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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER 1968, JANUARY, FEBRUARY 1969

April 1969

AEC RESEARCH &
DEVELOPMENT REPORT

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UC-80, Reactor Technology

PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER 1968, JANUARY, FEBRUARY 1969

By

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

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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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1.0 SUMMARY

FUELS DEVELOPMENT

The design of an instrumented FERTF experiment in which the thermal behavior of pellet and vibrationally compacted mixed-oxide fuel will be evaluated was completed. Fuel rod and element hardware fabrication is underway.

The peak burnup of internal pressure instrumented vibrationally compacted $\text{UO}_2\text{-PuO}_2$ fuel rods is 10,800 MWd/MTM. These experiments will be supplemented by the irradiation of similarly instrumented $\text{UO}_2\text{-PuO}_2$ pellet-containing rods in PRTR.

High exposure plutonium was recovered from Yankee fuel in sufficient quantity for neutron dosimetry measurements. Neutron yield measurements made on high exposure plutonium and plutonium compounds showed Al-20 wt% Pu alloy to have a neutron yield from 7.7 to 7.8 times the PuO_2 of similar isotopic content.

The peak burnup of High Power Density Elements in the Batch Core remains at 11,000 MWd/MTM because of an extended reactor outage.

Six preirradiated, 20,000 to 30,000 MWd/MTM, EBWR, Zircaloy-clad $\text{UO}_2\text{-1.5 wt% PuO}_2$ fuel rods were successfully removed from the three-rod cluster test assemblies. The rods will be assembled into the special 21-rod cluster assembly for continued irradiation in PRTR.

REACTOR PHYSICS

Analytical correlations are being made of the Batch Core fueled experiment conducted in the PRCF and the PRTR. Quantities correlated are the foil activation traverses and coolant

void coefficient in the PRCF experiment and values of k_{eff} , boron worth, peripheral fuel worth, and coolant worth in the PRTR initial critical experiment.

A comparison of exponential assembly measurements made using point detectors (indium foils) and finite-sized detectors (BF_3 chambers) has shown that the two methods yield the same results, within the experimental uncertainties. The comparisons were made in three lattices using Al-5 wt% Pu fuel and one lattice using UO_2 -4 wt% PuO_2 fuel.

The development of the Battelle Monte Carlo (BMC) code is completed. The BMC code is an improved version of the RBU Monte Carlo code and is written in FORTRAN V for use on the UNIVAC 1108 computer system. A number of test problems have been run successfully, and a code document is being written.

A computer code, KINPAR has been written to aid in nuclear kinetic studies. The code computes the ratio of effective delayed neutron fraction to the neutron lifetime, $\hat{\beta}/\lambda$. Presently, the code utilizes output from the one-dimensional diffusion theory code (HFN) as input in computing this ratio. The code is being used in theory-experiment correlation studies.

A digital computer code has been developed to analyze gamma-ray spectra to deduce intensities of the various gamma-ray photo peaks present. Initial applications have been made in studies of gamma-ray spectra from fission products in irradiated fuel, but the code is applicable in the analysis of gamma-ray spectra from any source.

Measurements have been made in the PRCF of the reactor transfer function for a core loading of 19-rod clusters of UO_2 -2 wt% PuO_2 rods in D_2O moderator. The measurements were accomplished by recording reactor noise with the assembly at delayed critical and analyzing the resulting recordings in the frequency domain. Results were obtained for three core sizes and associated boron concentrations in the moderator.

The measurements in the PRCF of the relative worths of control blade materials have been completed. The relative worths of eight different absorbing materials were measured in a UO_2 core and in a $\text{UO}_2\text{-PuO}_2$ core to precisions in the 1 to 2% range by use of the endogenous pulsed source method. The measurements also demonstrated that water gaps introduced in a uniform lattice would cause an increase in the prompt neutron lifetime in the lattice.

A model for correcting for the effect on the epithermal spectrum of upscattering by hydrogen in water has been adapted to the HRG group structure and water cross sections have been appropriately revised on the HRG data tape.

A procedure permitting the resonance contributions of each resolved resonance to be allocated among several fine groups, instead of restricted to a single fine group, has been developed and incorporated into the epithermal spectrum code HRG.

The first core of the Yankee reactor for which extensive experimental data is available is used to affirm the validity of the ZODIAC burnup code. The experimental data is subjected to least square analysis before the theory-experiment correlation is made.

The FLARE code is used to examine the detailed operation of the Oyster Creek initial uranium core. This will serve as a basis of comparison for plutonium substitution calculations.

The ALTHAEA code is examined as a low cost method of obtaining ZODIAC quality burnup predictions. The comparison shows good agreement on k_∞ , ^{235}U , ^{238}U and total Pu. Plutonium-236 was added to the ALTHAEA library. The ^{236}Pu and ^{238}Pu content of fuel from a BWR, PWR, and recycled PWR is calculated at six months after discharge.

OPERATIONS

The PRTR remained shut down following the discovery of the P-4 rupture disc failure in July 1968. Resumption of reactor operations required the installation of a satisfactory alternative to the P-4 rupture disc and resolution of other RDT requirements such as evaluation of the defects observed in the valve bodies of P-1, P-3, and P-4, and the need for single failure analyses on the containment and fog spray systems. Installation of the P-4 bypass line was completed and functionally tested.

The PRTR was prepared for startup, but evaluation of the defects in P-4 and P-1 valve bodies and the availability of a similar P-4 valve body caused the reactor outage to be continued. Instructions were received to replace the P-4 valve body and repair the P-1 valve body.

Cover plates on the primary pump check valves were removed to accomplish nondestructive testing on these primary system components. During cover plate replacement on the Primary Pump 2 Check Valve, a portion of the gasket was sheared off into the primary system. Subsequent primary system flushing has resulted in retrieval of almost all of the missing gasket material.

Sixty-three spent fuel elements were shipped to Atlantic Richfield Hanford Company for chemical processing.

During this quarter, 15 pressure tubes were visually examined. All active pressure tubes have been examined during the past year with no significant alteration in the condition of the tubes being observed.

The evaluation of the oxidation and deuterium absorption characteristics of PRTR pressure tubes indicated that the useful life will not be limited by either oxidation or hydriding at the rates indicated by Pressure Tube 5687.

Crack propagation tests of irradiated Pressure Tube 5687 indicated that the critical crack length at operating pressure is about two inches. Crack propagation tests of hydrided (275 ppm) sections of unirradiated tubing indicate that the critical crack length at operating pressure is about 1.1 in.

2.0 FUELS DEVELOPMENT

IN-REACTOR FUEL TEMPERATURE MEASURING EXPERIMENT

T. B. Burley, M. D. Freshley, and T. Itaki*

The design of an instrumented experiment in which the thermal behavior of vibrationally compacted and pelleted $\text{UO}_2\text{-PuO}_2$ fuels will be evaluated during irradiation in the FERTF was completed. The thermal behavior of the different fuel types will be compared and evaluated during programmed power increases in PRTR. The thermal characteristics of the fuel to be evaluated are: (1) fuel-to-clad heat transfer coefficients as affected by initial diametral gap size and power generation, (2) grain growth temperatures for pellet and vibrationally compacted powder fuels, (3) in-reactor grain growth kinetics (time-temperature dependence) and (4) effective thermal conductivities for pellet and vibrationally compacted powder fuels. The design variables for the experiment are summarized in Table 2.1.

Fuel element hardware fabrication is nearing completion and fuel rod fabrication is in progress.

TABLE 2.1. *In Reactor Fuel Temperature Measuring Experiment Design Variables*

Number of Rods	Fuel Form	Bulk Fuel Density, % TD	Plutonium Enrichment	Fuel Sheath Dimetral Gap, in.	Number of Thermocouples per Rod
2	Vipac	86%	$\text{UO}_2\text{-2 wt\% PuO}_2$	0.0	1 ^(a)
2	Vipac	86%	$\text{UO}_2\text{-2 wt\% PuO}_2$	0.0	2 ^(b)
1	Pellet ^(c)	92%	$\text{UO}_2\text{-1.914 wt\% PuO}_2$	0.006	1 ^(a)
1	Pellet	92%	$\text{UO}_2\text{-1.914 wt\% PuO}_2$	0.006	2 ^(b)
1	Pellet	89%	$\text{UO}_2\text{-1.914 wt\% PuO}_2$	0.012	1 ^(a)
1	Pellet	89%	$\text{UO}_2\text{-1.914 wt\% PuO}_2$	0.012	2 ^(b)

a. Thermocouple located at axial centerline of fuel. Steady state peak power generations at the thermocouple position will be 9, 11, 13, and 15 kW/ft.

b. Two thermocouples positioned at same transverse plane. Steady state power generations at the thermocouple position will be 11, 13, 15, and 17 kW/ft.

c. All pellets are 94% TD

* Exchange Scientist from PNC, Japan.

GAS PRESSURE MEASURING EXPERIMENTS

T. B. Burley, M. D. Freshley, and T. Itaki*

No additional burnup was accumulated on the internal pressure monitoring experiments in PRTR because of an extended reactor outage. The burnup levels for the high and low power fuel rods are 10,800 and 6,000 MWd/MTM respectively. Continued analysis and interpretation of the data obtained during the irradiation indicated that within the burnup and fuel temperature ranges tested:

1. Internal pressure due to the release of sorbed gases and moisture from the fuel is considerably less than predicted. Based upon the known amounts of sorbed gases and moisture present in the fuel and the volumetric averaged fuel temperature, the predicted release of these constituents should have resulted in a considerably higher pressure during the initial fuel heatup. It is suspected that most of the entrapped oxygen and moisture are released from the fuel during the initial stages of irradiation and react rapidly with the unautoclaved Zircaloy cladding to form an oxide layer on the inner surface and some of the hydrogen is absorbed. Some of the oxygen may also react with the lower temperature fuel material.
2. Internal pressure buildup rates are consistent with fission gas release fractions predicted from measurements made previously on similar PRTR fuel. The previous study related fission gas release fractions to volumetric-averaged fuel temperatures.
3. The rate of pressure buildup due to fission gas release is independent of burnup or burnup rate to the levels achieved thus far in the experiment.

* *Exchange Scientist from PNC, Japan.*

4. Internal gas pressure increases in a stepwise manner during reactor power cycles (shutdown-startup periods) and levels off during steady power operation in the high power rods. However, the stepwise pressure increases in the low power rods are comparatively small and, in most cases, are not detectable. This behavior suggests a different predominance of the contributing gas release mechanisms in the high and low power rods.

It is hypothesized that in the high power rods, fission gas mobility is a diffusion-controlled mechanism in which the gas release is retarded during short steady-state operating periods by trapping sites within the fuel. Release takes place primarily when thermal stress-induced cracking occurs during power cycling. This hypothesis is supported by the fact that the magnitude of the stepwise increases is related to the length of the steady-state operating period immediately preceding the power cycling.

Diffusion controlled fission gas mobility is less pronounced in the low power rods, and a recoil-knockout process is probably the predominant release mechanism. This mechanism results in a more continuous release of gaseous fission products during steady-state operation, and stepwise increases are less predominant. The hypothesized fission gas release behavior for the high and low power fuel rods is in agreement with the results of experiments by Carroll, Sisman et al., in which fission gas release mechanisms for high and low temperature pneumatically impacted UO_2 were studied.⁽¹⁾

Additional experiments similar to the present ones will be conducted to further test this hypothesis. In these experiments fission gas release and internal pressure buildup characteristics in two vibrationally compacted and four

cold-pressed-sintered mixed-oxide fuel rods will be monitored and compared. Two each of the four pellet-containing rods will be assembled with 0.006 and 0.012 in. fuel-to-clad diametral gaps. The rods were designed with small gas plenums to increase the sensitivity of the pressure monitoring. The sensitivity, as affected by the quantity of gas present, will be two to three times that of the rods in the present experiments. The increased sensitivity will allow fission gas release mechanisms to be studied more accurately and a more accurate assessment of other behavioral characteristics such as the small pressure fluctuations observed during steady-state operation in the low power-low temperature rods.⁽²⁾ The fluctuations are thought to be a result of either irradiation induced fission gas reabsorption or problems associated with measuring small changes in pressure.

