about the ability of the Haywood kernel to describe accurately the energy exchanges taking place in strongly anisotropic situations. It is difficult to be definite on this point since it should be recalled that the calculations depend on various approximations concerning the description of the cell geometry and the neutron source and the angular representation of the neutron flux.

Finally, let us point out that aspects of the experiments which are not fully understood prevent evaluation of the quality of certain cross sections used in the computation, particularly those of ^{239}Pu ."

Pu-H₂O LATTICE SUBCRITICAL EXPERIMENTS

J. H. Lauby

Experiments were conducted in the Critical Approach Facility (CAF) to measure the critical mass of some plutonium-fueled, H₂O-moderated lattices. These lattices were repeats of some of those in which measurements of the critical numbers of rods were made for Al-Pu-H₂O and UO₂-PuO₂-H₂O systems.⁽¹⁾ The UO₂-PuO₂ fueled lattices were "remeasured" to determine the effect of ^{241}Pu decay on the number of rods for critical.⁽²⁾ The Al-Pu fueled lattices were remeasured to determine the combined critical mass effect of removing Lucite tubes which surrounded each fuel rod and ^{241}Pu decay. The reduction of the data collected in these experiments are completed, and the results are summarized in Table 2.5 with results obtained from the earlier measurements.⁽¹⁾ A destructive analysis of one of each of these fuel rods is also being made to determine isotopic compositions.

The new results for the Al-1.82 wt% fueled lattices show a decrease in the critical mass for the 0.75 and 0.85 lattices which are undermoderated and an increase in the critical mass for the 0.93 in. lattice which is over-moderated. Decay of ^{241}Pu would increase the critical number of rods for all lattices. Since the ^{241}Pu concentration originally in these rods was quite small the major perturbation is the removal of Lucite sleeves. If it is assumed that Lucite and H₂O are identical from a reactor physics standpoint, then the observed effects could be explained by assuming voids existed between the sleeves and the fuel rods. The new results for the

TABLE 2.5. Comparison of Measured Critical Masses

Fuel	Hexagonal Lattice Spacing, in.	Critical Number of Rods			Date, mo/yr ^(a)
		New Values	Date, mo/yr ^(a)	Previous Values	
Al-1.83 wt% Pu (5.58 wt% ²⁴⁰ Pu 0.44 wt% ²⁴¹ Pu)	0.75	541 ± 1	1/70	563 ± 1	1/62
	0.85	486 ± 2	12/69	494 ± 1	2/62
	0.93	560 ± 2	1/70	547 ^(b)	--
Al-2.01 wt% Pu (16.46 wt% ²⁴⁰ Pu 2.31 wt% ²⁴¹ Pu)	0.75	861 ± 1	1/70	848 ± 1	9/63
	0.85	826 ± 1	2/70	786 ± 1	8/63
	0.93	1115 ± 5	1/70	1040 ± 2	1/66
UO ₂ -2 wt% PuO ₂ (23.5 wt% ²⁴⁰ Pu 4.08 wt% ²⁴¹ Pu)	0.93	308 ± 1	1/70	288 ± 1	7/65
UO ₂ -4 wt% PuO ₂ (18.10 wt% ²⁴⁰ Pu 5.08 wt% ²⁴¹ Pu)	0.93	180 ± 1	4/70	179 ± 1	12/68

(a) Month and year in which experiment was conducted.

(b) Value interpolated from curve of Critical Number of Rods Versus H/Pu Atom Ratio

A1-2 wt% Pu fueled lattices show that the critical number of rods increased for all lattices. Therefore it appears that the ^{241}Pu decay is the major effect since it is expected that removing the Lucite sleeves would decrease the critical mass for the 0.75 and 0.85 in. lattices.

The new values for the $\text{UO}_2\text{-PuO}_2$ systems illustrate the effect of ^{241}Pu decay. A four and one-half year decay period leads to a 6.9% increase (i.e. 20 rods) in the critical number of rods for the $\text{UO}_2\text{-2 wt% PuO}_2$ lattices.

The results of this study show that ^{241}Pu decay has a non-negligible effect on criticality. Repeating these measurements a couple of years hence would provide additional valuable data on the time dependent reactivity effect of ^{241}Pu decay to ^{241}Am .

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INTERPRETATION OF k_∞ FOR WATER MODERATED $\text{UO}_2\text{-PuO}_2$ LATTICES

D. F. Newman and C. R. Gordon

Introduction

The development of a water tank in the Physical Constants Testing Reactor (PCTR)⁽¹⁾ during FY 1968 enabled the performance of light water moderated lattice experiments in the PCTR. Measurements of k_∞ and other neutronic parameters were obtained for a uniform lattice of 0.5-in. diameter $\text{UO}_2\text{-2.0 wt% PuO}_2$ (7.74 wt% ^{240}Pu) rods at 1.0-in. square pitch in light water.⁽²⁾ Preliminary analysis of the experimental data indicated a 6% discrepancy between theoretical predictions and the

value for k_{∞} inferred from PCTR measurements.⁽³⁾ An extensive study of the theoretical models, analytical techniques, and experimental measurements used in the correlation was undertaken. The apparent discrepancy has been resolved by a detailed reevaluation of the experimental data.

Activation Foil Perturbation Corrections

Foil activation techniques are used in the determination of the relative reaction rates in all central cell constituents and the standard absorber (usually copper) inserted in the center of the PCTR. The presence of foil materials in the lattice perturbs the neutron flux in the vicinity of each foil, such that the induced activity is not truly representative of the flux in the surrounding unperturbed media.

Two-dimensional transport theory calculations in two energy groups were performed using DOTSN,⁽⁴⁾ explicitly including foils in the lattice cell geometry used to measure activation rates. Thermal group constants were calculated using BATTELLE-REVISED-THERMOS Code;⁽⁵⁾ epithermal group constants were calculated using the HRG⁽¹⁶⁾ Code. Calculated two-group neutron flux perturbations in the foil regions of the R-Z geometry lattice cell were used to correct the measured sub-cadmium and epithermal activation rates to represent the two group activation rates in the unperturbed media.

Axial position-dependent fluxes calculated for the fuel region relative to the unperturbed flux in the fuel are shown in Figure 2.1, with bare foils inserted in the fuel rod. Similar calculated results, with cadmium covered foils inserted in the fuel, are shown in Figure 2.2. Additional calculations, with a bare 0.005-in. thick copper sector foil in the water, show a 0.6% thermal flux dip and a 0.2% epithermal flux dip in the foil when compared with unperturbed fluxes. These

FIGURE 2.1. Neutron Flux Ratios for Bare Foils Irradiated in the Fuel

FIGURE 2.2. Neutron Flux Ratios for Cadmium Covered Foils Irradiated in the Fuel

foil perturbation correction factors were applied to the measured bare and cadmium covered foil activation rates to determine the reaction rates in the unperturbed central cell constituents.

PCTR Water Tank Calculational Model

As a means of (1) predicting neutron spectrum equilibration and central cell end effects, (2) identifying possible experimental problem areas, and (3) correlating experimental results, a PCTR calculational model can be a useful analytical tool. A two dimensional (R-Z) diffusion theory model of the PCTR, with the water tank installed in the central test region, was obtained by radial cylindrification of equivalent volumes for all PCTR zones. Using the 2 DB Code,⁽⁷⁾ calculations were performed for the PCTR water tank experiment with UO_2 -2.0 wt% PuO_2 rods.^(2,3) The neutron spectrum in the vicinity of the central cell was matched to the fundamental mode value, as shown in Figure 2.3; the calculated radial spectral variation was in agreement with gold cadmium ratio measurements in the water tank. Comparison of the calculated axial variation of the thermal flux on the outer surface of the central cell fuel clad with measured subcadmium activation rates of 0.020-in. diameter gold pins on the clad surface, is shown in Figure 2.4. The good agreement in the thermal flux peaking in the ends of the lucite and polyethylene central cell containers builds confidence in the model. Thermal flux peaking in the ends of the central cell fuel rods was calculated to increase the axially averaged thermal flux in the fuel 1.9% over the value calculated at the reactor centerline. Replacing the plastic container ends with aluminum removed most of the moderating effect from the end cap planes, and reduced the axially averaged thermal flux in the fuel to a factor 1.006 times the value calculated at the reactor centerline.

FIGURE 2.3. Neutron Spectrum Equilibration in the PCTR Water Tank

FIGURE 2.4. Axial Variation of Thermal Flux on the Outer Surface of the Fuel Clad

As a result of these studies, containers with aluminum ends have been fabricated for use with this water tank lattice. The null reactivity technique, using boric acid in the water at a concentration adjusted to give k_{∞} of unity, will be used for the next set of experiments in this lattice in the PCTR.