References

1. R. M. Carroll, O. Sisman, T. W. Fulton, R. B. Perez and K. Inoue. "Fission Gas Release During Fissioning of PuO_2 ," Reactor Chemistry Division Annual Progress Report for Period Ending January 31, 1965, ORNL-3789, p. 223. Oak Ridge National Laboratory, Oak Ridge, Tennessee, April 1965.
2. T. B. Burley and M. D. Freshley. "Gas Pressure Measuring Experiments," Plutonium Utilization Program Technical Activities Quarterly Report March, April, and May 1968, BNWL-828, p. 2.13. Pacific Northwest Laboratory, Richland, Washington, September 1968.

HIGH EXPOSURE PLUTONIUM STUDIES

R. C. Smith and L. G. Faust

Plutonium in sufficient quantity for neutron dosimetry measurements was recovered from one of the highest burnup Yankee fuel subassemblies. The average fuel burnup is in excess of 38,000 MWd/MTM. Special chemical purification steps were taken to assure that the high purity plutonium product does not contain any americium and uranium, because the plutonium will be used to study the early growth of the two daughter products of ^{241}Pu namely, ^{237}U and ^{241}Am . This is the first large-batch sample of high burnup Yankee plutonium that has become available for detailed plutonium radiation dosimetry measurements which can be associated with specific reactor parameters. It is of particular value for neutron studies.

Neutron yield measurements were made on a series of plutonium metal, plutonium oxide, and Al-Pu alloy samples. A comparison of the yields from the various forms of plutonium is given in Table 2.2. As previously reported, the neutron yield is primarily a function of the ^{238}Pu and ^{240}Pu isotopic content and the plutonium fuel form although ^{242}Pu , if encountered in large concentrations, could become an important contributor after several recycles.

TABLE 2.2. *Comparison of Neutron Yields Versus Plutonium Form and Isotopic Composition*

Sample	Form	Pu Isotopic					Yield, neutrons/g-sec
		238	239	240	241	242	
DZA-28-7-245	Metal	0.20	40.48	42.16	12.50	4.66	527
27-1-7/9	Oxide	0.53	65.4	23.4	7.26	3.34	493
Al-Pu Alloy	Alloy	0.50	65.3	23.2	7.6	3.4	3850

The neutron yield measured from the pure plutonium metal sample represents spontaneous fission rate. The contribution of α, n reactions, in the presence of light elements as in the case of PuO_2 or Al-Pu, may equal or even exceed the basic or spontaneous fission neutron yields and, therefore, contribute significantly to the resultant dose rate. The measurements on the Al-20 wt% Pu alloys gave neutron yields of 7.7 to 7.8 times those from high exposure PuO_2 samples of similar isotopic content.

HIGH POWER DENSITY FUEL IRRADIATIONS IN PRTR

M. D. Freshley and T. B. Burley

No additional burnup has been accumulated on PRTR fuel elements since the last reporting period because of an extended reactor outage. The present burnup status of elements in the core is summarized in Table 2.3.

Neodymium burnup analysis of the highest burnup vibrationally compacted UO_2 -2 wt% PuO_2 fuel rod (FR-62) removed from the PRTR core (156 full power days) as part of the fuel surveillance program agrees with the calculated burnup to within 2.0%. The analytical sample yielded a peak burnup value of 7700 MWd/MTM as compared to a calculated value of 7820 MWd/MTM. The corrected fission gas release fraction for this rod is 47%. This value is higher than expected (47% versus 32%) for a peak power generation of ~ 18 kW/ft, and may be the result of a burnup effect. However, such an effect has not been indicated from the in-reactor pressure measuring experiments.

A ZrO_2 layer formed on the inner and outer surfaces of the Zircaloy cladding in the peak power region (18 kW/ft) of the rod (FR-62) and the lower power region (11.5 kW/ft) near the bottom end of the rod with no abnormal hydriding effects. However, the thickness of the ZrO_2 layer on the inner and outer cladding surfaces is greater in the high power region of the rod

TABLE 2.3. *PRTR Core Loading Status as of March 1, 1969*

<u>Element Type^(a)</u>	<u>Number of Elements</u>	<u>Actual Peak Rod Power Gen., kW/ft</u>	<u>Peak Burnup, MWd/MTM</u>
<u>Batch Core Experiments</u>			
Vipac PI ^(b) UO ₂ -2 wt% PuO ₂ (Batch Core Experimental Elements)	64	20	11,000
Commercial UO ₂ -1.94 wt% PuO ₂ Pellet Fuel (Hot Pressed and Cold Pressed and Sintered)	2	21	10,000
8-Rod FERTF Test Element	1	>18	--
<u>Fringe Position Tests</u>			
Vipac PI UO ₂ -2 wt% PuO ₂ (Pilot HPD Elements)	6	20	13,000
Swaged PI UO ₂ -2 wt% PuO ₂	1	20	8,500
Swaged PI UO ₂ -1 wt% PuO ₂	1	17	14,000
Vipac PI UO ₂ -1 wt% PuO ₂	2	15	11,500
Vipac MM ^(c) UO ₂ -0.5 wt% PuO ₂	1	16	18,500
Swaged MM UO ₂ -0.5 wt% PuO ₂	2	13	17,000
Swaged UO ₂	1	14	15,000

a. *PRTR elements are 19-rod clusters of 0.565 in. OD
Zircaloy-clad rods with active fuel lengths of 58.5 and
88.5 in.*

b. *PI = High-energy-rate pneumatically impacted fuel.*

c. *MM = Mechanically mixed fuel.*

and is indicative of flux-induced corrosion rates. A maximum ZrO_2 thickness of about 150 mg/dm^2 in the peak flux position is comparable with the growth rates measured on Zircaloy pressure tubes in PRTR by Corrosion Research Section personnel. These flux-induced corrosion rates are about a factor of 10 greater than the expected corrosion rate for prefilmed Zircaloy exposed to a similar environment in the absence of a neutron flux. Also, the character of the ZrO_2 layer on the inner cladding surface is different in the two locations. The ZrO_2 layer on the inner surface in the peak power region appears porous and friable, whereas the ZrO_2 layer is dense and more characteristic of what is normally observed in the lower power region. The porous or friable character of the oxide on the inside surface in the peak flux region is attributed to the effects of fission fragment damage due to the closeness of the fuel material. Similar observations have been reported by other experimenters.

Fuel structures formed during irradiation at the 11.5 kW/ft plane in the upper section of this rod are similar to those formed at the 11.5 kW/ft plane in the lower section of the rod (Figure 2.1). No central void or well defined columnar grains were formed at either plane although slightly more sintering occurred in the sample examined from the lower section of the rod, i.e., 72% versus 62% of the radius. The difference in sintering radii is within the variation expected for vibrationally compacted fuel. The degree of sintering in both samples, however, is considerably greater than expected for a power generation of 11.5 kW/ft under these operating conditions. Based upon the indicated maximum center temperature of approximately 1700°C and the observed sintering boundary, calculated temperature isotherms indicate that the apparent in-reactor sintering temperature of the fuel is in the range of 1100 to 1200°C , which is considerably lower than expected. Possible reasons for this apparent anomaly are being studied.

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FIGURE 2.1. Transverse Section from PRR Rod FR-62 Containing Pneumatically Impacted, Vibrationally Compacted UO_2 -2 wt% PuO_2 Fuel. The local power generation was nominally 11.5 kW/ft with maximum fuel temperatures of approximately 1700 °C.

UO₂-PuO₂ HIGH BURNUP IRRADIATIONS IN PRTR

W. J. Bailey

Irradiated, EBWR, Zircaloy-clad UO₂-1.5 wt% PuO₂ fuel rods were successfully removed from the three-rod cluster test assemblies for reassembly and continued irradiation in PRTR. Dismantling the assemblies and the removal of the rods from the stainless steel spacer plates was accomplished without difficulty. The appearance of the rod ends is shown in Figure 2.2. The next operations to be performed on the rods (peak burnup approximately 20,000 to 30,000 MWd/MTM) are to: (1) screw on the three-fin stainless steel spacers on the plenum end (presently threaded) of each rod, (2) cut threads on the end cap projections on the opposite (bottom) end of each rod, and (3) screw on the bayonet fitting adapters. The rods will then be assembled into the special 21-rod cluster assembly for continued irradiation in PRTR.

Neg 17417

Top End Rod E0-08

Neg 17416

Bottom End Rod E0-08

FIGURE 2.2 View of Top (Threaded and Plenum End) and Bottom Ends of Six Preirradiated, EBWR, Zircaloy-Clad, UO_2 -1.5 wt% PuO₂ Fuel Rods After Removal from Cluster Assemblies. The rods are 1.08 cm OD by 148 cm long with a peak burnup of 20,000 to 30,000 MWD/MTM.

Neg 17415

Top End Rod EW-48

Neg 17414

Bottom End Rod EW-48
FIGURE 2.2. (contd)

Neg 17413

Top End Rod EK-59

Neg 17412

Bottom End Rod EK-59

FIGURE 2.2. (contd)

Neg 17411

Top End Rod EJ-86

Neg 17410

Bottom End Rod EJ-86
FIGURE 2.2. (contd)

Neg 17407

Top End Rod EG-63

Neg 17406

Bottom End Rod EG-63

FIGURE 2.2. (contd)

Neg 17409

Top End Rod EJ-70

Neg 17408

Bottom End Rod EJ-70

FIGURE 2.2. (contd)

3.0 REACTOR PHYSICS

THEORETICAL CORRELATION OF THE PRTR BATCH CORE EXPERIMENT

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

Introduction

A detailed theoretical analysis is being made of the Batch Core fuel experiments conducted in the Plutonium Recycle Critical Facility (PRCF)⁽¹⁾ and the Plutonium Recycle Test Reactor (PRTR).⁽²⁾ The purpose of the analysis is to determine the limitations of certain calculational methods in their application to predicting the behavior of the PRTR Batch Core Experiment and more fundamentally in their application to predicting the behavior of plutonium fueled thermal reactor systems. The calculational methods selected for this analysis are HTH⁽³⁾ and HAMMER⁽⁴⁾ multigroup cross section codes and the HFN,⁽⁵⁾ FLOG,⁽⁴⁾ EXTERMINATOR,⁽⁶⁾ 2DB,^(7,8) PERT4,⁽⁹⁾ and KINPAR⁽¹⁰⁾ reactor codes. The results of the analysis will provide a basis for selecting methods appropriate to the calculation of the PRTR Batch Core Experiments and identify those areas in which further theoretical development is required.

Calculational methods and results of the analysis of the PRCF critical experiments were reported earlier.⁽¹¹⁾ This report contains results of a theory-experiment correlation of the axial and radial flux traverse, the coolant void coefficient, and the worth of a peripheral fuel element in the PRCF. It further contains some values of the effective multiplication factors, k_{eff} , loss of coolant coefficients, boron coefficients, and the worth of a peripheral element in the critical experiments conducted in the PRTR.

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

PRCF Critical Experiments

The gold foil activation traverses in the 13-element PRCF core with 4.64 wppm ^{10}B in the moderator were calculated using EXTERMINATOR(R-Z). The experimental and calculated axial and radial traverses are given in Figures 3.1 and 3.2 respectively. As shown in both figures, the results of the correlations in the active core regions are quite good. However, there are serious discrepancies in the reflector regions. The calculated values are much higher than the experimental values in the radial reflector and the bottom reflector region.

There are a number of aspects of the experimental procedure and the analytical correlation of experimental traverses that are inconsistent.

- 1) The moderator level used in the bare axial and radial irradiation was 2.25 in. higher than used in the just critical experiment. This increased moderator level compensated the negative worth of the foils and foil holders, but the calculated activation was obtained using fluxes from the standard critical experiment calculations. However, this difference in moderator level does not affect the activation rates in the radial direction nor in the bottom reflector region.
- 2) As shown in Figure 3.3, the radial foil holding tube passes near a control assembly and a safety rod guide tube. The absorption due to the safety rod guide tube is relatively smaller than that of a control rod assembly. The absorption difference causes an asymmetry in the activation traverse in the reflector region. The calculated traverse was obtained by calculating a core without a control rod assembly and by homogenizing the safety rod guide tube with the reflector. Furthermore, the usual radial symmetry condition was assumed in the calculation.

FIGURE 3.1. Comparison of Calculated and Experimental Radial Activation Traverses for the 13 Element Core

FIGURE 3.2. Comparison of Calculated and Experimental Axial Activation Traverses for the 13 Element Core

FIGURE 3.3. Critical Assembly Configurations

- 3) The foil holding tube for the radial traverse was placed in the middle of the moderator region between fuel elements (Figure 3.3). In this position, the fluxes seen by detector foils within the active core region near the reflector are not too different from the fluxes in the adjacent reflector, and the experimental traverse shows little peaking in the reflector. A homogeneous core with cell-averaged parameters is used in the calculation. Therefore, fluxes obtained are cell-average fluxes. In the homogenized core model, there is a discontinuity in the absorption cross section at the core-reflector interface. Therefore, the calculated gold activity traverse has a large peak in the reflector. A simple correction for the heterogeneity effect was applied to the calculated flux traverses in the reflector region. The calculated values are multiplied by the ratio of the average thermal flux in the cell to the average thermal flux in the moderator of the cell obtained from THERMOS calculations. As shown in Figure 3.2, corrections to the activation decreases the discrepancy between the calculated and experimental traverses in the reflector region, but a substantial discrepancy still remains. The calculated axial traverse in the bottom reflector shows a peak which is not seen experimentally as was the case with radial reflector. This overestimation of calculated thermal fluxes in the reflector regions undoubtedly contributes to the overestimation of k_{eff} in the 7- and 13-element cores and suggests that the standard homogenization model is not appropriate for these small and highly heterogeneous assemblies.

Coolant Void Coefficient

The change in reactivity due to loss of coolant from the central channel of the 13-element core with 4.64 wppm ^{10}B in the moderator was calculated using the HAMMER code. The

calculated and experimental values are shown in Table 3.1. Calculated values are higher than experiment and lie outside the experimental uncertainties. There is a considerably higher discrepancy when H₂O coolant is used.

TABLE 3.1. *Reactivity Change Due to Loss of Coolant from Central Channel in PRCF Experiments*

¹⁰ B in D ₂ O, wppm	D ₂ O Coolant		H ₂ O Coolant	
	Experimental	Calculated	Experimental	Calculated
4.64	-0.65 ± 0.1 mk	-0.8936 mk	-3.8 mk	-6.61 mk

PRTR Critical Experiments

The analytical correlation of the PRCF critical experiments yields useful information regarding potential problems in the analytical correlation of the experiments conducted in the PRTR. In applying the same theoretical methods to the analysis of the PRCF and the PRTR experiments certain trends in correlation results are expected. The observation of these trends helps to identify limitations in the methods.

Effective Multiplication Factor, k_{eff}

The k_{eff} values were calculated for two of the initial critical experiments conducted in the PRTR. These experiments were the 55-element core with 21.00 wppm ¹⁰B in the moderator and 31-element core with 12.4 wppm ¹⁰B in moderator, which contains 99.4% D₂O and 0.6% H₂O. The coolants used in these experiments contained 97% D₂O and 3% H₂O. The calculational methods⁽¹¹⁾ were the same as used in the PRCF analysis and were performed using the HAMMER, HTH, and 2DB computer codes. The calculated values of k_{eff} are compared with experimental values in Table 3.2. The results obtained using HTH and 2DB are in good agreement with experiment for the 55-element core. The discrepancy in the HAMMER result is -11 milli-k for the 55-element core. In the PRCF analyses,⁽¹¹⁾

TABLE 3.2. Comparison of Calculated and Measured Values of the Effective Multiplication Factor, k_{eff} for PRTR Experiments

Number of Elements	^{10}B Concentration, wppm	Calculated k_{eff}			Experimental k_{eff}
		HAMMER (4 Groups)	HTH (6 Groups)	2DB (6 Groups)	
55	21.0	0.989	1.002	0.995	1.0
31	12.4	1.016			1.0

k_{eff} obtained with HTH, HAMMER, and EXTERMINATOR were higher than the measured value for the smaller cores (approximately 2%) but decreased with increasing core size. For the 55-element PRTR core k_{eff} is closer to the experiment for all cases corroborating the trend. This trend is due to the fact that a smaller fraction of the core cells lie at the core reflector interface as the core size increases.

Reactivity Worth of ^{10}B Addition

The reactivity worth of ^{10}B addition to the moderator of the 55-element core was calculated with HAMMER. The calculated and experimental values are compared in Table 3.3. There is excellent agreement between the calculated and experimental results as was the case in the PRCF analyses.⁽¹¹⁾

TABLE 3.3. Reactivity Worth of Boron-10 Additions

Concentration of ^{10}B in 55-element Core	Coefficient of Reactivity, milli-k/wppm	
	Measured	Calculated
20.95	-8.6	-8.52
20.85	-8.6	-8.554

Worth of a Peripheral Fuel Element

The reactivity worth of a peripheral fuel element addition to the 31-element core with 12.4 wppm ^{10}B in the moderator and to the 55-element core with 21.0 wppm ^{10}B in the moderator was calculated with HAMMER. A fuel cell was homogenized and added as an annulus to the basic cylindrical core. The calculated

and the experimental results are shown in Table 3.4. The calculated values are lower than the experiment and fall within experimental uncertainty for the 32-element core and outside the experimental error for the 56-element core.

TABLE 3.4. *Worth of a Peripheral Fuel Element*

Cluster Number	^{10}B in D_2O , wppm	Worth of a Cluster, milli-k	
		Measured	Calculated
32	12.4	4.0	3.61
56	21.0	2.0	0.389

Coolant Worth

The reactivity worth of coolant in all the channels (fuel and reflector) for the 31-element core with 12.4 wppm ^{10}B in moderator and the 55-element core with 21.0 wppm was calculated with HAMMER. The worth of coolant in the central channel only and in the fuel channels only was also calculated for the 55-element core. The calculated and experimental values are shown in Table 3.5. The calculated values are consistently much higher than the experimental values. Further investigations are being made to determine the causes of this discrepancy.

TABLE 3.5. *Core Coolant Worth*

Number of Fuel Elements	^{10}B Concentration in D_2O , wppm	Reactivity Change, milli-k		
		Experimental		Calcu- lated Δk
		Dry	D_2O	
31	12.4	0.0 (100.40) (a)	20 (85.92)	52.637
55(b)	19.6	0.0 (91.56)	0.80 (91.10)	6.120
55	21.0	-	-	9.835
55(b)	21.0	-	-	6.039
55(c)	21.0	0.0 (98.40)	0.02 (96.35)	0.526

- Number in parentheses are experimental critical heights.
- No coolant in fueled channel only, unfueled channels are filled with D_2O .
- No coolant in the central fuel channel only. All other channels are filled with D_2O .

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COMPARATIVE EXPONENTIAL MEASUREMENTS USING BF_3 AND INDIUM DETECTORS

J. H. Lauby and W. P. Stinson

In order to test the relaxation length measurement method in exponential assemblies, recent experiments in the Critical Approach Facility (CAF) included comparable flux measurements using indium foils (1/2 in. diameter, 0.007 in. thick) in addition to the usual measurements made using a BF_3 proportional counter (1/2 in. diameter, 4 in. long). Indium foils were used in four lattice spacings: one with UO_2 -4 wt% PuO_2 fuel rods and three with Al-5 wt% Pu fuel rods. The Al-5 wt% Pu fuel rods are 1/2 in. diameter and 24 in. long, clad in 0.030 in. Zircaloy-2. The UO_2 -4 wt% PuO_2 fuel rods are 1/2 in. diameter and 36 in. long, clad in 0.034 in. Zircaloy-2. The moderator and reflector were H_2O .

An indium foil is effectively a point flux detector while a BF_3 proportional counter integrates the flux over the length of the counter. If the flux is assumed to vary exponentially over the counter length, the ratio of the flux average ($\bar{y}$) to the flux at the center point (y_c) is given by;

$$\frac{\bar{y}}{y_c} = \frac{1}{2\lambda} \sinh 2\lambda \text{ (for a counter length of 4 inches),}$$

where λ is the exponential slope.

The expected difference for measurements made with point and integral detectors would not be a function of position but only a constant correction dependent on the counter length and exponential slope.

Both BF_3 and indium measurements were made in the central fuel rod position with the fuel rod removed. The aluminum tube containing the indium foils and 3/8 in. diameter polyethylene spacers was filled with water and polyethylene. The BF_3 traverse was made in an air-filled aluminum tube and included a varying amount of coaxial cable as a function of counter position.

With the Al-5 wt% Pu fuel, measurements were made using indium foils in the 0.85 in., 1.30 in., and 1.05 in. triangular pitch arrays. In the 1.05 in. lattice, relaxation lengths were measured as a function of boron concentration in the moderator for five nearly equal steps between 0 and 285 wppm natural boron. For the 229 wppm case, relaxation lengths were measured as a function of radius. The core sizes included 127-, 229-, and 240-rod arrays which contained 46, 83, and 87% of the critical mass respectively. Only the 127-rod core data could be analyzed as an exponential measurement. The axial flux shape in the two larger core loadings was no longer exponential.

With the UO_2 -4 wt% PuO_2 fuel, indium measurements were made only in the 0.80 in. lattice. These measurements were also done as a function of core radius with loadings of approximately 50, 57, 69, and 81% of the critical mass.

Relaxation lengths were obtained by using program LEARN⁽¹⁾ to fit the equation

$$y = A \sinh B(C-x),$$

where y is the source-corrected counting rate, x is the position measured relative to the bottom of the fuel, A is a function of counting rate, B is the slope of an exponential without end effects, and C is the height above the bottom of the fuel

at which the extrapolated flux goes to zero. The positions measured with both detectors relative to the core for both types of fuel are shown in Figure 3.4. The relaxation lengths obtained with both types of detectors are listed in Table 3.6. About half of the values agree within their errors. For those that do not agree, the sign of the difference is about equally divided, implying that no systematic error is present. If the length of core traversed and the number of points measured with indium were equal to those measured with the BF_3 counter, the errors also would be more nearly equal in magnitude. The conclusions that can be drawn from these measurements are that the results are not significantly influenced by the method of measurement. The BF_3 detector, by virtue of a higher sensitivity, provides more usable data points to determine the relaxation lengths and axial extrapolation lengths with greater precision.

Indium foils were also used to measure the radial flux distributions in the 0.80 in. lattice with 277 rods of UO_2 -4 wt% PuO_2 and in the 1.05 in. lattice with 229 rods of Al-5 wt% Pu fuel and 229 wppm boron in the moderator. The indium foils to be irradiated were attached to fuel rods on a diameter of the core. The data were fitted with a J_0 Bessel function using program LEARN. The radial extrapolation lengths measured in each core agreed within their errors for irradiations made with the source in and with the source out of the exponential assembly. For the 0.80 in. and 1.05 in. lattices, the radial extrapolation lengths obtained were 7.4 ± 0.5 cm and 6.1 ± 0.2 cm respectively. Three values of the axial extrapolation length obtained from the hyperbolic sine fits to the BF_3 data agreed within their errors for the 0.80 in. lattice measurements. The error-weighted average value of these three measurements is 4.3 ± 2.3 cm, which is much smaller than the measured radial extrapolation length (7.4 ± 0.5 cm) obtained from the radial flux measurement.