Evaluation of k_{∞}

The value for k_{∞} was determined from the experimental reactivity worth data and relative reaction rates in the unpoisoned lattice using the two group expression:⁽⁸⁾

$$k_{\infty} = 1 + \left(- \frac{\Delta \rho_{\text{cell}}}{\Delta \rho_{\text{Cu}}} \right) \frac{(\Sigma \phi V)_2^{\text{Cu}}}{(\Sigma \phi V)_2^{\text{Cell}}} \left[1 + \frac{(\phi_1^+ \Sigma_1 \phi_1)^{\text{Cu}}}{(\phi_2^+ \Sigma_2 \phi_2)^{\text{Cu}}} \right] \\ - \frac{(1+\tau B^2)(1+L^2 B^2)(1-p)L^2 B^2}{1 + (1-p)L^2 B^2} + \frac{\tau B^2 (1+L^2 B^2) [\tau B^2 - \eta_1 f_1 (1-p)]}{1 + \tau B^2 - \eta_1 f_1 (1-p)}$$

The k_{∞} defined in this expression is consistent with the two group fundamental mode calculation:

$$k_{\infty} = \frac{\text{Total Neutron Production}}{\text{Total Neutron Absorption}} = \frac{\nu_1 \Sigma_1^f \phi_1' + \nu_2 \Sigma_2^f \phi_2'}{\Sigma_1^a \phi_1' + \Sigma_2^a \phi_2'}$$

where ϕ_1' and ϕ_2' are equilibrium fluxes.

All measured and calculated quantities used in the evaluation of k_{∞} are listed in Table 2.6. Using these values, the inferred value of k_{∞} is:

$$k_{\infty} = 1 + (0.306 \pm 0.011) - (0.006 \pm 0.001) + (0.071 \pm 0.016) \\ = 1.371 \pm 0.019$$

TABLE 2.6. Quantities Used in the Evaluation of k_{∞} Measured Quantities

$$\left(- \frac{\Delta \rho^{\text{cell}}}{\Delta \rho^{\text{Cu}}} \right) = \frac{85.819 \pm 0.057\%}{2.598 \pm 0.056\%} = 33.03 \pm 0.71$$

$$\frac{(\Sigma \phi V)_2^{\text{Cu}}}{\left(\phi_o \right)_{\text{column}}^{\text{thermal}}} = 13.72 \pm 0.13$$

$$\frac{(\Sigma_1 \phi_1)^{\text{Cu}}}{(\Sigma_2 \phi_2)^{\text{Cu}}} = 0.097 \pm 0.001$$

$$\frac{(\Sigma \phi V)_2^{\text{Cell}}}{\left(\phi_o \right)_{\text{column}}^{\text{thermal}}} = 1585 \pm 40$$

Calculated Quantities

$\frac{\phi_1^+}{\phi_2^+}$	= 0.7115	2DB PCTR Model	
τ	= 30.74 cm ²	HRG	
p	= 0.8326	HRG	
$\eta_1 f_1$	= 0.6739	HRG	
L^2	= 2.090 cm ²	THERMOS	
$\eta_2 f_2$	= 1.507	THERMOS	
B^2	= 0.01157 cm ⁻²	HFN ⁽⁹⁾	} Fundamental Mode (Just Critical Bare Sphere)
$\frac{\phi_1'}{\phi_2'}$	= 2.748	HFN ⁽⁹⁾	
$\frac{\phi_1}{\phi_2}$	= 2.734	2 DB PCTR Model	

Conclusion

The previously reported value for $k_{\infty} - 1$, determined by the unpoisoned technique, was 0.452 ± 0.015 . The reevaluated $k_{\infty} - 1$ value of 0.371 ± 0.019 results from an accumulation of several small negative systematic errors which previously had been either neglected or improperly evaluated. Corrections for central cell end cap effects, thermal flux peaking in the ends of the plastic containers, activation foil perturbations, and the analytical model used to infer k_{∞} resulted in reductions of 7%, 2%, 6%, and 2%, respectively, of the previously reported value for $k_{\infty} - 1$. The calculated values for $k_{\infty} = 1.3646$ from THERMOS-HRG-HFN⁽⁹⁾ and $k_{\infty} = 1.3645$ from THERMOS-EGGNIT⁽¹⁰⁾-HFN are in excellent agreement with the value inferred from the experiment.

A considerable improvement in the precision of the k_{∞} measurement is predicted if the null reactivity technique is used, since the relatively large epithermal leakage correction term in the unpoisoned k_{∞} analytical model is eliminated. The uncertainty in k_{∞} for the UO_2 -2.0 wt% PuO_2 water lattice poisoned to null reactivity is estimated at ± 0.003 in k_{∞} . An experiment to test this prediction of improved precision is planned for the near future.

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3.0 FUELS TECHNOLOGY

PRTR FUEL EVALUATIONS

D. G. Carter and M. D. Freshley

With the shutdown of the PRTR, the irradiation performance of selected rods irradiated in the reactor is being evaluated by means of destructive postirradiation examination. In order to report the postirradiation evaluation results more logically, the PRTR fuel evaluations are divided into three groups as follows:

- High power density fuel evaluation.
- Extended burnup fuel evaluation.
- Internal pressure monitoring experiment evaluation.

The postirradiation examination status of each group is reported as follows:

High Power Density Fuel Evaluation

Postirradiation examination of selected PRTR High Power Density Fuel rods which were irradiated as part of the Batch Core Experiment (BCE) is continuing and the current status of the examination is summarized in Table 3.1. Most of the 66 High Power Density 19-rod cluster fuel elements irradiated in the BCE were vibrationally compacted powder fuel, although pellet fuel fabricated by both the hot-press and the cold-press-sinter processes was included.

Ceramographic examination of two Zircaloy-clad rods containing cold-press-sinter UO_2 - PuO_2 pellet fuel is in progress. The two rods (A-102 and A-103) were irradiated under approximately the same conditions in PRTR as summarized in Table 3.2. Operating conditions during irradiation, as inferred from the fuel structures formed, agree with the predicted conditions for cold-press-sinter pellet fuel operating at these power generations (Figure 3.1). Examination of transverse sections from the peak power locations

TABLE 3.1. Summary of PRTR High Power Density Fuel Rod Postirradiation Examination Status

Element Number	Rod No.	Fuel Type	Estimated Peak Burnup, MWd/MTM	Gamma Scan	Profilometer Scan	Fission Gas Analysis	Burnup Analysis	Ceramography
6065	FE-74	UO ₂ -2 wt% PuO ₂ (VP) ↓	4,863*	X	Z	X	X	Y
	FO-17		5,837*	X	Z	X	X	Y
	FS-07		9,176*	X	Y	X	X	Y
6519	FR-72		5,700	X	Y	X	Y	Y
	FN-86		3,297*	X	Z	X	X	Y
	FE-69		2,884*	X	Z	X	X	Y
6520	FR-78		9,630*	X	Y	X	X	Y
6700	A-14		9,600	X	Y	X	Y	Y
	A-103		9,600	Z	Y	X	Y	X
	A-102		9,600	X	Y	X	Y	X
6701	A-23		3,800	X	Y	X	Y	Y
	A-12		3,800	Z	Y	X	Y	Y

(VP) = Vibrationally compacted, pneumatically impacted fuel rod.

(CPS) = Cold-pressed-sintered pellet fuel rod.

(HP) = Hot-pressed pellet fuel rod.

X = Examination completed.

Y = Examination in progress.

Z = Examination not scheduled.

* = Burnup values determined by ¹⁴⁸Nd analysis.

1769

Rod A-102

1768

F-1982

 α - Autoradiographs β - γ Autoradiographs

Photomicrographs

1773

Rod A-103

1772

F-1896

FIGURE 3.1. Transverse Sections from Cold-Press-Sinter Pellet $\text{UO}_2\text{-PuO}_2$ Fuel Rods
Irradiated in PRTR at a Peak Power of ~ 16 kW/ft. (4.7X)

TABLE 3.2. Summary of Postirradiation Examination Results for Two Cold-Press-Sinter $\text{UO}_2\text{-PuO}_2$ Fuel Rods Irradiated in PRTR

Rod No.	Estimated Peak Power, kW/ft	Estimated Peak Burnup, MWd/MTM	Estimated Max. Fuel Temp, °C	Columnar Grain Growth Radius, % r	Fission Gas Release, %
A-102	16.0	3650	~1900	37	27.5
A-103	16.5	3650	~1980	42	27.5

show that extensive pellet cracking has significantly reduced the relatively large 0.010-in. as-fabricated diametral gap on the 0.565-in. diameter rods. Fission gas release fractions agree with predicted values based upon the fractional release versus volumetric average fuel rod temperature relationship for oxide fuels.

Extended Burnup Fuel Evaluation

Postirradiation evaluation of selected fuel rods irradiated to extended burnups in PRTR fringe positions continued. The status of the examination of these rods is summarized in Table 3.3.

Ceramographic examination of swage-compacted UO_2 rods irradiated in PRTR to estimated peak burnups of 14,500 MWd/MTM shows that the irradiation performance of these rods was satisfactory (Figure 3.2). Maximum fuel temperatures attained at the estimated peak power of 10 kW/ft were not sufficient to cause in-reactor sintering and grain growth to occur.

The fuel-cladding interface was examined on two Zircaloy-clad rods containing swage-compacted $\text{UO}_2\text{-PuO}_2$ fuel material that performed satisfactorily in PRTR. One rod (CN-5) containing heterogeneously enriched $\text{UO}_2\text{-0.5 wt% PuO}_2$ fuel was irradiated at an estimated peak power of 11 kW/ft to a peak burnup of 12,500 MWd/MTM. The other rod (CS-27) containing pneumatically impacted $\text{UO}_2\text{-1 wt% PuO}_2$ fuel material was irradiated at an

TABLE 3.3. Summary of the Postirradiation Examination Status for Fuel Rods Irradiated to Extended Burnup in PRTR Fringe Positions

Element Number	Rod No.	Fuel Type	Estimated Peak Burnup, MWD/MTM	Gamma Scan	Profile-meter Scan	Fission Gas Analysis	Burnup Analysis	Ceramo-graphy
1037	A-1330	UO ₂	14,500	X	Z	X	Y	X
	A-1377		---	X	Z	X	Y	X
	A-1398		---	X	Z	X	Y	X
	A-1325		---	Z	Z	X	Z	Y
5118	DB-63	UO ₂ -0.5 wt% PuO ₂	18,500	X	Z	X	Z	Y
	DA-48		18,500	Z	Z	X	Z	Y
	ZZ-17		---	X	Z	X	Y	Y
	ZZ-7		---	X	Z	X	Y	Y
5186	ZZ-12		12,500	X	Z	X	Y	Y
	CN-5		12,500	X	Z	Lost	Z	Y
	CN-7		12,500	Z	Z	X	Z	Z
	CM-85		12,500	Z	Z	X	Z	Z
5224	CS-27	UO ₂ -1 wt% PuO ₂	13,500	X	Z	X	Y	Y
	DF-80		11,500	X	Z	X	Y	Y
	6003	UO ₂ -2 wt% PuO ₂	13,000	X	Z	X	Y	Y
	6005		8,000	X	Z	X	X	Y
6521	No. 6	ThO ₂ -5 wt% PuO ₂	Low	X	Z	Lost	Z	Y

(VP) = Vibrationally compacted fuel rod

(SW) = Swage compacted fuel rod

(CPS) = Cold-pressed-sintered pellet fuel rod

(MM) = Heterogeneously enriched or incrementally loaded fuel material

(PI) = High-energy-rate pneumatically impacted fuel

X = Examination completed

Y = Examination in progress

Z = Examination not scheduled

O = Examination not started

FIGURE 3.2. Transverse Section from the Midplane of a Swage-Compacted UO_2 Fuel Rod Irradiated in PRTR at an Estimated Peak Power of 10 kW/ft to a Burnup of 14,500 MWd/MTM

estimated peak power of 14 kW/ft to a peak burnup of 13,500 MWd/MTM. Fuel-cladding reaction or bonding occurred in both rods (Figures 3.3 and 3.4). A large UO_2 particle caused an indent in the cladding in rod CN-5 during the swaging operation (Figure 3.3). The uniform oxide layer that formed on the inner surface of rod CS-27 is porous and has a friable appearance (Figure 3.4). This type of oxide layer has been observed on the inner surface of other PRTR rods. The particular conditions which are conducive to the formation of this type of oxide layer on the inner surface of Zircaloy cladding are unknown.

Internal Pressure Monitoring Experiment Evaluation

Vibrationally compacted UO_2 - PuO_2 fuel rods instrumented to monitor internal gas pressure during irradiation were irradiated in PRTR to determine the effects of fuel temperature and burnup on both sorbed and fission gas release. Following completion of the experiments in PRTR, the four instrumented rods were destructively examined, and the results of the post-irradiation evaluation are being related to the experimental results obtained during irradiation.

Analytically determined burnup values for the instrumented rods obtained by the ^{148}Nd isotopic dilution technique do not agree with the calculated burnup values based on flow and coolant temperature increase measurements. Because burnups as determined by ^{148}Nd and heavy element isotopic ratios agree to within 2%, the ^{148}Nd values are considered to be more accurate than the calculated values, and the data were adjusted in accordance with the ^{148}Nd burnup values. In addition to changing the reported burnup values for the instrumented rods, the estimated rod power generation and fission gas release fractions during irradiation are also affected accordingly. Revised data based upon the ^{148}Nd burnup results are summarized in Table 3.4.

F-1274

FIGURE 3.3. Fuel-Cladding Interface from a Swage-Compacted $\text{UO}_2\text{-0.5 wt\% PuO}_2$ PRTR Fuel Rod (CN-5). 250X

F-1269

FIGURE 3.4. Fuel-Cladding Interface from a Swage-Compacted $\text{UO}_2\text{-1 wt\% PuO}_2$ PRTR Fuel Rod (CS-27). 250X

TABLE 3.4. Operating Conditions for the Instrumented Rods in PRTR Based on ^{148}Nd Burnup Data

Rod No.	Peak Burnup, MWd/MTM Calculated	^{148}Nd	Nominal Peak Power, kW/ft	Nominal Max. Fuel Temp, $^{\circ}\text{C}$	Maximum Peak Power, kW/ft	Fission Gas Release, %
2	5,750	5,540	9.2	~ 1400	11.5	11.8
4	5,750	5,100	8.4	~ 1300	10.5	(a)
7	10,940	9,480	14.3	~ 2200	18.0	33.6
8	10,940	9,500	14.3	~ 2200	18.0	34.3

(a) Gas lost during sampling.

Fission gas release data for several high power density fuel rods irradiated in PRTR were re-evaluated based upon recent ^{148}Nd burnup results (Figure 3.5). The plot includes data obtained from postirradiation gas sampling of cold-press-sinter pellet $\text{UO}_2\text{-PuO}_2$ fuel and vibrationally compacted $\text{UO}_2\text{-PuO}_2$ fuel rods. The actual fission gas release values for the instrumented rods, as determined by both the rate of internal pressure increase during irradiation and postirradiation sampling, are consistent with the values obtained from other similar rods irradiated in PRTR. The plot of fission gas release versus volumetric average fuel rod temperature includes data from rods with widely varying irradiation histories. Peak rod power generations ranged from 8 to 27 kW/ft with significant fuel melting and peak burnups ranged from 200 to 9600 MWd/MTM . The curve in Figure 3.5 was fitted to the data using a least squares fit calculated at the 95% confidence level.

INSTRUMENTED FUEL TESTS

F. E. Panisko and M. D. Freshley

Instrumented experiments to monitor internal gas pressure buildup and fuel temperatures in mixed oxide fuels during

FIGURE 3.5. Fission Gas Release Versus Volumetric Average Fuel Temperature for $\text{UO}_2\text{-PuO}_2$ Fuel Rods Irradiated in PRTR

irradiation will be conducted in the ATR Water Loop and the ETR M-3 Loop, respectively.

Internal Pressure Monitoring Experiment

Fabrication of an experiment in which internal gas pressure will be monitored in mixed oxide fuel rods during irradiation at typical PWR operating conditions is in progress. Peak rod power generations of approximately 17 kW/ft with peak fuel temperatures of 2500 °C are anticipated to a goal peak burnup of above 40,000 MWd/MTM. Operating conditions include a coolant pressure of 2000 psi at a bulk coolant temperature of 400 °F.

The performance of different pellet fuel designs (solid, dished, and cored) will be evaluated and compared and internal gas pressure buildup as affected by different cladding materials (Zircaloy-4 and stainless steel) will be investigated (Figure 3.6). The possible effect of burnup rate will be evaluated by using different rod diameters and plutonium enrichments.

The instrumented assembly will contain five rods, each of which will be equipped with a null balance pressure transducer to measure internal gas pressure (Figure 3.7). A composite hafnium-stainless steel neutron absorber basket will enshroud the assembly to provide the desired flux level.