FIGURE 3.4. Relative Positions of the BF₃ Chamber and Indium Foils in the 24 in. Al-5 wt% Pu Core and in the 36 in. UO₂-4 wt% PuO₂ Core

TABLE 3.6. *Relaxation Lengths and Extrapolation Lengths Measured Using Indium Foils and a BF₃ Chamber*

Lattice Spacing, in.	Exponential Loading		Concentration, Natural Boron	Relaxation Lengths, cm		Extrapolation Lengths, cm	
	(No. of Rods)	(% Critical)		Indium Foils	BF ₃ Chamber	Axial (BF ₃ Detector)	Radial (Indium)
Al-5 wt% Pu Fuel							
1.05	85	51	0	17.6 ± 0.2	17.48 ± 0.05	6.6 ± 0.5	
1.05	91	49	48	16.4 ± 0.2	16.33 ± 0.03	7.1 ± 0.5	
1.05	109	52	101	18.1 ± 0.2	18.23 ± 0.07	7.9 ± 0.7	
1.05	121	52	164	19.8 ± 0.3	19.0 ± 0.03	7.4 ± 0.3	
1.05	127	46	229	17.9 ± 0.2	18.1 ± 0.4	7.9 ± 0.7	
1.05	229	83	229	-	-	-	6.1 ± 0.2
1.05	240	87	229	-	-	-	
1.05	151	52	285	20.5 ± 0.1	21.28 ± 0.02	8.1 ± 0.2	
1.30	109	51	0	20.7 ± 0.2	20.23 ± 0.06	4.1 ± 1.5	
0.85	109	49	0	17.6 ± 0.2	18.0 ± 0.1	6.8 ± 0.3	
UO ₂ -4 wt% PuO ₂							
0.80	169	50	0	17.5 ± 0.3	17.33 ± 0.07	4.6 ± 6.6	} *
0.80	193	57	0	20.8 ± 0.3	19.86 ± 0.08	3.8 ± 3.5	
0.80	235	69	0	27.3 ± 0.6	27.5 ± 0.4	7.4 ± 8.1	
0.80	277	81	0	50.3 ± 0.9	59.2 ± 1.3	21.2 ± 7.4	7.4 ± 0.5
						*4.3 ± 2.3 (average)	

Measured relaxation length are usually used to evaluate bucklings by equating the material and geometric buckling formulas and iterating on λ , the extrapolation length, which is assumed to be equal in the axial and radial directions. If the measured extrapolation length (λ_r), relaxation length (γ), and radius (R_e) for the 277 rod loading are substituted into the formula

$$B_m^2 = \left(\frac{2.405}{R_e + \lambda_r} \right)^2 - \left(\frac{1}{\gamma} \right)^2 ,$$

a measured buckling value of 8.54 m^{-2} is obtained. Since the measured material buckling for a given fuel configuration is independent of the diameter of the exponential assembly used in the measurement, substitution of the other three measured values of relaxation length and related core radii into the buckling formula yields λ_r as a function of the radius of the exponential assembly. A plot of these results is shown in Figure 3.5. The

FIGURE 3.5. Variation of Radial Extrapolation Length with Core Radius, for Rods of UO_2 -4 wt% PuO_2 Fuel in Light Water Arranged in a 0.80 in. Triangular Pitch Lattice

calculated points have been fitted by inspection in order to extrapolate into the range of the critical radius for this core. Using the same buckling value and the measured value of the axial extrapolation length, $\lambda_a = 4.3 \pm 2.3$, in the geometric buckling formula

$$B_g^2 = \left(\frac{2.405}{R_c + \lambda_r} \right)^2 + \left(\frac{\pi}{H + 2\lambda_a} \right)^2 ,$$

one can solve for $(R_c + \lambda_r)$. R_c can be determined by reading λ_r from the curve in Figure 3.5 at $(R_c + \lambda_r)$ and subtracting. The number of fuel rods at critical obtained by this method is 342. The buckling and number of rods at critical for the 0.80 in. lattice are compatible with results reported previously for other UO_2 -4 wt% PuO_2 lattices.⁽²⁾

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THE BATTELLE MONTE CARLO (BMC) CODE

D. H. Thomsen

Theoretical methods and analytical techniques utilized for reactor design are based upon the use of approximations and assumptions. Monte Carlo techniques are useful in evaluating the effects of assumptions and approximations on the accuracy of computed results. Frequently, these studies dictate the need for improvements in the design methods.

The RBU Monte Carlo code was developed for this purpose.⁽¹⁾ The code was written in machine language for use on the IBM-7090 digital computer. A more flexible and faster running

code than RBU was desirable and became possible with the availability of the UNIVAC 1108 computer at the Hanford project. Thus the Battelle Monte Carlo (BMC) code⁽²⁾ was developed.

The BMC code is an updated version of the RBU Monte Carlo code. The BMC code is coded in FORTRAN V for use on the UNIVAC 1108 computer system.

The BMC code uses many of the theories as formulated in the RBU Monte Carlo code. However, there are several major differences.

- Statistics are obtained as an integral part of the output.
- The batch method is used to obtain statistical samples instead of equilibrium and census times.
- The weight reduction method of accounting for absorption is used.
- Provisions were added for controlling the time spent in the thermal calculation to that spent in the epithermal calculation.
- The cross sections are obtained from the ENDF/B cross-section format.⁽³⁾

Coding of the BMC code and the BMC cross-section library code, BMCLIB, is complete. A number of test problems have been run, and testing continues. Details of the changes, input requirements, and output format are being documented.

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KINPAR

N. E. Carter and W. L. Purcell

An accurate prediction of the nuclear kinetic characteristics is important to the safety of reactor design. An important nuclear kinetic parameter is the ratio of effective delayed neutron fraction to the neutron lifetime, β/ℓ . To aid in kinetic studies, a computer code called KINPAR has been written for calculating this ratio.

The code uses standard perturbation theory expressions⁽¹⁾ for β and ℓ . The code in its present form utilizes information (such as direct and adjoint fluxes) obtained from a one-dimensional diffusion theory solution (code HFN)⁽²⁾ to the problem to perform the calculation of β/ℓ . Modifications were made to routines of HFN to manipulate the data necessary for KINPAR. The KINPAR problem directly follows the HFN problem.

The code is presently being used in studies of analytically correlating measured values of β/ℓ .⁽³⁾

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ANALYSIS OF GAMMA-RAY SPECTRA BY DIGITAL COMPUTER TECHNIQUE

G. D. Seybold

The problem of using high-resolution spectral data from fission-product gamma rays to infer fuel burnup is under study at PNL. An essential part of the solution to this problem is a means of obtaining realistic determinations of the intensities of the various gamma-ray photopeaks from the fission products of interest. A computer program has been developed which performs this portion of the analysis.

Programs General Capabilities

Program GSSLRN is a photon spectrum analysis code written in Fortran V for the UNIVAC 1108, which examines the digitized data from a multichannel analyzer, searches for and locates the photopeaks present, fits each photopeak with an analytic function (Gaussian (s) + exponential), test each fitted peak against statistical criteria such as the full-width at half-maximum, and integrates the resolved area under each photopeak to yield the intensity for that peak. Multiplets with mean values considerably closer than their fitted full-widths-at-half-maximum would normally allow can be identified and resolved, including small intensity peaks lying under the wings of larger peaks. The mean energies and their errors for each peak and intensities are also determined. The program is still under development, but has proven to be highly successful with spectral data from fuel irradiated in the EBWR.

Program Specifics

Program GSSLRN has been written using the fitting logic of program LEARN⁽¹⁾ as the heart of its analysis. The intent is to apply the most stringent statistics available, using the second order Taylor expansion techniques as they are applied in the Newton-Raphson iteration method.

The program is directed toward the problem of resolving multiplets which can be very closely coupled. The resolution of Gaussians implies that the theoretical function can be recursive for the assumed Gaussian shapes. The analytical function is of the form

$$Y = A_1 e^{A_2 (x_m - x_1)} + \sum_{i=3, JJ, 3}^{JJ=32} A_i \cdot \exp \left[-1/2 (A_{i+1} x_m - A_{i+2})^2 \right]$$

where A_1 = amplitude of the exponential background,
 A_2 is a parameter defining the slope of the background exponential,
 A_i is the amplitude of the symmetric part of each Gaussian,
 $A_{i+1} = \frac{1}{\sigma}$ for each Gaussian, and
 $A_{i+2} = \bar{X}/\sigma$ where $\bar{X}$ is the mean energy or channel number, and
 $\sigma = \text{FWHM}/2.355$ for each Gaussian.

Because of the recursive properties of the theoretical function, it is possible to obtain up to ten Gaussian-fitted shapes in multiplet form characterizing a "set." This method has proven to be successful and allows the analysis of resolved Gaussians to be more complete.

Currently, GSSLRN uses a peak-searching technique,⁽²⁾ which tends to be overly conservative. This technique tends to locate false peaks in addition to those which really exist. The logic coded into GSSLRN, along with the regression analysis in LEARN,⁽¹⁾ has proven successful in eliminating false peaks. In addition, the analysis has shown that unresolved photopeaks can be located and resolved with good statistical determination. This technique has been applied to high as well as low intensity spectra.

Input for GSSLRN

Except for the photopeak spectra, the input required for executing the code can be put on one or two cards.

The major options available are the following:

1. Execution with no plotting
2. Same as (1) except with uncertainties for all data being printed
3. Execution with computer plots
4. Execution with CALCOMP plots.

One can expect that the execution time using (3) would be approximately 3.25 sec per peak on the average. This average includes trying to analyze false peaks referred to earlier. Presently, an additional 20 sec is required for initial allocation within the computer, making the total processing time approximately $(20 + 3.25 N)$ sec.

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REACTOR NOISE EXPERIMENTS IN THE PRCF WITH UO_2 -2 WT% PuO_2
FUEL IN D_2O

J. H. Lauby

A series of reactor experiments with 19-rod clusters of UO_2 -2 wt% PuO_2 fuel in D_2O ⁽¹⁾ were performed in the Plutonium Recycle Critical Facility (PRCF) prior to loading that fuel into the Plutonium Recycle Test Reactor (PRTR) for the Batch Core Experiments.⁽²⁾ Reactor noise techniques were used to measure reactor transfer function. The transfer function was measured for three core sizes and associated boron concentrations in the moderator.

Three BF_3 ion chambers (1.4 in. OD by 8 in. long) were spaced symmetrically at a radius of 33 in. around the axial midplane of the core. A block diagram of the recording circuit used is shown in Figure 3.6. Because the ion chambers were not of the guarded high voltage type, it was necessary to isolate the high voltage by feeding the signal through a capacitor-resistor network having the proper time constant. The signal was amplified and recorded on magnetic tape.

FIGURE 3.6. Block Diagram of the Instrumentation Used for Recording Reactor Noise

Analysis of the recorded noise was accomplished using a 10-channel frequency analyzer. Each frequency channel integrates the amplitudes over the length of the recording, and an analog-to-digital conversion gives numbers proportional to the sum of amplitudes for each frequency.⁽³⁾ The recorded signals were analyzed at four tape speeds (1-7/8, 3-3/4, 7-1/2, and 15 in./sec), resulting in amplitudes measured at each of 40 frequencies. To obtain data at sufficiently low frequencies to define the transfer function for the 0.93 wppm case, the

original data tape was re-recorded to increase all frequencies by a factor of eight, thus providing amplitudes at 80 frequencies by a factor of eight, thus providing amplitudes at 80 frequencies to define the transfer function.

Each data set was fit with the equation,

$$|TF|^2 = A_1 + \frac{A_2^2}{\omega^2} \left(\frac{\omega^2 + A_3}{\omega^2 + A_4} \right),$$

where $|TF|^2$ is the square of the amplitude divided by the frequency, A_1 is the white noise background, A_2 characterizes the amplitude of the plateau of the function, $\sqrt{A_3}$ is 2π times the low frequency break, $\sqrt{A_4}$ is β/ℓ where β is the delayed neutron fraction and ℓ the prompt neutron lifetime, and ω is 2π times the frequency. Program LEARN⁽⁴⁾ was used to fit the data. The values obtained for β/ℓ are listed in Table 3.7 along with values calculated using a multienergy group diffusion theory code, KINPAR.⁽⁶⁾ The cause of the systematic discrepancies between experiment and calculation has not been determined.