Fuel Temperature Monitoring Experiment

Design of an experiment in which fuel centerline temperatures will be measured in rods containing different mixed oxide pellet fuel designs is being developed. Fuel temperatures will be monitored during a programmed power increase, during subsequent steady state operation, and during a programmed power decrease to study and compare the thermal behavior and in-reactor grain growth kinetics of the

Composition and Dimensions

Number	Part	Material	OD, in.	ID, in.	Length, in.	Pellet Fuel Material	OD, in.	ID, in.	Length, in.
1	Fuel Rod	Zr-4 Clad	0.426	0.306	54.370	UO ₂ -PuO ₂	0.300	---	48
2	Fuel Rod	Zr-2 Clad	0.426	0.372	54.370	UO ₂ -PuO ₂	0.365	---	48
3	Fuel Rod	Zr-2 Clad	0.426	0.372	54.370	UO ₂ -PuO ₂	0.365	---	48
4	Fuel Rod	304 SS Clad	0.426	0.372	54.370	UO ₂ -PuO ₂	0.365	---	48
5	Fuel Rod	Zr-2 Clad	0.926	0.372	54.370	UO ₂ -PuO ₂	0.365	---	48
6	Support Rod	304 SS Clad	0.312	---	62.3				
7	Basket	Hf	1.572	1.462	59.0				
8	Basket	304 SS Clad	1.712	1.572	59.0				

FIGURE 3.6. Arrangement of Different Fuel Rod Types and Parts in the Internal Pressure Monitoring Experiment Element.

FIGURE 3.7. Schematic Diagram of the Instrumentation Associated with the Internal Pressure Monitoring Experiment.

different pellet fuel designs; namely, solid, dished, and cored. The instrumented assembly will be irradiated in the ETR M-3 loop facility.

HIGH EXPOSURE PLUTONIUM STUDIES

R. C. Smith, L. G. Faust, and H. H. Van Tuyl

Dose rate measurements were made on thermal reactor mixed oxide fuel pins at United Nuclear Corporation,⁽¹⁾ General Electric Company, and PNL. The pins were arranged in different arrays designed to simulate BWR fuel assemblies.

Gamma dose rates were measured at General Electric Company, Vallecitos on single pins and a BWR fuel assembly containing the 9/16 in. OD fuel pins arranged in a 9 × 9 pin array. The mixed oxide pins contained different PuO₂ enrichments. However, the ²⁴⁰Pu content of all the plutonium used was approximately 10 wt%.

Results of the gamma dose rate and neutron yield measurements obtained from the GE-EEI fuel are summarized in Tables 3.5 and 3.6. Gamma dose rate measurements were made on bare UO₂-PuO₂ fuel material of varying composition that would comprise a single pin, on a single pin clad with Zircaloy, and on pins arranged in a 9 × 9 fuel assembly.

The maximum calculated neutron dose from the full 81 pin fuel assembly would be about 2.2 mrem/hr at 2 ft. At 6 in. from the fuel assembly, the neutron dose would be between 6 and 7 mrem/hr.

Neutron yield measurements will be made on plutonium enriched PWR fuel now being manufactured at Westinghouse for irradiation in the San Onofre reactor as part of the joint Westinghouse/EEI program. The measurements are scheduled to be made in August or September before the fuel is charged into the reactor.

Reference

1. Plutonium Utilization Program Technical Activities Quarterly Report, December 1969, January, February 1970, BNWL-1358, pp. 3.1-3.3, Battelle-Northwest. April 1970.

TABLE 3.5. Measured Gamma Dose Rates from GE-EEI Fuels

Fuel Source	Dose Rate, mrad/hr ^(a)			
	Surface	0.75 in.	6 in.	24 in.
UO ₂ -1.6 wt% PuO ₂ fuel material	79.0			
UO ₂ -1.6 wt% PuO ₂ -UO ₂ fuel material shielded with 0.030 in. thick zircaloy cladding	4.5			
UO ₂ -5.5 wt% PuO ₂ fuel material	174.5			
UO ₂ -9 wt% PuO ₂ fuel material	237.0			
Fuel Assembly (9 × 9 array)		12	3.2	1.9

(a) The large gamma attenuation resulting from 0.030 in. thick Zircaloy cladding is significant

TABLE 3.6. Neutron Yields of GE-EEI Fuel

Fuel Source	Neutrons/Second		Percent Difference
	Measured	Calculated	Calc/Meas
UO ₂ -9 wt% PuO ₂ pins	7.16×10^6	7.60×10^6	+5.8
Fuel Assembly (9 × 9 array)	1.01×10^6	1.19×10^6	+15

PLUTONIUM FUEL DISSOLUTION STUDIES

R. E. Lerch and G. F. Schiefelbein

The purpose of this activity is to study the dissolution behavior of mixed-oxide thermal reactor fuels. It is expected that commercial processing of such fuels will be accomplished by the Purex process, an aqueous solvent extraction process. Successful application of this process requires a knowledge of the dissolubility and dissolution rate of thermal reactor mixed oxide fuels as affected by fuel fabrication and irradiation history.

Initial dissolubility studies have been completed on 0.5 in. OD dished and undished cold-press-sinter mixed oxide pellets. All pellets contained 4 wt% PuO_2 and were fabricated by mechanical mixing, pressing, and sintering in Ar-8% H_2 . Fabrication variables included pressing pressure (20, 25, 30, and 35 tons/in.²), sintering temperature (1650 to 1700 °C), and sintering time (6, 8, or 10 hr). Dissolubility of the pellets, each weighing about 16 g, was determined by exposing the pellets in 200 ml of boiling 12M nitric acid for one or, if necessary, two 6-hr periods. All pellet samples examined in this study were essentially dissolved in the first 6-hr treatment (residues remaining after the first 6-hr treatment ranged from 0.001 to 0.01 wt%). The results indicate that for the fabrication conditions used for these mixed oxide pellets (i.e., sintering temperature greater than 1650 °C, sintering time greater than 6 hr, pressing pressure greater than 20 tons/in.²) extensive UO_2 - PuO_2 solid solution formation occurs such that the pellets are readily soluble in nitric acid alone. These data are consistent with previous results⁽¹⁾ obtained on dissolution of unirradiated cold-press-sinter UO_2 -0.5 wt% PuO_2 pellets.

Dissolution rate studies have been performed on selected pellets from the above samples to determine if

preferential dissolution of the uranium takes place, an important consideration in assuring criticality safety. Analyses of the dissolution rate samples are incomplete at this time.

Dissolubility studies have also been initiated on selected samples from irradiated PRTR fuel rods. Primary emphasis is being focused on cold-press-sinter fuel samples containing from 0.5 to 2.0 wt% PuO_2 with vibrationally compacted fuels being of secondary interest. Dissolubility tests on fuel samples taken from different axial positions along the length of irradiated mixed oxide rods is currently underway. Samples from all available irradiated PRTR fuel rods have been obtained and stored for future studies.

Dissolubility data accumulated to date with cold-press-sinter mixed oxide fuels containing 0.5 and 2.0 wt% PuO_2 indicate that the fuels are nearly completely dissolved after one 6-hr treatment with boiling 12M nitric acid. Insoluble residues from 5-g samples exposed to 25 ml of the acid ranged from 0.002 to 0.03 wt%.

Reference

1. W. W. Schulz. Aqueous Decladding and Dissolution of Plutonium Recycle Test Reactor Fuels, Part 2: $\text{PuO}_2\text{-UO}_2$ Fuels, BNWL-204 Pt. 2, Battelle-Northwest. November 1966.

DEFECT TESTING

M. D. Freshley and J. K. Bahl*

Irradiation testing of defected and non-defected mixed oxide fuels is being performed in the ETR M-3 Loop facility. The defect testing program includes an investigation of the possible effects of the following parameters: (1) fuel form (small and large gap solid pellet, dished pellet, and cored pellet), (2) power generation (14 kW/ft to 22 kW/ft), and (3) burnup. In addition to evaluating fuel performance, comparative activity release data will be obtained for the different fuel designs while operating under different irradiation conditions.

The cold-press-sinter UO_2 -4 wt% PuO_2 pellet fuel used in the irradiation tests was fabricated by standard mixed oxide mechanical blending techniques. This includes mixing ceramic grade natural UO_2 and -325 mesh PuO_2 powders in a V-blender for 15 min followed by slugging, granulating, pressing, and sintering in an 8% H_2 -92% A gas mixture at 1650 °C for 10 hr. Sintered pellet densities were $93.5 \pm 1.5\%$.

Ceramographic examination of the UO_2 - PuO_2 pellet fuel material shows a single phase indicating that the PuO_2 has dissolved in the UO_2 (Figure 3.8). The average grain size is approximately 3 μm although there is a large variation in grain size. The porosity is considered typical of cold-press-sinter pellet fuel.