TABLE 3.7. Comparison of Measured and Calculated Values of β/ℓ

Core Size (No. Clusters)	Moderator Boron Concentration, wppm ¹⁰ B	Measured β/ℓ	Calculated ⁽⁵⁾ β/ℓ
7	0.93	6.2 ± 0.3	4.42
13	4.70	9.8 ± 0.3	8.71
28	11.23	13.2 ± 0.4	12.38

The change in the ratio β/ℓ is related primarily to the decrease in the lifetime, ℓ , as the absorption cross section of the core increased, due to increased ¹⁰B concentration in the moderator.

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MEASUREMENT OF THE RELATIVE IMPORTANCE OF CONTROL BLADE MATERIALS BY THE ENDOGENOUS PULSED SOURCE METHOD

N. Pacilio* and J. B. Edgar

The importance of various reactor poison materials in the form of cruciform control blades has been measured in the PRCF under the joint USAEC-CNEN Program to study the use of plutonium in H_2O -moderated reactors. The measurements were conducted in an H_2O -moderated square pitch lattice of 0.75 in. spacing. The core was fueled with the basic core configurations for the joint USAEC-CNEN Program⁽¹⁾ of either UO_2 -2 wt% PuO_2 (8% ^{240}Pu) clad in Zircaloy-2 or 2.35 at% $^{235}UO_2$ clad in aluminum and with moderator-to-fuel volume ratios of 1.55 and 2.41 respectively. The final results of the blade importance measurements are contained in this report.

Two techniques were used for the control blade measurements. First, the endogenous pulsed source method^(2,3) was employed to determine the blade's importance in α units (sec^{-1}). An explanation of the method used in these

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measurements has been previously described.^(4,5,6) The second method used was to load the reactor to critical with each of the various blades in place. The number of fuel elements for each critical loading in a fully reflected core reflects the relative importance of the control blade. Table 3.8 displays the results of the two techniques for the UO_2 and UO_2 -2 wt% PuO_2 fueled loadings.

Three fuel configurations (A, B, and C) were used in the α measurements. In Configuration A, the core was loaded to critical with no blade in place. Each blade was inserted into this basic core in turn and the α value determined for the basic core with the blade in place. Fuel rods were then added to determine the number of rods needed to achieve criticality with each blade in place.

In Configuration B, the core was loaded to critical with a Boral blade in place. Each of the listed blades was inserted in turn and the α value for that array determined. No additional critical loadings were made.

In Configuration C, the core was loaded to critical with the blade containing B_4C cylinders in place. Each of the listed blades was inserted in turn, and the α value for that array determined.

The α measurements in Configuration A were made in assemblies which were far subcritical. This made it difficult to determine the relative worths of blades which were quite close together in strength. The α measurements made in Configurations B and C were made in cores relatively close to critical, thus reducing the size of the total uncertainty on the measured α values and permitting blades with similar worths to be distinguished from each other.

TABLE 3.8. Control Blade Importance Measurements

Blade Material	UO ₂ (2.35 wt% ²³⁵ U)			UO ₂ -2 wt% PuO ₂ (8 at% ²⁴⁰ Pu)		
	Alpha (sec ⁻¹)	No. of Rods Loaded in		Alpha (sec ⁻¹)	No. of Rods Loaded in	
		α Measurement	Fully Reflected Critical		α Measurement	Fully Reflected Critical
Configuration "A"						
Blade-free	134.0±1.1(0.8%)	461(94.68)	430.3	100.5±0.4	257(94.14)	241.2
HfO ₂ Cylinders	606.0±10 (1.6%)		525.6	855 ± 10(1.2%)		295.2
HfO ₂ Flat Tubes	652.0±14 (2.1%)		529.9	991 ± 9 (1.0%)		299.1
Hf Metal (0.09" thickness)	761.0±11 (1.4%)	812.0±7 (0.9%)	548.8	1132 ± 15(1.3%)	1199 ± 15(1.2%)	312.7
Al-B ₄ C	808.0±7 (0.9%)		557.8	1194 ± 19(1.6%)		315.4
Hf Metal (0.155" thickness)	812.0±7 (0.9%)		559.7	1199 ± 15(1.2%)		315.2
0.75 HfO ₂ -0.25 B ₄ C Flat Tubes	834.0±8 (1.0%)	841.0±8 (0.9%)	563.6	1268 ± 18(1.4%)	1244 ± 17(1.3%)	317.2
B ₄ C Cylinders	841.0±8 (0.9%)		568.2	1244 ± 17(1.3%)		318.4
Rare Earth-Pyrophafnate Tube	850.0±7 (0.8%)		1426.0±18 (1.3%)	567.4	1292 ± 18(1.4%)	1322 ± 15(1.1%)
B ₄ C Flat Tubes	881.0±11 (1.2%)	575.4		1322 ± 15(1.1%)	322.6	
Al-B ₄ C(with wings)	1426.0±18 (1.3%)	□		1924 ± 25(1.3%)	377.3	
Configuration "B"						
Base (Al-B ₄ C)	135.5±1.2(0.9%)	593(94.81)	557.8	□	333(94.67)	315.4
0.75 HfO ₂ -0.25 B ₄ C Flat Tubes	153.5±2.1(1.4%)			140.7±1.6(1.1%)		
B ₄ C Cylinders	159.8±1.7(1.1%)			153.7±1.3(0.8%)		
Rare Earth-Pyrophafnate Tube	163.9±1.6(1.0%)	188.2±1.9(1.0%)	535.0±10 (1.9%)	160.8±1.4(0.9%)	831.0±14 (1.7%)	
B ₄ C Flat Tubes	188.2±1.9(1.0%)			216.0±1.7(0.8%)		
Al-B ₄ C(with wings)	535.0±10 (1.9%)			831.0±14 (1.7%)		
Configuration "C"						
Base (B ₄ C Cylinders)	135.9±1.0(1.1%)	601(95.03)	568.2	□	333(95.38)	318.4
B ₄ C Flat Tubes	167.6±2.0(1.2%)			179.6±2.4(1.3%)		
Al-B ₄ C(with wings)	500.0±6 (1.2%)			736.0±10 (1.4%)		

() Number in parentheses is the actual critical moderator height corresponding to the number of rods given.

□ No critical measurement made

A suspicion existed that the water gaps in the basic 9×9 core increased the prompt neutron lifetime, ℓ , thereby decreasing α , since at criticality, $\alpha = \beta/\ell$. This suspicion was confirmed by the results of a critical α measurement made in the "regular" (no water gaps) UO_2 core. Table 3.9 shows the results of the critical α measurements that have been analyzed to date.

TABLE 3.9. Critical Alpha Measurements $\alpha(\text{sec}^{-1})$

	UO_2 (2.35 wt% $^{235}\text{UO}_2$)	UO_2 -2 wt% PuO_2 (8% ^{240}Pu)
Regular Core	165.7 ± 1.8 (1.1%)	135.1 ± 1.0 (0.7%)
Basic 9×9	134.0 ± 1.1 (0.8%)	100.5 ± 0.4 (0.4%)

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UPSCATTERING CORRECTIONS TO H₂O IN HRG GROUP STRUCTURE

J. L. Carter

Epithermal neutron spectra are usually calculated by methods which assume the target atoms are at rest, so that a neutron does not gain energy in a collision. This assumption breaks down as the neutron energy approaches the thermal region and leads, particularly in H₂O systems, to the calculated epithermal flux lying several percent below the measured value. The elastic scattering cross sections for H₂O on the HRG data tape have recently been modified to correct for this effect, and their values now agree well with experiment.^(1,2)

Table 3.10 illustrates the magnitudes of the changes from previous values.

TABLE 3.10. *Percent Change in Scattering Cross Section and Average Cosine of Scattering Angle*

<u>Fine Group</u>	<u>$\frac{\sigma_0}{\bar{u}}$</u>	<u>$\bar{u}$</u>
50 ^(a)	1.9 ^(c)	-0.4
66 ^(b)	6.9	-17.0
68	8.7	-25.0

a. *Highest energy group modified.*

b. *Commonly used lower group, giving 0.683 eV cutoff.*

c. *Most of change due to use of 20.36 b for free hydrogen cross section in present calculations, 20.0 b in previous calculations.*

The model used for these modifications is basically that used by Cady, et al.⁽³⁾ This model assumes first that scattering in water at these epithermal energies can be treated as though hydrogen is a mass-one gas at an elevated temperature and oxygen is at rest. To obtain better agreement with the Nelkin model,⁽⁴⁾ the Cady model further assumes that the

upscattering occurs with no energy change. In addition to improving agreement with a better model, this second assumption also uncouples the higher energy group fluxes from the lower energy group fluxes and permits the slowing down equations to be solved in the usual step-by-step process from higher to lower energies. This assumption can be replaced in the present calculations by the somewhat weaker and more satisfying assumption that upscattering remains within the group containing the energy of the incident neutron; the results are unaffected.

The mass-one gas kernels used for hydrogen in this model can be written as

$$\begin{aligned}\sigma_0(E' \rightarrow E) &= \frac{\sigma}{E} \operatorname{erf} p & E' \geq E \\ &= \frac{\sigma}{E} \exp \left[-\left(p^2 - p'^2 \right) \right] \operatorname{erf} p' & E' \leq E\end{aligned}\quad (1)$$

and

$$\begin{aligned}\sigma_1(E' \rightarrow E) &= \frac{\sigma}{E p} \left[\left(p - \frac{1}{p} \right) \operatorname{erf} p + \frac{2}{\sqrt{\pi}} \exp \left(-p^2 \right) \right] & E' \geq E \\ &= \frac{\sigma}{E p} \exp \left[-\left(p^2 - p'^2 \right) \right] \left[\left(p - \frac{1}{p} \right) \operatorname{erf} p' \right. & (2) \\ &\quad \left. + \frac{2}{\sqrt{\pi}} \exp \left(-p'^2 \right) \right] & E' \leq E\end{aligned}$$

Here $\sigma_0(E' \rightarrow E)$ and $\sigma_1(E' \rightarrow E)$ are the P_0 and P_1 scattering kernels, σ is the free atom cross section for hydrogen, $p = E/T^*$, $p' = E'/T^*$, where T^* is the effective temperature in energy units, and

$$\operatorname{erf} x = \frac{2}{\sqrt{\pi}} \int_0^x \exp(-t^2) dt$$

Values of 20.36 b and 0.1168 eV have been used for σ and T^* , respectively.