Alpha autoradiography of the UO_2 - PuO_2 pellet fuel shows a nonhomogeneous distribution of plutonium, indicating that complete homogenization of the UO_2 and PuO_2 did not occur during sintering (Figure 3.9). However, since the microstructure shows only single phase, the sintered pellet fuel material is a heterogeneous UO_2 - PuO_2 solid solution.

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250X

1000X

FIGURE 3.8. Typical Microstructure of Cold-Press-Sinter UO₂-4 wt% PuO₂ Pellet Fuel Material for the Defect Tests in the ETR M-3 Loop

30X

250X

FIGURE 3.9. Alpha Autoradiographs of Typical Cold-Press-Sinter UO_2 -4 wt% PuO_2 Pellet Fuel Material for the Defect Tests in the ETR M-3 Loop. Dark Areas Indicate High Plutonium Concentrations.

A quantitative evaluation of plutonium homogeneity was obtained from electron probe microanalysis of the sintered pellet fuel material. The plutonium concentration varied from almost zero to over 13% PuO_2 along the length of a 2000 μm scan (Figure 3.10). The scanning beam was 1 μm diameter and the plutonium and uranium distributions were recorded on a scale of 50 μm to the inch. The fact that the maximum plutonium concentration indicated by the 1 μm diameter beam was 13% PuO_2 whereas the maximum diameter of the as-mixed PuO_2 particles was 44 μm is further indication that nonhomogeneous UO_2 - PuO_2 solid solution formed during sintering. An X-ray photomicrograph obtained with the electron microprobe also illustrates typical plutonium distribution in the cold-pressed-sintered UO_2 - PuO_2 fuel material (Figure 3.11).

The defect test assembly containing twelve fueled Zircaloy-clad UO_2 - PuO_2 rods is being irradiated in the ETR M-3 Loop. No defected rods were included in the assembly for the first irradiation period during which time the element accumulated an average burnup of approximately 1600 MWd/MTM or a peak burnup of 2660 MWd/MTM. Visual examination of the element showed that the most noticeable effect of irradiation is cladding distortion in the form of ridging at pellet interfaces that is associated with core-clad mechanical interaction.

After the underwater handling and defecting techniques were approved, rod exchanges were made in the assembly, and one of the rods was defected by drilling a 0.020 in. diameter hole through the cladding at the peak power plane. The small amount of off-gas from the defected rod was easily handled with the underwater gas collection equipment without creating a radiation hazard.

Initial irradiation of the defected rod, which contains UO_2 -4 wt% PuO_2 solid pellet fuel assembled with a cold

FIGURE 3.10. Electron Probe Scan of a Mechanically Mixed Cold-Press-Sinter UO₂-4 wt% PuO₂ Pellet Showing the Uranium and Plutonium Concentrations. Beam Diameter was 1 μm .

300X

FIGURE 3.11. X-ray Photomicrograph of Cold-Press-Sinter UO_2 -4 wt% PuO_2 Pellet Fuel Material. Dot Density is Related to PuO_2 Concentration.

diametral gap of 0.010 inch, is in progress. The peak power attained thus far in the irradiation period is approximately 10 kW/ft. Steady state activity release levels within the loop system are well within acceptable limits, and there have been no burst activity releases associated with reactor power cycles. The peak power will be increased to the desired value of approximately 14.5 kW/ft at the defect for most of the operating period. It will then be increased to about 23 kW/ft toward the latter part of the irradiation period to study the effects of increased power on activity release. Different power generations can be achieved by rotating the element in the flux gradient that exists across the loop facility. The irradiation conditions for the defect test assembly operation in the ETR M-3 loop with a peak rod power generation of 23 kW/ft are summarized in Table 3.7.

TABLE 3.7. Operating Conditions for the Defect Test Assembly in the ETR M-3 Loop

Peak Linear Rod Power Generation	~23 kW/ft
Average Linear Rod Power Generation	~14 kW/ft
Peak Surface Heat Flux	546,000 Btu/hr-ft ²
Average Surface Heat Flux	329,000 Btu/hr-ft ²
Calculated Total Power Generation	410 kW
Heat Transfer Film Coefficient	5169 Btu/hr-ft ² -°F
Maximum Cladding Surface Temperature	317 °C
Maximum Cladding Temperature	410 °C
Maximum Fuel Surface Temperature	630 °C
Maximum Fuel Temperature:	
Solid Pellet, 0.006 in. gap	~2500 °C
Solid Pellet, 0.010 in. gap	~2650 °C
Cored Pellet, 0.006 in. gap	~2100 °C
Loop Flow Rate	150 gal/min
Loop Pressure	2000 psi
Coolant pH	9.8 to 10.2
Coolant Oxygen Concentration	<0.05 ppm
Saturation Temperature of Loop	635 °F
Coolant Velocity Over the Fuel	11.85 ft/sec
Inlet Water Temperature	480 °F
Outlet Water Temperature	502 °F

4.0 REACTOR SAFETY

TRANSIENT TESTING

M. D. Freshley, J. K. Bahl*, E. A. Aitken**, and
R. L. Johnson***

A knowledge of the behavior of reactor fuels when subjected to high energy, short duration power excursions is essential to permit a complete evaluation of fuel performance under accident conditions as related to reactor safety. Irradiations in SPERT are being used to investigate the transient behavior of oxide fuels to determine whether special considerations associated with adding PuO_2 enrichment to UO_2 fuels effects transient behavior. Irradiations were also conducted in TREAT to compare the transient behavior of different fuel types.

SPERT Tests:

The possible effects of adding PuO_2 enrichment to UO_2 fuels in different ways is being investigated in a joint transient testing program in SPERT involving Pacific Northwest Laboratory, General Electric Company, and Idaho Nuclear Corporation.

The first test series to study PuO_2 particle size effects involves Zircaloy-clad fuel pins containing cold-pressed-sintered UO_2 - PuO_2 fuel in SPERT. PuO_2 agglomerates are simulated by placing single 500 μm diameter PuO_2 spherical particles in strategic positions within alternate enriched UO_2 pellets in the 5 in. long fuel column. The PuO_2 particles were placed at different radial positions ranging from the center to the surface within the 0.272 in. diameter pellets. A standard pin design (Figure 4.1) was used to permit a

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** General Electric Company

*** Idaho Nuclear Corporation

FIGURE 4.1. SPERT Fuel Pin Assembly

comparison of the results from these tests with the results of other tests involving different fuel types.

Preliminary results have been obtained from the irradiation of the first two pins in the test series. One of the capsules was irradiated at an energy input of 189 cal/g UO_2 and the other was irradiated at 213 cal/g UO_2 . Since the pins were not autoclaved prior to irradiation, the major visible effect of irradiation in the capsule irradiated at the lower energy was general and somewhat nonuniform clad discoloration in the fueled region (Figure 4.2). Small discolored areas on the cladding surface may be evidence of localized overheating associated with PuO_2 particles near the surface of the pellets. Maximum cladding temperatures of 1450 °C were measured during the irradiation.

General cladding discoloration also occurred in the fueled region of the pin irradiated at an energy of 213 cal/g UO_2 (Figure 4.3). Two small holes approximately 1200 μm diameter penetrated the cladding (Figure 4.4 and Figure 4.5). The penetrations are unusual in that they are very localized and there is no apparent associated cladding distortion. Maximum cladding temperatures of 1725 °C were measured in the region of the penetrations during irradiation.

Neutron radiography of the two irradiated pins conducted by G.E. shows that the two cladding penetrations in the pin irradiated at the higher energy are adjacent to PuO_2 particles located on the surface of pellets. In fact, the PuO_2 particles adjacent to the cladding penetrations appear to be missing and were apparently expelled through the holes. PuO_2 particles not located on the surface did not have any noticeable effect on the cladding. A cavity about 900 to 1000 μm diameter has formed around the 500 μm diameter PuO_2 particles located in the pellet interiors in both capsules. This is indicative

FIGURE 4.2. Zircaloy-Clad Pin Containing UO_2 Pellets with Discretely Positioned $500\text{ }\mu\text{m}$ diam PuO_2 Particles After Irradiation in SPERT at an Energy Input of 189 cal/g UO_2 . (Test No. 860).

FIGURE 4.3. Zircaloy-Clad Pin Containing UO_2 Pellets with Discretely Positioned $500\text{ }\mu\text{m}$ diam PuO_2 Particles after Irradiation in SPERT at an Energy Input of 213 cal/g UO_2 . (Test No. 858). Note small clad penetration near center.

FIGURE 4.4. Clad Penetration Near the Top of a Zircaloy-Clad Pin Caused by the Ejection of a PuO_2 Particle During Irradiation in SPERT (Test No. 858).

FIGURE 4.5. Clad Penetration Near the Mid-Plane of a Zircaloy-Clad Pin Caused by the Ejection of a PuO_2 Particle During Irradiation in SPERT (Test No. 858).

of localized overheating and/or localized high hydrostatic pressures caused by the high power generation in the PuO_2 particles.

The approximate effect of the PuO_2 particles was to reduce the cladding failure threshold energy from the range of 223 to 250 cal/g UO_2 for homogeneous oxide fuel to the range of 189 to 213 cal/g UO_2 although no significant fuel dispersal was associated with the cladding penetrations. Based upon the external appearance of the pins, the failure threshold energy is probably closer to 189 cal/g UO_2 than to 213 cal/g UO_2 .

The destructive postirradiation examination of the pins will provide detailed information related to the transient behavior.

TREAT Tests:

The use of hafnium absorber discs and cups on each end of pins irradiated in TREAT successfully flattened the axial power profile over the 5.5-in. fuel length and prevented power peaking at the ends of the fuel columns. Irradiation results obtained from a calibration pin show that the average power in the end pellets is approximately 80% of the power at the midplane of the fuel column and confirms the results of THERMOS and MONTE CARLO calculations (Figure 4.6).

A group of five pins (three pellet and two vibrationally compacted) were irradiated at near threshold energies in the transparent autoclave in TREAT to compare the transient behavior of the different fuel types. The pins, which were fabricated by prototypic processes, are nominally 9/16 in. OD with 0.030 in. thick Zircaloy cladding and contain enriched UO_2 (4.79% ^{235}U) fuel. A summary of the test conditions and irradiation results for the five pins is given in Table 4.1. These pins were not irradiated prior to the transient tests in TREAT.

FIGURE 4.6. Measured Axial Power Profile in a TREAT Calibration Capsule with Hafnium Absorber Discs and Cups at Each End of the Fuel Column

TABLE 4.1. Summary of Test Conditions and Irradiation Results for Pins Irradiated in TREAT

	Pin No.				
	MDF-1	MDF-2	MDF-3	MDF-4	MDF-5
Fuel Type	Vipac	Vipac	Pellet	Pellet	Pellet
Integrated Reactor Power, MW-sec	457	439	421	427	435
Peak Reactor Power, MW	1572	1483	2183	1470	1484
Reactor Period, msec	65	67	60	67	67
Total fission energy deposition, cal/g UO ₂	288	276	266	269	274
Max. recorded cladding Temperature, °C(a)	---	1601	1239	1348	---
Cladding Failure	Yes	Yes	Yes	No	Yes
Energy deposition at time of failure, cal/g UO ₂	255	270	Not evident	---	Not evident
Post-test Appearance	Large hole near mid-plane.	Hole, blisters, bulging, rod bending.	Small hole, blisters, bulging.	No hole, blisters, slight bending.	Two holes, blisters, slight bulging.
Zircaloy-water reaction, equiv. Zr, g(b)	4.93	2.4	0.69	0.94	3.2
Percent of Zr reacted	16.26	7.91	2.28	3.10	10.55

(a) Does not necessarily represent the maximum cladding temperature attained.

(b) Correction for the reaction $\text{UO}_2 + \text{H}_2\text{O} \rightarrow \text{H}_2 + \text{UO}_{2+}$ has not been made.

The fuel pins were submerged in water within the transparent autoclave in TREAT, and high speed motion pictures were taken of the pins during irradiation. The film permits an evaluation of the pin behavior during the transient.

A plot of instantaneous and integrated power as a function of time during each transient permits a comparison of the irradiations (Figures 4.7 through 4.11). Cladding temperatures during the transient were indicated on some pins by thermocouples attached at the midplane and in the lower portion of the fuel column. The thermocouples may have become dislodged from the pins before maximum temperatures were achieved.

The total fission energy deposition for the tests ranged from 266 to 288 cal/g UO_2 , and the reactor period ranged from 60 to 67 msec giving reasonably uniform test conditions for the five pins. The extent of Zircaloy-water reaction was evaluated by determining the amount of hydrogen produced during the irradiation. The capsules suffered varying amounts of damage as a result of the transient irradiations (Figures 4.12 through 4.16). Use of the hafnium absorber cups on each end of the pins caused the noticeable irradiation effects to occur in the central regions of the fuel columns and the irradiation results should not have been affected by any nonprototypic end effects.

Boiling on the pin surfaces was the first evidence of heating during the irradiations. In all cases, visible heating of the cladding did not occur until near the end of the transient. The two capsules containing vibrationally compacted fuel (MDF-1 and MDF-2) lifted from the bottom hafnium cup soon after surface boiling commenced or when the fission energy input was approximately 230 cal/g UO_2 . MDF-1 rose to a height of about 1/4 of its length when it ruptured

FIGURE 4.7. Transient Irradiation of Vibrationally Compacted UO₂ Pin MDF-1

FIGURE 4.8. Transient Irradiation of Vibrationally Compacted UO_2 Pin MDF-2

FIGURE 4.9. Transient Irradiation of Pellet - Containing UO₂ Pin MDF-3

FIGURE 4.10. Transient Irradiation of Pellet - Containing UO_2 Pin MDF-4

FIGURE 4.11. Transient Irradiation of Pellet - Containing UO_2 Pin MDF-5

FIGURE 4.12. A Zircaloy-Clad Vibrationally Compacted UO₂ Pin (MDF-1)
After Irradiation in TREAT at an Energy Input of
288 cal/g UO₂.

FIGURE 4.13. A Zircaloy-Clad Vibrationally Compacted UO₂ Pin (MDF-2) After Irradiation in TREAT at an Energy Input of 276 cal/g UO₂

FIGURE 4.14. A Zircaloy-Clad Cold-Press-Sinter UO₂ Pellet Pin (MDF-3) After Irradiation in TREAT at an Energy Input of 266 cal/g UO₂.

FIGURE 4.15. A Zircaloy-Clad Cold-Press-Sinter UO₂ Pellet Pin (MDF-4) After Irradiation in TREAT at an Energy Input of 269 cal/g UO₂.

FIGURE 4.16. A Zircaloy-Clad Cold-Press-Sinter UO₂ Pellet Pin (MDF-5) After Irradiation in TREAT at an Energy Input of 274 cal/g UO₂

at an energy deposition of 255 cal/g UO_2 . MDF-2 rose to a height of about 1/8 of its length and ruptured after settling back down at an energy deposition of 270 cal/g UO_2 . Fuel was expelled from both vibrationally compacted fuel pins at the time of cladding rupture.

Surface boiling commenced on the pins containing pellet fuel (MDF-3, MDF-4, and MDF-5) at an energy deposition of approximately 112 cal/g UO_2 . None of the pellet-containing pins lifted from the bottom hafnium cup during irradiation. Cladding penetrations resulted in MDF-3 and MDF-5 although the energy deposition at the time of rupture could not be discerned because no fuel was expelled from the pin into the surrounding water.

Because no fuel was expelled from the pellet-containing pins, the effects of the transients appear to be less severe for these than for the pins containing vibrationally compacted fuel. The failure mode in both types of fuel appears to be the result of clad melting. Higher fuel temperatures in the vibrationally compacted pins caused more cladding to melt and allowed fuel to be expelled into the water.

The failure threshold energy is about 270 cal/g UO_2 for both types of fuel and is in good agreement with the value obtained from previous tests. This indicates that the non-prototypic fabrication method and the power peaking that existed at the ends of the fuel columns in the previous test pins, did not appreciably affect the cladding failure threshold energy. The extent of the Zircaloy-water reaction is comparable for both pellet and vibrationally compacted fuel and although small, increases with increasing total fission energy deposition.

Evaluation of the data associated with these irradiations, which includes metallographic examination, is continuing.

5.0 PUBLICATIONS AND PRESENTATIONS

PUBLICATIONS

1. J. W. Kutcher. Power Tests with the UO_2 -2 wt% PuO_2 Batch Core in PRTR, BNWL-1383, May 1970.
2. R. P. Matsen. DBUFIT-I A Least Squares Analysis Code for Nuclear Burnup Data, BNWL-1396, May 1970.
3. V. O. Uotinen. The Effect of ^{241}Pu Decay on the Analysis of Plutonium Critical Experiments, BNWL-SA-3157, March 1970; Accepted for publication in Nuclear Science and Engineering.
4. S. R. Dwivedi and C. R. Richey. Heterogeneity Correction to Fast-Fission, BNWL-SA-3313, May 1970; Submitted for publication in Nuclear Applications and Technology.
5. V. O. Uotinen, J. H. Lauby, W. P. Stinson and S. R. Dwivedi. Measured and Calculated β_{eff}/λ for Uniform H_2O Lattices of Al-Pu, UO_2 and UO_2 - PuO_2 /Fuel Rods, BNWL-SA-3236, April 1970; Submitted for publication in Nuclear Science and Engineering.