The following results are obtained by integrating Equations (1) and (2) over appropriate energies with a $1/E$ flux, and using suitable limiting approximations for $\operatorname{erf} x$ and $\exp(-x^2)$. Group cross sections:

$$\frac{\sigma_o(i)}{\sigma} = 1 + \frac{1}{2\Delta u_i} \left(1 - \frac{E_i}{E_{i-1}} \right) \frac{1}{p_i} \quad (3)$$

$$\begin{aligned} \frac{\sigma_1(i)}{\sigma} = & \frac{2}{3} + \frac{1}{\Delta u_i} \left[- \left(1 - \frac{E_i}{E_{i-1}} \right) \frac{1}{p_i} + \frac{20}{9\sqrt{\pi}} \left(1 - \frac{E_i^{3/2}}{E_{i-1}^{3/2}} \right) \frac{1}{p_i^{3/2}} \right. \\ & \left. - \frac{3}{4} \left(1 - \frac{E_i^2}{E_{i-1}^2} \right) \frac{1}{p_i^2} + \dots \right] \end{aligned} \quad (4)$$

In-group scattering:

$$\frac{\sigma_o(i \rightarrow i)}{\sigma} = 1 - \frac{1}{\Delta u_i} \left(1 - \frac{E_i}{E_{i-1}} \right) \left(1 - \frac{1}{p_i} \right) + \dots \quad (5)$$

$$\begin{aligned} \frac{\sigma_1(i \rightarrow i)}{\sigma} = & \frac{2}{3} - \frac{1}{\Delta u_i} \left\{ \frac{4}{9} \left(1 - \frac{E_i^{3/2}}{E_{i-1}^{3/2}} \right) + \frac{1}{p_i} \left[\left(1 - \frac{E_i}{E_{i-1}} \right) \right. \right. \\ & \left. \left. - \frac{4}{3} \left(1 - \frac{E_i^{3/2}}{E_{i-1}^{3/2}} \right) + \dots \right] + \frac{4}{3p_i^2} \left(1 - \frac{E_i^2}{E_{i-1}^2} \right) + \dots \right\} \end{aligned} \quad (6)$$

Between-group scattering:

$$\begin{aligned}
 \frac{\sigma_o(i \rightarrow j)}{\sigma} &= \frac{1}{\Delta u_i} \left(1 - \frac{E_i}{E_{i-1}} \right) \left(1 - \frac{E_j}{E_{j-1}} \right) \frac{E_{j-1}}{E_i} + \dots & E_j \neq 0 \\
 &= \frac{1}{\Delta u_i} \left(1 - \frac{E_i}{E_{i-1}} \right) \left(\frac{E_{j-1}}{E_i} - \frac{1}{2p_i} \right) + \dots & E_j = 0
 \end{aligned} \tag{7}$$

$$\begin{aligned}
 \frac{\sigma_1(i \rightarrow j)}{\sigma} &= \frac{2}{3\Delta u_i} \left(1 - \frac{E_i^{3/2}}{E_{i-1}^{3/2}} \right) \left[\frac{2}{3} \left(1 - \frac{E_j^{3/2}}{E_{j-1}^{3/2}} \right) \frac{E_{j-1}}{E_i} \right. \\
 &\quad \left. - 2 \left(1 - \frac{E_j^{1/2}}{E_{j-1}^{1/2}} \right) \frac{1}{p_i} \right] \frac{E_{j-1}^{1/2}}{E_i^{1/2}} + \dots & E_j \neq 0 \\
 &= \frac{2}{3\Delta u_i} \left(1 - \frac{E_i^{3/2}}{E_{i-1}^{3/2}} \right) \left[\left(\frac{2}{3} \frac{E_{j-1}}{E_i} - \frac{2}{p_i} \right) \frac{E_{j-1}^{1/2}}{E_i^{1/2}} \right. \\
 &\quad \left. + \frac{10}{3\sqrt{\pi}} \frac{1}{p_i^{3/2}} + \dots \right] & E_j = 0
 \end{aligned} \tag{8}$$

In the above equations, E_{k-1} and E_k are respectively the upper and lower energies of group k , $\Delta u_i = \ln E_{i-1}/E_i$ and $p_i = E_i/T^*$.

Equations (5) through (8) have been evaluated to obtain the hydrogen cross sections which were used in the modified H_2O cross sections now on the HRG data tape.

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THE ALLOCATION OF RESONANCE CONTRIBUTIONS TO FINE GROUPS IN HRG

J. L. Carter

According to the GAM-I procedure used in HRG for treating resonances, the effective cross section for each resolved resonance is allocated to the fine group in which the resonance energy lies. For a broad, strong resonance, this excessive concentration of absorption in a single fine group, and the resulting excessive depression of the fine group flux, is an inaccurate approximation of the true situation. To remedy this, the scheme described below for allocating the effective cross section of a resonance among several fine groups has recently been included in HRG. The effect of this change can be significant, since both the 6.7 eV resonance of ^{238}U and the 1.06 eV resonance of ^{240}Pu are examples of such a resonance. To illustrate, resonance absorption increases of about 5% in ^{238}U and 10% in ^{240}Pu and a reactivity decrease of about 25 milli-k were calculated for a representative light water lattice containing mixed oxide fuel with 2 wt% PuO_2 containing 8% ^{240}Pu .

The effective cross section for a reaction process for fine group n can be written as

$$\sigma_{n,\text{eff}} = \frac{\int_n dE \sigma(E) \phi_{\text{reg}}(E)}{\int_n dE \phi_{\text{cell}}(E)} \quad (1)$$

where it has been implicitly assumed that the isotope involved in the reaction is in a region characterized by flux $\phi_{\text{reg}}(E)$, the homogenized cell flux is $\phi_{\text{cell}}(E)$, and the homogenized and lumped nuclear densities for the isotope are related so that $N_{\text{reg}} V_{\text{reg}} / N_{\text{cell}} V_{\text{cell}} = 1$. Now for a resonance, Equation (1) can be written as

$$\begin{aligned} \sigma_{n,\text{eff}} &= \frac{\int_{\text{res}} dE \sigma(E) \phi_{\text{reg}}(E)}{\int_n dE \phi_{\text{cell}}(E)} \cdot \frac{\int_n dE \sigma(E) \phi_{\text{reg}}(E)}{\int_{\text{res}} dE \sigma(E) \phi_{\text{reg}}(E)} \\ &= \frac{\text{RI}}{\int_n dE \phi_{\text{cell}}(E)} \cdot F_n \end{aligned} \quad (2)$$

where the factor F_n can be interpreted as the fraction of the resonance integral to be allocated to fine group n . In this interpretation, the GAM-I procedure has been to set $F_n = 1$ if the resonance energy lies in n , but set it zero otherwise. The revised procedure to evaluate F_n in HRG is to homogenize the flux by an equivalence principle which approximates the collision probability by the Wigner rational approximation and to use unbroadened cross sections. Then, specifying the capture cross section for illustration, and using the usual notation,⁽¹⁾ the general form of the integrals involved in F_n can be written as

$$\int_{E_1}^{E_2} \sigma_{\gamma}(E) \phi_{\text{reg}}(E) dE = \frac{\sigma_o \Gamma_{\gamma} \beta_e}{2E_o} \int_{x_1}^{x_2} \frac{\psi}{\psi + \beta_e} dx \quad (3)$$

where

$$\psi = (1 + x^2)^{-1}, \text{ with } x = 2(E - E_o)/\Gamma$$

$$\beta_e = \frac{(1 - \lambda) \sigma_m + \lambda \sigma_p + \sigma_e}{\sigma_o} \cdot \frac{\Gamma}{\Gamma_{\gamma} + \lambda \Gamma_n}$$

$$\sigma_e = 1/N_{\text{reg}} \bar{\ell}, \text{ with } \bar{\ell} = \text{effective average chord length,}$$

and $\lambda = 0$ or 1 for IM or NR approximation, respectively.

Equation (3) can be evaluated analytically to give

$$\begin{aligned} \int_{E_1}^{E_2} \sigma_{\gamma}(E) \phi_{\text{reg}}(E) dE = \frac{\sigma_o \Gamma_{\gamma}}{2E_o} \sqrt{\frac{\beta_e}{1 + \beta_e}} & \left(\tan^{-1} \sqrt{\frac{\beta_e}{1 + \beta_e}} x_2 \right. \\ & \left. - \tan^{-1} \sqrt{\frac{\beta_e}{1 + \beta_e}} x_1 \right) \end{aligned} \quad (4)$$

The integration limits for F_n are the group boundaries of fine group n and the entire energy range for the numerator and denominator integrals, respectively, giving

$$F_n = \frac{1}{\pi} \left(\tan^{-1} \sqrt{\frac{\beta_e}{1 + \beta_e}} x_{n-1} - \tan^{-1} \sqrt{\frac{\beta_e}{1 + \beta_e}} x_n \right) \quad (5)$$

Although in principle this method gives a contribution to every fine group, in practice some cutoff procedure is needed. For this purpose, two energies equally distant from E_0 are selected such that they include between them a specified fraction of the resonance integral as given by Equation (4); this fraction has arbitrarily been chosen as 0.9. The fine groups containing these two energies are then determined, and the resonance integral is allocated to these and all intermediate groups, with the contributions beyond the cutoff energies included in the bounding groups. No additional input to HRG is required. In the illustrative calculations quoted earlier, this procedure allocated the 6.7 eV ^{238}U resonance to four fine groups and the 1.06 eV ^{240}Pu resonance to three.

The F_n depend on geometry and the scattering properties of the admixed moderator through β_e . Because unbroadened cross sections are used, the F_n are not dependent on the temperature. To check on the validity of neglecting temperature dependence, a series of calculations was made for the 1.06 eV resonance of ^{240}Pu for a range of scattering cross sections per absorber atom and temperatures from 300 to 1200 °K. The code NIT, which uses the ZUT⁽²⁾ procedure for calculating resonance integrals and allocates the contributions to the GAM-I fine group structure, was used for this check. The very slight temperature dependence found for these allocations confirms that it can reasonably be neglected.

The calculation of $\sigma_{n,\text{eff}}$ from Equation (2) also requires the evaluation of $\int_n dE \phi_{\text{cell}}(E)$. The evaluation of this integral in HRG has been described previously.⁽³⁾ However, some minor inconsistencies between the assumptions used to evaluate F_n and those of Reference 3 made it desirable to modify slightly the calculation of the flux integral. In terms of the present notation, the flux integral becomes

$$\begin{aligned}
 \int_n \phi_{\text{cell}}(E) dE &= \int_{E_n}^{E_{n-1}} \frac{dE}{E} - \frac{N_{\text{cell}}}{N_{\text{reg}}} \cdot \frac{\Gamma}{2E_0} \int_{x_n}^{x_{n-1}} \frac{\psi}{\psi + \beta_e} dx \\
 &= 0.25 - \frac{N_{\text{cell}}}{N_{\text{reg}}} \cdot \frac{\Gamma}{2E_0 \sqrt{\beta_e(1 + \beta_e)}} \cdot \pi F_n \quad (6)
 \end{aligned}$$

Equation (6) is now being used in HRG.

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AN ANALYSIS OF URANIUM FUEL IRRADIATED IN YANKEE REACTOR

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A large amount of burnup data has been obtained from fuel irradiation experiments⁽¹⁾ conducted in the Yankee reactor. The data from the unperturbed (asymptotic) region of the core have been analyzed using specially developed least squares fitting techniques,^(2,3) and the results are compared to predicted values using the ZODIAC G burnup code.⁽⁴⁾

The burnup equations used in the least square analysis of uranium fueled systems are similar to those previously developed for plutonium fueled systems.^(2,5) Because the equations have been manipulated to eliminate the unmeasured flux time variable, the flux averaged (effective) cross

sections appear in the equations as ratios. The least squares fitting techniques utilized within Program LIKELY⁽³⁾ adjust these ratios of effective cross sections to provide the best fit of the burnup equations to the data. The plutonium data obtained from these irradiation experiments and the least squares fit of the burnup equations to the data are shown in Figure 3.7 along with the values obtained for the ratios of effective cross sections. Once the fit has been completed, it is possible to calculate the number of fissions from each fissile isotope and make the conversion to megawatt days per metric ton of heavy metal (MWd/MTM) as a function of the ^{235}U left in the fuel (also plotted in Figure 3.7). The good agreement between this burnup curve and the experimental data as determined from ^{137}Cs analyses lends confidence to the results obtained by the least square analysis. The results of the least squares analysis provide a concise set of data for testing the accuracy of calculational methods. The additional information obtained from this type of analyses (i.e., ratios of effective cross sections) is very useful in gaining an understanding of the causes and effects of calculational errors.

The calculated results obtained using the ZODIAC G code are compared with the experimental data and the fitted results in Figure 3.7. The ZODIAC G code is a coupled system of the codes HRG,⁽⁶⁾ Battelle Revised THERMOS,⁽⁷⁾ HFN,⁽⁸⁾ and ALCHEMY.⁽⁹⁾ The predicted concentrations of ^{239}Pu , ^{241}Pu , and ^{242}Pu lie within the bounds of the experimental data. The calculated ratios of effective cross sections, $\hat{\sigma}_c^{28}/\hat{\sigma}_a^{25}$ and $\hat{\alpha}^{49}$, bracket the values deduced from the fit. The calculations predict more ^{240}Pu than is observed in the measured data. A comparison of the ratios of effective cross sections shows that this discrepancy occurs because the calculated burnout (which is related to $\hat{\sigma}_a^{40}/\hat{\sigma}_a^{25}$) of ^{240}Pu is underpredicted. This indicates that a problem exists in the calculation of the ^{240}Pu absorption rate.