6.0 COMPUTER CODE ABSTRACTS

BMC-I: THE BATTELLE MONTE CARLO CODE

D. H. Thomsen

1. Name: BMC-I (Battelle Monte Carlo Code)
2. Computer: The BMC Code is designed to operate on the UNIVAC 1108 computer system. Approximately 65K words of core are required.
3. Problem Solved: The BMC code is a general purpose neutronic Monte Carlo Code. The code was developed primarily for use in solving some of the more complex problems encountered in predicting the neutronic characteristics of thermal reactors. Some of the problems which the BMC code has been used to solve are: lattice cell parameters such as resonance escape, thermal utilization, and cross section averaging, criticality problems, power distributions, shielding problems, and first flight collision probabilities.

The BMC code is flexible as to the size and type of problem solved and in the cross section and geometry detail which may be used. The neutron fluxes and reaction rates in the energy range from 0 to 10 MeV are calculated in one, two, or three space dimensions. Statistical confidence limits are assigned to the results. The cross sections utilized in the BMC code are processed from data in the ENDF/B format.

4. Method of Solution: Analog Monte Carlo techniques are used to determine neutron histories. The neutron fluxes, reaction rates, and leakage are obtained from the histories by using an exponential track length estimator. Variance reduction techniques such as absorption weight ratioing, importance weighting, and Russian Roulette are also available.

The reactions which are treated are fission, capture, anisotropic, isotropic, and thermal elastic scattering, and inelastic $n - n'$ and $n - 2n$ reactions. Thermal scattering makes use of the ideal gas model with a corrective technique to account for thermal binding effects.

The cross sections are defined using micro-group averaged cross sections. The resonance cross sections can also be treated at each energy using the Doppler broadened Breit-Wigner resonance formulae.

The geometry is defined using regions enclosed by planes, cylinders, spheres, or boundaries of the form $A(X - X_0)^2 + B(Y - Y_0)^2 + C(Z - Z_0)^2 - K = 0$. Provisions are made for voids and for reflecting symmetry boundaries. A general source routine is available for starting the neutrons. For eigen value problems the fission neutrons are used as the source. Statistics are obtained on all calculated quantities by processing batches and taking a standard deviation of the results from each batch. A special geometry routine is included for a square array of clad fuel rods. The fuel in each rod may be different.

The BMC code contains restart capability so that the results can be examined at a number of steps in the calculation.

5. Restrictions on the Complexity of the Problem: The BMC code was designed to make it easy to change the dimensions and hence make it available for a large range of problem types. A typical problem might have 5 materials, 30 geometry regions, 10 tally regions and 190 energy groups with 60 energy groups below 1.0 eV. However, problems have been run with 2 materials, 132 geometry and 132 tally regions and 96 energy groups and with 10 materials, 60 regions, 40 tally regions, and 190 energy groups.

One restriction is imposed by the approximate model used for the thermal scattering. It has been developed for light and heavy water at 293 °K. To use it at other temperatures or for other moderators requires a further special cross section preparation.

6. Typical Running Time: The running time for the BMC code as with all Monte Carlo Codes is very dependent on the complexity of the problem. The time varies from 2 to 10 min for the simplest problems up to several hours for the more complex problems. Typical times for single rod lattice cells would be in the range of 2 to 20 min while that for a power distribution in a fuel bundle would be in the range of 1/2 to 1 1/2 hr.
7. Unusual Features of the Program: The BMC code uses data in the ENDF/B format so use can be made of the ENDF/B data files. Also the BMC code does a complete energy calculation from fast to thermal.
8. Related and Auxiliary Programs: The BMC code is an update of the RBU Monte Carlo code which was coded in machine language for the IBM-7090-7094 computer system. The BMC code package consist of three codes: BMC, BMCLIB, and LIBR. BMC is the Monte Carlo code. The BMCLIB code is used to process cross sections from the ENDF/B format and prepare a cross section library for the BMC code. The LIBR code can be used to make corrections to or listings of the BMC cross section tape. The LIBR code makes use of the generalized input routine, NAMELIST.
9. Status: The BMC code is in production use of the UNIVAC-1108 computer at Pacific Northwest Laboratory, Richland, Washington.
10. References:
  - D. H. Thomsen and T. M. Traver, BMC-I. The Battelle Monte Carlo Code, To be published, Battelle-Northwest.

- J. R. Triplett, E. T. Merrill and J. R. Burr. The BRU Reactor Burnup Code: Formulation and Operating Procedures, HW-70049, July 1961.
11. Machine Requirements: Approximately 65K words of directly addressable core storage are required by the program. Approximately 1,000,000 decimal words of scratch drum space are needed. Three tapes are needed in addition to the card reader and printer. A Calcomp plotter is required to exercise the plotting options of the code.
  12. Programming Language Used: The program is coded primarily in Fortran-V (95%) and SLEUTH (5%).
  13. Operating System: UNIVAC 1108 computer with Fortran-V compiler and CSCX operating system. The Calcomp plotter model 763 used for the plotting options.
  14. Program Information: The BMC code and the BMCLIB code each consist of approximately 50 subroutines. They make use of overlay links to conserve on memory requirements. The LIBR code is a smaller code of approximately 12 subroutines. The entire code package is contained on about 14,000 cards.
  15. User Information: The BMC code and report may be obtained either from the Argonne Code Center at Argonne National Laboratory or from Pacific Northwest Laboratory in Richland, Washington.
  16. Acknowledgment: This work is based on work performed under U. S. Atomic Energy Commission Contract AT(45-1)-1830.

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HRG-3

J. L. Carter

1. Name: HRG-3
2. Computer: The HRG-3 Code is designed to operate on the UNIVAC 1108 computer system.
3. Problem Solved: The code computes the slowing-down spectrum over the energy range 10 MeV to 0.414 eV in either the B-1 or P-1 approximation, using 68 groups of neutrons with a constant group width of $\Delta u = 0.25$. The calculated flux and current spectra are used to reduce the original 68-group cross section data to average values over as many as 32 broad groups.
4. Method of Solution: The 68 fine group fluxes and currents are calculated by one sweep through the group structure, starting from a specified source distribution. The source may be selected from among the 8 available on the data tape or may be read in. The multigroup model uses a full down-scattering matrix, with inelastic, n_2n , and P0 and P1 components of elastic scattering explicitly included. Macroscopic fine group parameters are constructed from input nuclide concentrations and microscopic parameters, available on the data tape for more than 200 individual nuclides. There is no restriction, other than availability, on the number of nuclides usable in a case. A special calculation is made in the resonance range for certain nuclides, using an adaptation of the Adler, Hinman, and Nordheim method to an intermediate resonance approximation for both the absorber nuclide and an admixed moderator. The resonance contribution is allocated to the fine groups in a consistent manner providing self-shielding in both space and energy. Additional self-shielding factors may be read in for any nuclide, if desired. The fine group fluxes and currents

are used as weighting functions in averaging macroscopic and microscopic parameters over the specified broad group structure. Output is printed and may also be punched in formats for input to any of several spatial multigroup codes. As an option, the neutron age in an infinite medium may be calculated by the moments method.

5. Restrictions on the Complexity of the Problem: Number of broad groups ≤ 32 . Broad group boundaries adjusted to coincide with one of the 68 fine group boundaries.
6. Typical Running Time: For 10 nuclides, including 4 with resonance calculation and no punched output, 40 sec for first case in a run, 20 sec for each succeeding case.
7. Unusual Features of the Program: The code strikes a balance between accuracy and speed at a level highly suitable for both survey and normal reactor design and analysis applications.
8. Related and Auxiliary Programs: NUTAPE-II, updates and/or prints the HRG data tape.
9. Status: HRG3 is in production use on the UNIVAC-1108 computer at Pacific Northwest Laboratory, Richland, Washington.
10. References:
  - J. L. Carter, HRG3: A Code for Calculating the Slowing Down Spectrum in the P₁ or B₁ Approximations, BNWL-1432, Battelle-Northwest, 1970.
  - G. D. Joanou and J. S. Dudek. GAM-I: A Consistent P₁ Multigroup Code for the Calculation of Fast Neutron Spectra and Multigroup Constants, GA-1850. General Atomics, San Diego, California, June 28, 1961.
  - F. T. Adler, G. W. Hinman, and L. W. Nordheim. "The Quantitative Evaluation of Resonance Integrals,"

paper No. P/1988, Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, vol. 16, pp. 155-171, United Nations, Geneva, 1958.