Neg 0691102

FIGURE 3.7. Plutonium Isotopic Concentrations and Burnup for the Yankee Reactor. The data and curves are plotted as a function of U-235 which decreases with exposure. The quantities N^{49} and N^{40} are the isotopic concentrations in units of 10^{-4} atoms/b-cm of Pu-239 and Pu-240, respectively.

The correlation between the measured and calculated heat output is good with the largest discrepancy ($\sim 3\%$ at 25,000 MWd/MTM) being well within experimental uncertainties.

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CYCLE ANALYSIS STUDIES

N. E. Carter and W. C. Wolkenhauer

The objective of the cycle analysis subprogram of the Plutonium Utilization Program is to investigate core and fuel designs for optimum utilization of plutonium in Boiling and Pressurized Water Reactors (BWR's and PWR's). The value of plutonium may be enhanced through improvements in fuel design or overall operating performance. The economic evaluation of improved fueling schemes is based upon changes in plutonium value relative to a direct cycle situation. To provide a firm base for cost comparisons, both of these situations must be evaluated relative to a uranium fueled cycle.

The initial study is the uranium fuel cycle for a reference boiling water reactor. The reference reactor selected was the Oyster Creek plant.

Calculations of the physics and thermal hydraulic characteristics for the uranium fueled core have been made and a summary of results is reported here. The core operational characteristics were computed using the FLARE code.⁽¹⁾ The input data for the FLARE code were obtained from calculations using the HRG,⁽²⁾ Battelle Revised THERMOS,⁽³⁾ COBRA,⁽⁴⁾ and 2DB⁽⁵⁾ codes. The calculated results are presented in Table 3.11.

The intent of the calculations performed to date has been to provide a base case to compare fuel cycle costs for normal and low density plutonium fuels relative to UO_2 fuels. The calculated values summarized in Table 3.11 compare quite well with the values quoted in the Final Safety Analysis Report for the Oyster Creek reactor.⁽⁶⁾

Calculations for the Oyster Creek UO_2 fueled core are completed except for the economic calculations. The next stage of the study is to conduct calculations assuming a normal-density plutonium fueled core in Oyster Creek.

TABLE 3.11. *Calculated Values of Core Parameters for Uranium Fueled Oyster Creek Reactor*

<u>k_{eff}</u>	<u>U</u>	<u>P</u>	<u>E</u>	<u>Rods In</u>	<u>Curtains</u>	<u>Comments</u>
1.02816	.697	2.645	0.	0	IN	This is base case, full power
0.9995	.697	2.78	0.	29	IN	Just critical, full power
0.998	.733	1.86	2900.	0.	IN	All rods out; burn to critical
1.07512	.712	2.23	2900.	0.	OUT	Remove poison curtains
1.0009	.739	2.45	2900.	87.	OUT	Just critical, curtains out
1.0198	.740	2.15	2900.	66.	OUT	New rod pattern, now burn core
1.0112	.742	2.19	3500.	66.	OUT	Continue to expose
1.0044	.747	2.12	4100.	66.	OUT	Continue to expose
1.0393	.739	2.44	4100.	35.	OUT	New rod pattern, now continue to expose
1.03175	.740	2.52	4600.	35.	OUT	Continue to expose
1.0277	.745	2.42	5100.	35.	OUT	Continue to expose
1.0205	.744	2.31	5600.	35.	OUT	Continue to expose
1.016	.754	2.065	6100.	35.	OUT	Continue to expose
1.00965	.756	1.87	6600.	35.	OUT	Continue to expose
1.0036	.760	1.78	7100.	35.	OUT	Continue to expose
1.00853	.758	2.07	7600.	21.	OUT	New rod pattern, continued to burn
1.0007	.760	2.17	8200.	21.	OUT	Continue to expose
1.0135	.760	2.13	8200.	9.	OUT	New rod pattern
1.008	.760	2.10	8600.	9.	OUT	Continue to expose
1.0149	.760	2.04	8600.	0.	OUT	New rod pattern, now continue to burn
1.00842	.761	2.12	9100.	0.	OUT	Continue to expose
1.00205	.763	2.15	9600.	0.	OUT	Continue to expose
0.996	.765	2.14	10,100.	0.	OUT	Final Run

Nomenclature

k_{eff} = effective multiplication factor

U = relative moderator density

P = ratio of peak to average power

E = core average exposure in MWD/MT

Rods IN = number of fully inserted control rods (137 total)

Curtains = poison curtains in core.

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ZODIAC-ALTHAEA COMPARISON

D. E. Deonigi, E. A. Schnaible

The ALTHAEA code is capable of generating burnup data (isotopic concentrations and k_{∞}) at much less cost than the ZODIAC system. However ALTHAEA requires either experimental data or results from a more rigorous code like ZODIAC to set the adjustable ALTHAEA input parameters. Once the parameters are known the ALTHAEA code can produce accurate burnup data over the range of conditions made available for setting the parameters. Table 3.12 shows a comparison of ZODIAC results and the variation from these using the ALTHAEA code. Both codes were run to the same exposure using the same MeV per fission for each isotope. The comparison shows good agreement on k_{∞} , ^{235}U , ^{238}U , and total Pu. The variation in the individual Pu isotope is caused by a difference in the α^{49} , which

TABLE 3.12. ZODIAC-ALTHAEA (a) Comparison

Lattice Condition	Fuel Exposure Mwd/MT	Isotope Concentrations, g/MT							
		k_{∞}	²³⁵ U	²³⁸ U	Total Pu	²³⁹ Pu	²⁴⁰ Pu	²⁴¹ Pu	²⁴² Pu
Standard	4,000	1.1392	20,725	972,150	2181.5	1,945	195	40	1.5
0.93 cm Rod		+0.16	+0.17	+0.004	-0.13	+0.77	-8.20	-5.00	-6.67
Percent Variation (b)									
H/U = 3.97	8,000	1.0967	17,255	969,210	3858.0	3,110	542	190	16.0
Percent Variation (b)									
ZODIAC	10,000	1.0748	15,670	967,720	4587.2	3,545	728	283	31.2
Percent Variation (b)									
ZODIAC	4,000	+0.02	+0.04	+0.010	-1.02	+0.03	-3.71	-7.42	-0.96
Percent Variation (b)									
ZODIAC	8,000	1.1587	20,715	972,380	2016.5	1,795	185	35	1.5
Percent Variation (b)									
ZODIAC	10,000	+0.46	0.00	+0.002	+0.09	+1.28	-9.73	-8.57	-6.67
Percent Variation (b)									
ZODIAC	4,000	1.1133	17,110	969,710	3558.5	2,865	515	164	14.5
Percent Variation (b)									
ZODIAC	8,000	+0.06	0.00	0.000	+0.02	+1.40	-4.85	-8.54	-2.07
Percent Variation (b)									
ZODIAC	10,000	1.0895	15,490	968,270	4215.7	3,235	699	253	28.7
Percent Variation (b)									
ZODIAC	4,000	+0.14	+0.09	+0.006	-0.13	+1.33	-3.58	-9.09	-1.04
Percent Variation (b)									
ZODIAC	8,000	1.1293	20,730	972,050	2231.6	1,990	198	42	1.6
Percent Variation (b)									
ZODIAC	10,000	+0.15	+0.15	+0.008	-0.31	+0.75	-9.09	-9.52	0.00
Percent Variation (b)									
ZODIAC	4,000	1.0860	17,250	969,070	3967.0	3,210	548	193	16.0
Percent Variation (b)									
ZODIAC	8,000	0.00	+0.01	+0.014	-1.08	+0.19	-5.84	-8.81	0.00
Percent Variation (b)									
ZODIAC	10,000	1.0681	15,720	967,510	4725.8	3,665	736	293	31.8
Percent Variation (b)									
ZODIAC	4,000	-0.20	+0.15	+0.020	-1.76	-0.68	-4.35	-8.87	-0.31
Percent Variation (b)									

a. ALTHAEA Parameters
 RAYK1 = 0.61 GSHLD = 0.30 $Z_{28} = 35,000$
 RHYK2 = 0.58 $Z_{40} = 330,000$ $Z_{49} = 6,000$
 b. Percent Variation = $\frac{\text{ALTHAEA calculation} - \text{ZODIAC calculation}}{\text{ZODIAC calculation}} \times 100.$

can be adjusted only by changing basic input cross section data. Such an adjustment would make all isotopes agree within 1%.

As a result of this comparison it is concluded that ALTHAEA can be used to generate FLARE input data for analysis of fueling methods in LWR's when data for setting input parameters are supplied by ZODIAC.

^{238}Pu and ^{236}Pu Content in Recycled Pu Fuel

E. T. Merrill

Plutonium-236 was added to the ALTHAEA cross section library. The cross sections of ^{237}Np and ^{236}Pu were adjusted to obtain agreement with experimental results from Yankee. With cross sections now in use, the calculated $^{236}\text{Pu}/^{238}\text{Pu}$ ratio is 1 or 2% low at 10,000 MWd/MTM and about 7% high at 23,000. These ratios are well within experimental uncertainty. The composition of plutonium produced in PWR's and BWR's was then computed and the results are presented in Table 3.13.

TABLE 3.13. ^{236}Pu and ^{238}Pu Content of
LWR-Produced Plutonium Six
Months After Discharge

<u>Reactor</u>	<u>Exposure</u>	<u>ppm ^{236}Pu</u>	<u>% ^{238}Pu</u>
BWR Enriched U	27,500	0.057	1.4
PWR Enriched U	32,000	0.081	1.4
PWR Pu + Natural U			
1st recycle	32,000	0.044	2.2
2nd recycle	32,000	0.036	2.7
3rd recycle	32,000	0.030	3.1

In each of the plutonium recycle cases PuO_2 and natural UO_2 are assumed to be mixed to produce the fuel of proper enrichment to reach the goal exposure. The plutonium is assumed to be chemically separated six months after discharge from the reactor. The plutonium isotopes are then assumed to decay for six months before the fuel is charged to the

reactor, and all the daughter isotopes are assumed to be in the fuel. Plutonium for the first recycle is obtained by reprocessing slightly enriched uranium fuel from a PWR after an exposure of 32,000 MWd/MTM. Plutonium for the second recycle is obtained by reprocessing the fuel from the first recycle after an exposure of 32,000 MWd/MTM and adding the necessary amount of plutonium, recovered from spent slightly enriched uranium fuel, to reach the necessary enrichment. Plutonium for the third recycle is obtained by reprocessing the spent second recycle fuel and adding plutonium recovered from spent slightly enriched uranium fuel.

4.0 OPERATIONS

OPERATING EXPERIENCE

K. L. Young

The reactor remained shut down in a continuation of the outage started on July 14, 1968. The outage followed the discovery of a failed rupture disc in the P-4 valve. This problem was corrected early in December with the installation of an independently controlled bypass valve around P-4 and a blank in the rupture disc opening of the P-4 valve gate. The primary system was successfully hydrostatically pressure tested to 1875 psig on December 5, 1968 following installation of the bypass valve.

Resumption of reactor operation has been further delayed by the evaluation of casting defects observed in some of the primary system valves.

The cover plates on the primary pump check valves were removed as part of a nondestructive test examination of cast components of the primary system. During cover plate replacement on Primary Pump 2 check valve, a portion of the gasket was sheared off. A portion of the stainless steel spiral-wound asbestos-filled gasket was introduced into the primary system by this event. The primary system was flushed through screens temporarily installed in each of the outlet nozzles of the reactor's process tubes to recover this material. To date, the flushing has recovered almost all of the missing stainless steel spiral wrap. The uncertainty in the amount of material yet to be recovered is caused by uncertainty in determining how much material was actually lost from the gasket. Final flushing of the primary system will be performed after replacement of P-4 is completed.

Seventeen shipments of spent PRTR fuel material were delivered to Atlantic Richfield Hanford Company for chemical

processing during the quarter. Sixty-three fuel elements and two cans of fuel material returned from Radiometallurgy following destructive examination of PRTR fuel elements were transferred.

An additional 15 process tubes were visually examined during the quarter. All in-reactor process tubes have been examined since April 1, 1968.

PLANT ENGINEERING

D. L. Cornell

Work relating to the installation of the bypass line around the P-4 valve and closing of the rupture disc orifice in the P-4 gate were completed this quarter. Final acceptance tests were also concluded successfully.

Following completion of the bypass, Plant Engineering efforts were concentrated in preparing procedures for the repair of the P-1 check valve casting defects and replacement of the P-4 valve. Procedures for both valves have been completed and work has begun.

PROCESS TECHNOLOGY

R. H. Purcell

Evaluations of PRTR Components

Further radiography of the P-1, P-3, and P-4 valve castings (and extensive ultrasonic measurements of the P-4 casting) were performed to provide the best possible evaluation of the flaws in these castings. Evaluation of the complete radiographs provided the following descriptions:

P-1 Valve. A linear indication 2-1/2 in. long is located on the inlet side of the valve (about 300°) at the transition point from the main valve body to the top section. The flaw begins at the exterior surface of the valve body. The maximum depth visible in radiographs is about 0.126 in.

One additional area about 1 in. long is located at about 270° as disclosed by dye-penetrant indications. This indication was not detectable by radiographic techniques.

P-3 Valve. A linear indication 3/4 in. long is located on the inlet side of the top section of the valve, about 3 in. below the flange. Radiographic indications of the flaw are at a depth of about 0.190 in. from the outside surface of the casting. The flaw does not show any dye-penetrant indications on either interior or exterior surfaces.

P-4 Valve. A shrink condition at radiographic area 2-3 (Valve Zone B on Print SK-3-15052) starts at the inside wall surface and continues to a maximum radiographically visible depth of 0.628 in. The area in which this shrinkage occurs is about 2-1/2 in. long and 1/8 in. wide. Wall thickness in this area is nominally 2 in.

A discontinuity disclosed by dye-penetrant at area 4-5 (Valve Zone E on SK-3-15052) was not detectable by radiographic techniques.

A region of incomplete weld fusion is located at the plug weld beneath the gate guide 28 in. from the top of the valve (Zone C). Radiographic indications of this discontinuity are 1.480 in. from the outside surface of the casting.

A region of incomplete fusion and internal shrinkage is located beneath the gate guide about 22 in. from the top of the valve (Zone A on SK-3-15052). This region is located at a depth of about 1.160 to 1.826 in. from the outside surface of the valve body, and occurs in an area about 3 in. long and 2 in. wide. Maximum percentage of wall thickness penetrated by the radiographically visible portion of this defect region is 39%.

A 5/8 in. long linear indication is located on the opposite (north) side of the valve 1 in. from the bottom of the

valve gate guide (Zone D). This indication is located in the metal buildup immediately adjacent to the valve gate guide (HX-1 side).

A spiral-wound metal-asbestos (flexitallic) gasket from Primary Pump No. 1 (disassembled with some difficulty) suffered significant corrosion damage to portions of seven of the outside metal wraps on one side of the gasket. Representative samples of the gasket are being studied in an effort to determine causes for the corrosion and find ways to minimize such damage in the future. Laboratory analysis showed that the metal wrap in this gasket is 316 stainless steel.

Primary Pump Impeller 352473 was radiographed to determine the present condition of a defect originally detected in 1962. The present dimensions of the radiographic image from this dendritic defect are 0.700 in. maximum length and 0.150 in. maximum width, which represents no change from the original condition. No other significant discontinuities were revealed by radiography of all accessible regions (about 90%) of the impeller.

Penetrant dye inspection was performed on the visibly eroded stellite seating surface of the flapper of the P-1 check valve. The stellite surface appeared porous under examination with the penetrant dye. It is believed that this effect was produced primarily by selective corrosion of one phase of the alloy by the alkaline permanganate step used in decontamination of the primary system in 1962 and again in 1965. A new stellite surface was penetrant dye inspected and found to be relatively nonporous. Extensive grinding and polishing of the stellite surface is being performed; results thus far indicate that an adequate seating surface will be obtained.

Corrosion Surveillance

The addition of 1.0 ppm hydrazine to the PRTR primary coolant was authorized to remove oxygen as soon as possible after the coolant was put back into the primary system. Initial use of hydrazine, plus pressurization of the system with helium, was beneficial in reducing O_2 to satisfactory levels. A few days after the initial filling, it was necessary to drain the primary coolant for additional work on the system. It is planned to add hydrazine again when the primary system is refilled.

Corrosion, and corrosion product deposition, were measured on polished specimens of Zircaloy-2, stainless steel, and stainless-to-carbon steel weldments exposed for 52 operating days in the PRTR primary system (530 °F, 1050 psig, neutral pH). Weight gains on polished Zircaloy-2 specimens ranged from 7.8 to 13.5 mg/dm², in agreement with typical values for as-etched Zircaloy-2 in comparable exposures. Total corrosion penetrations were 0.02 mil for 309 stainless steel specimens (about 2 to 3 times as great as measured in pilot scale tests at 555 °F and pH 9 using LiOH). An average penetration of 0.11 mil was measured for the stainless-to-carbon steel weldment (about 80% stainless steel area) in the neutral pH coolant.

Corrosion product isotopes (⁵⁸Co, ⁵⁹Fe, ⁵¹Cr, Zr-⁹⁵Nb, ⁵⁴Mn) from the stainless steel and Zircaloy were the predominant radioactive materials present on the samples: ⁶⁰Co activity was relatively very low. The amount of deposited material (crud) on these specimens was small. For example, the average crud deposition on seven Zircaloy-2 specimens in this test was only 6.5 mg/dm² compared to 18 mg/dm² on specimens exposed 77 days in an ETR loop in pH 10 NH₄OH at about 525 °F.

Corrosion, boron deposition, and induced radioactivity were measured on specimens removed from the PRTR moderator in

November, 1968. Maximum corrosion penetration was 0.1 mil for aluminum alloys 5050, 5052, 6061, and 6063, after total reactor residence times of 330, 918, or 1218 days (3.3 years). Maximum corrosion of Zircaloy-2, Haynes Stellite, 304 stainless steel, and Inconel-600 was 0.005 mil after 918 or 1218 days. Average boron deposition on aluminum was $70 \mu\text{g}/\text{dm}^2$ for specimens exposed 330 days, $93 \mu\text{g}/\text{dm}^2$ for 918 days exposure, and $120 \mu\text{g}$ at 1218 days, indicating that deposition is about the same as on specimens removed in June and December of 1967. Induced radioactivity (after about 4-month decay) was a maximum of 132 mR/hr at 4 in. on aluminum specimens (about 6 g, surface area 0.33 dm^2) located in the mid-plane of the reactor core, but was only about 10 mR/hr, or less, on specimens located near either the top or bottom of the calandria.

Storage Basin Algae and Sediment Control

The effectiveness of low cost materials (hydrogen peroxide and hydrazine) for control of algae in the PRTR storage basin is being studied. Hydrogen peroxide is at least partially effective, but optimum dosage may be near to 100 ppm. Testing is continuing.

Development of effective methods for removal of sediment and algae from the basin requires knowledge of sediment radioactivity, particle sizes, and settling rate. A 16 oz sample of water and sediment collected from the basin floor showed a radiation dose level of 8 mR/hr, indicating that standard radiation precautions should be adequate for performing this sediment removal. Particle sizes are being measured on this sample. Microscopic examination revealed that the majority of the material was translucent and gelatinous (as from algae); heavier residue seemed to consist of paint flakes and corrosion scale. Little sand was evident. It is estimated that 90% of the material settled through a 12 in. depth in 3 min. Remaining fine material was completely settled in less than 12 hr.

Extended Batch Core Operation

A document "Review of the Extended Batch Core Operation," BNWL-CC-1960, which determined there were no unreviewed safety questions in the operation of the post batch core, was issued.

General Safeguards Activities

A study of the technical feasibility of a periodic leakage test of the air locks was completed. Technically, the air locks can be tested by blocking the inner door to prevent opening and by internally pressurizing the air lock chamber with an existing air supply. The main personnel air lock test pressure would be 15 psig and the acceptable leakage rate would be $\leq 1\%$ per day of the volume contained in the chamber. The test duration would be about 7 hr minimum.

The emergency air lock could be tested using the same design specification criteria or Technical Safety Guide 7.5.1 criteria: that is, 10 psig with an acceptable leak rate of 0.1% of the acceptable containment vessel leakage rate. The latter criteria would shorten the emergency air lock test to about 2 hr. The test frequency given in the guide for this type of test is every four months.

An operational review was made and a letter report issued on the consequences and likelihood of fuel rod bowing and wire wrap accumulation for the PRTR 19-rod cluster fuel elements. The report concluded that minimizing of fuel band displacement and damage is essential to eliminate conditions under which fuel rod bowing could occur.

Calculations of the allowable primary system bulk outlet temperature and process tube outlet temperature for various downward adjustments of the pressurizer low pressure safety circuit trip setting were completed. Revision of Process Specifications and Operating Standards was begun.

Simplified single failure analyses (as contrasted to the detailed criteria established by the IEEE Single Failure Criteria) were completed for the ventilation containment and fog spray systems. These analyses were transmitted to the AEC during January.

Three-hour sessions on reactor physics are scheduled as a part of the Reactor Operations Retraining Program. These sessions have begun and will continue next month. They cover general reactor physics, with emphasis on problems peculiar to the PRTR and the batch core experiment.

PRTR PRESSURE TUBE EVALUATION

T. R. Ostrom

The PRTR Reactor Safeguards Program includes destructive examination of discharged tubes and investigations of unirradiated material that will improve our basis for judging the structural adequacy of the tubes remaining in the PRTR environment. Tube 5687⁽¹⁾ has been destructively examined at Radiometallurgy. Work on unirradiated tubing is continuing with the testing of specimens with sharp cracks produced by pressure cycling to determine the "worst case" for crack propagation. A "worst case" crack propagation test was made on a tube section which was removed from a low flux position in the downstream end of PRTR Pressure Tube 5687. The section, 19-N, was prepared for testing by milling a slot 1/2 in. long by 80% through the wall with a 1/16 in. end mill and then pressure cycling it until a crack grew through the remaining 20% web. After the crack had grown through the web, a "rubber mouse" was placed inside the tube to stop the leaking so the pressure cycling could be continued and the crack grown out to the desired length.

Section 19-N was cycled between 100 and 2200 psi; approximately 170,000 cycles were required to fail the web. A "mouse"

was inserted and the tube was cycled to failure. Failure occurred after 8500 cycles at a crack length of approximately 1.15 in. The hoop stress at failure was 23,000 psi. This test was in good agreement with the results obtained from Section 19-B of the same tube which failed at a stress of 24,000 psi hoop and a crack length of 1.03 in. Both of these tests show results similar to those obtained from N-Reactor tubing and tend to show that there is disagreement with the results obtained from Section 19-E. (See Figure 4.1 for all results from Tube 5687 and existing base case results.)

FIGURE 4.1. Hoop Stress at Fatigue Versus Crack Length for Milled Slot and Sharp Crack (Fatigue) PRTR Specimens in the As-Received, Irradiated and Hydrided Conditions

Four additional tests have been run on annealed sections of unirradiated PRTR tubing with 275 ppm hydrogen to extend the hoop stress versus crack length line for "sharp" cracked hydrided tubing (see Figure 4.1). All four of the tube sections had starter notches 0.25 in. long by 100% wall thickness introduced by an electro-jet process. The tube sections were then patched with a piece of 5-mil thick stainless steel which was held in place with a piece of vinyl plastic. The results obtained on the four tubes were as follows:

<u>Test</u>	<u>Maximum Pressure, psi</u>	<u>Number of Cycles</u>	<u>Final Crack Length, in.</u>	<u>Hoop Stress at Failure, psi</u>
1	1300	63,000	1.28	15,500
2	1150	82,000	1.03	13,700
3	1000	127,000	1.11	12,200
4	1000	361,000	1.27	12,200

These four lower stress tests show that the critical crack length at operating pressure is about 1.1 or 1.2 in. and not the 1.9 in. predicted from the two previous higher stress tests.

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