11. Machine Requirements: 64K memory, normal input, output, program, and punch units, 1 unit for library, 1 to 4 scratch units or their equivalent on drum.
12. Programming Language Used: FORTRAN-IV.
13. Operating System: UNIVAC-1108 computer with FORTRAN-IV compiler and CSCX operating system.
14. User Information: The code and report may be obtained either through the Argonne Code Center at Argonne National Laboratory or from Pacific Northwest Laboratory in Richland, Washington.
15. Material Available: Magnetic tape transmittal
HRG3 Source deck (approximately 3000 cards)
NUTAPE-II Source deck (approximately 1000 cards)
Sample problem (20 cards)
Library Data Tape.
16. Acknowledgment: This work is based on work performed under U. S. Atomic Energy Commission Contract AT(45-1)-1830.

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BRT-I: BATTELLE-REVISED-THERMOS-I

C. L. Bennett and W. L. Purcell

1. Name: BRT-I (Battelle-Revised-THERMOS)
2. Computer: BRT-I is designed to operate on the UNIVAC 1108 computer system.
3. Problem Solved: The code computes the space dependent thermal neutron density, flux and current spectra over the energy range 0 to 0.683 eV in either slab or cylindrical geometry.
4. Method of Solution: The neutron density is computed from the collision probability form of the integral transport-theory matrix equation using either a combination of power iteration, overrelaxation and extrapolation or straight power iteration. The neutron currents are computed from either the gradient of the scalar flux or the uncollided flux matrix. The flux and current spectra is used to weight point thermal cross sections over an arbitrary thermal energy range for use in multigroup transport or diffusion theory codes.
5. Restrictions on the Complexity of the Problem: Number of space points ≤ 30 , number of isotopes ≤ 30 , number of speed points ≤ 30 , number of material mixtures ≤ 8 , slab or cylindrical geometry.
6. Typical Running Time: With the random access library: 1 min with a reflecting boundary condition and 30 sec with a white boundary condition. Succeeding cases using the same cross sections take about 15 sec each.
7. Unusual Features of the Code: White albedo boundary condition, current calculation, transverse buckling, linear anisotropic scattering correction, and smeared cell punched card output which can be used as region input for a

succeeding case, are several of the options available to the user. A random access library data element can be stored on drum or disk memory, if available, resulting in a considerable decrease in running time.

8. Related and Auxiliary Programs: RLITHE, updates and/or prints the BRT data tape or random access data element.
9. Status: BRT-I is in production use on the UNIVAC-1108 computer at Pacific Northwest Laboratory, Richland, Washington.
10. References:
  - C. L. Bennett and W. L. Purcell, BRT-I: Battelle-Revised-THERMOS, BNWL-1434, Battelle-Northwest, 1970.
  - H. C. Honeck. THERMOS, A Thermalization Transport Theory Code for Reactor Lattice Calculations, BNL-5826. Brookhaven National Laboratory, Upton, New York, September 1961.
  - D. R. Skeen and L. J. Page. THERMOS/BATTELLE: The Battelle Version of the Thermos Code, BNWL-516, Battelle-Northwest. June 1967.
  - J. E. Suich and H. C. Honeck. The HAMMER System, DP-1064, Savannah River Laboratory, Aiken, South Carolina, January 1967.
  - B. J. Toppel and I. Baksys. The Argonne-Revised THERMOS Code, ANL-7023, Argonne National Laboratory, Lemont, Illinois, March 1965.
11. Machine Requirements: 64K memory, normal input, output, program, and punch units, 1 unit for library, 3 scratch units or their equivalent on drum.
12. Programming Language Used: FORTRAN-IV.
13. Operating System: UNIVAC-1108 computer with FORTRAN-V compiler and CSCX operating system.

14. User Information: The code and report may be obtained either through the Argonne Code Center at Argonne National Laboratory or from Pacific Northwest Laboratory in Richland, Washington.
15. Material Available: Magnetic Tape transmittal.
BRT-1 Source deck (approximately 3000 cards)
RLITHE Source deck (less than 1000 cards)
Library deck (8000 cards)
Sample problem (19 cards).
16. Acknowledgment: This work is based on work performed under U. S. Atomic Energy Commission Contract AT(45-1)-1830.

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DBUFIT-I: A LEAST SQUARES ANALYSIS CODE FOR NUCLEAR BURNUP DATA

R. P. Matsen

1. Name: DBUFIT-I
2. Computer: UNIVAC-1108
3. Problems Solved: DBUFIT-I is designed to extract integral cross section information from isotopic burnup data. This information is obtained by fitting burnup equations to the isotopic data using least squares fitting techniques. Burnup equations for the following transmutation chains have been encoded:

<u>Transmutation Chain</u>	<u>Type of Fuel</u>	<u>Subroutine Name</u>
$^{239}\text{Pu} \rightarrow ^{240}\text{Pu} \rightarrow ^{241}\text{Pu} \rightarrow ^{242}\text{Pu}$	Pu	THEORY/PUAL
$^{238}\text{U} \rightarrow ^{239}\text{Pu} \rightarrow ^{240}\text{Pu} \rightarrow ^{241}\text{Pu} \rightarrow ^{242}\text{Pu}$	U + Pu	THEORY/URAN
$^{242}\text{Pu} \rightarrow ^{243}\text{Am} \rightarrow ^{244}\text{Cm}$	U + Pu	AMCM
$^{235}\text{U} \rightarrow ^{236}\text{U} \rightarrow ^{238}\text{U} \rightarrow ^{237}\text{Np} \rightarrow ^{236}\text{Pu} \rightarrow ^{238}\text{Pu}$	U + Pu	UCHAIN

4. Method of Solution: An iterative technique is used to find the best least squares fit of the transmutation equations to the measured burnup data. The values of the adjustable least squares parameters at this best least fit contain the desired integral cross section information.
5. Restrictions on the Complexity of the Problem: The data from as many as 150 samples may be analyzed at one time.
6. Typical Machine Time: Computational time depends upon the number of iterations required to obtain least squares convergence and the number of data points being fitted. On the UNIVAC-1108, a test case with nine data points requires approximately four seconds per iteration. Initial guesstimates for the fitting parameters are usually good enough that most problems converge in 25 to 50 iterations.

Doubling the number of data points doubles the running time per iteration. The only other significant use of machine time occurs during the generation of plotting information which may require up to thirty seconds for each plot. Timing information has been incorporated in the printed output so that the analyst can make good timing estimates for subsequent cases.

7. Unusual Features of the Program: The DBUFIT-I code provides for a simultaneous least squares fit of all transmutation equations in a given chain to the associated burnup data. One standard deviation uncertainties for all of the measured isotopes can be accommodated by the code.
8. Related and Auxilliary Programs: DBUFIT-I supercedes the DUBLIK code. Input is read in via the NAMELIST format. If need be, a comparable generalized input routine can be substituted for NAMELIST.
9. Status: DBUFIT-I is in production use at Pacific Northwest Laboratories on the UNIVAC-1108 computer at Richland, Washington.
10. References:
  - R. P. Matsen, "DBUFIT-I, A Least Squares Analysis Code for Nuclear Burnup Data", USAEC Report BNWL-1396, Pacific Northwest Laboratory, May 1970.
  - B. H. Duane, "Maximum Likelihood Nonlinear Correlated Fields (BNW Program LIKELY)", USAEC Report BNWL-390, Pacific Northwest Laboratory, September 1967.
11. Machine Requirements: Core memory requirements are 40,000 octal locations for instructions and almost 50,000 octal locations for data storage. The DBUFIT-I code consists of approximately 3500 card images and therefore is usually stored on either magnetic tape or drum. A card reader, printer and a card punch are also required. A CALCOMP plotter is necessary if plotted results are desired.

12. Programming Language Used: The program is written in UNIVAC FORTRAN-V.
13. Operating System: UNIVAC-1108 with FORTRAN-V compiler. The equipment and processing is under the control of Computer Sciences Conversational Executive (CSCX).
14. Other Programming Information: Computations are performed in double precision in order to avoid instabilities due to round off error by taking advantage of the efficient UNIVAC-1108 double precision hardware. With some decrease in efficiency and reliability, most cases can also be run in the single precision mode, thereby saving considerable data storage space in the core memory.
15. User Information: The DBUFIT-I code and report may be obtained from either the Argonne Code Center at Argonne National Laboratory or from Pacific Northwest Laboratory in Richland, Washington.
16. Acknowledgment: This code is based on work performed under U. S. Atomic Energy Commission Contract AT(45-1)-1830.

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