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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1969

July 1969

AEC RESEARCH &
DEVELOPMENT REPORT

ROUTE TO	P. R. NO.	LOCATION	FILES ROUTE DATE
Artes for	60013	271-4	JAN 30 1970
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BNWL-1106
UC-80, Reactor Technology

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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1969

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

July 1969

BATTELLE MEMORIAL INSTITUTE
PACIFIC NORTHWEST LABORATORY
RICHLAND, WASHINGTON 99352

BNWL-1106

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

PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
MARCH, APRIL, MAY 1969

FOREWORD

The Plutonium Utilization Program is conducted by the Pacific Northwest Laboratory for the USAEC. The objective of the Technical Activities Quarterly Report is to inform the scientific community in a timely manner of the technical progress made on the program. The report contains brief technical discussions of accomplishments in all areas where significant progress has been made during the quarter. The results presented should be considered preliminary, and do not constitute final publication of the work. A list of publications and papers is given in the report. Anyone wishing to obtain additional information on the work presented is encouraged to contact the author directly.

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December 1967, January, February 1968	BNWL-739
March, April, May 1968	BNWL-828
June, July, August 1968	BNWL-907
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1.0 SUMMARY

FUELS DEVELOPMENT

Extensive gamma dose rate measurements were obtained from Yankee high burnup (approximately 36,000 MWd/MTM) fuel.

The peak burnup of internal pressure instrumented, vibrationally compacted, mixed-oxide fuel rods remains at 9900 MWd/MTM because of an extended reactor outage. Preparation of additional internal pressure monitoring experiments is underway on vibrationally compacted and pellet mixed-oxide fuel rods.

Fabrication of an instrumented eight-rod test assembly is underway in which fuel temperatures will be monitored during irradiation in the FERTF.

The peak burnup of High Power Density Elements in the Batch Core Experiment remains at 11,000 MWd/MTM because of an extended reactor outage. Postirradiation examination is underway of seven vibrationally compacted, mixed-oxide fuel rods irradiated to peak burnups covering the range of 3400 to 10,200 MWd/MTM being conducted as part of the Batch Core surveillance program.

Detailed metallography was conducted on two Zircaloy-clad, vibrationally compacted UO_2 fuel capsules that were irradiated in TREAT. Preparation of five additional capsules for irradiation in TREAT is underway.

REACTOR PHYSICS

The ratio of effective delayed-neutron fraction to mean neutron lifetime (β/ℓ) has been calculated for several

H₂O-moderated and reflected lattices using the computer code KINPAR. The calculated ratios compare well with measured values.

Analytical correlations were made of measured worths of peripheral and internal fuel elements in PRTR. The spectral-spatial effects of non-uniform core boundary for the PRCF Batch Core fueled experiments were also studied.

The computer code, FLARE which was imported from Argonne's Code Center, has undergone considerable revision in order to make it operational on the UNIVAC 1108 computer and to expand its range of applicability to commercial nuclear power plants.

Revisions are being incorporated into the HRG code which relax the restrictions on the theory of neutrons colliding with absorber and/or moderator nuclei and allow for a more accurate calculation of resonance absorption events.

Calculations of power distributions have been performed using various models to represent the reflector region of a critical assembly. Improvements between measured and calculated power profiles resulted from use of a "spectral hardened" reflector region.

The endogenous pulsed source method of determining the reactivity status of a critical (or subcritical) assembly has been used in an experiment to measure the temperature coefficient of reactivity for a small core of 0.50-in. diameter rods of UO₂-2 wt% PuO₂ (8% ²⁴⁰Pu) in a 0.75-in. square pitch lattice, moderated with H₂O. Over a temperature range from 30 to 75 °C, the measured result was $-(1.02 \pm 0.05) \times 10^{-4} \frac{\Delta k}{k}/^{\circ}\text{C}$.

An empirical study has been started of the shape of "full-energy" peaks in gamma-ray spectra obtained from Ge(Li) detectors. The method of attack uses a recently developed general scheme for holding fixed any subset of parameters in the nonlinear least-squares program LEARN.

An analysis of the use of natural uranium versus recovered, previously-enriched uranium as the carrier for recycled plutonium has been carried out for a limited set of conditions. The plutonium from the recovered uranium system contains about twice as much ^{236}Pu as the natural uranium system. This high ^{236}Pu content causes serious radiation problems during fabrication, and thus the used, recycled uranium will lose some of its apparent economic advantage. The ALTHAEA burnup code was used to study the effects of void fluctuations in a boiling-water reactor on the reactivity and isotopic concentrations during the burnup life of a fuel element. Standard practice is to use the time-average void condition throughout the calculations rather than vary the void as a function of control rod position and other operating conditions. The study investigates the differences between assuming average conditions and detailed void history. The results of the study indicate that the void fluctuations would have to be less than 10% in order to maintain an error of less than 1% in reactivity and fuel composition. The errors seem to strongly depend upon the void fraction history of a particular module. Whether averaging can adequately represent the void fluctuations is debatable.

OPERATIONS

The reactor remained shut down for the report period in a continuation of the outage started on July 14, 1968. The lengthy outage was caused by 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.

Resumption of reactor operation was further delayed by discovery of casting defects in several of the primary system valves. Corrective action consisted of repairing the defect in the primary pump No. 1 check valve (P-1) and replacing

the 14-in. gate valve (P-4) on the reactor outlet piping. Current activities are directed at resumption of reactor operation.

Interior surfaces of the primary system were examined visually during the repair of the P-4 valve. All surfaces were typical of clean stainless steel after exposure to high temperature water.

General safeguard activities were continued to provide improved reactor safety.

2.0 FUELS DEVELOPMENT

HIGH EXPOSURE PLUTONIUM STUDIES

R. C. Smith, L. G. Faust, H. H. VanTuyl

A high exposure plutonium sample in sufficient quantity to permit extensive gamma dose rate measurements was recovered from Yankee high burnup fuel. The average burnup of the fuel sample was approximately 36,000 MWd/MTM. ^{238}Pu and ^{241}Pu contents of the plutonium are 1.9 wt% and 12 wt%, respectively. The dose rate from the plutonium sample was measured frequently during the first 2 months following chemical purification during which time the short half-life ^{237}U (6.75 days) attains an equilibrium concentration with its parent ^{241}Pu . The concentration of ^{241}Am , the second daughter isotope of ^{241}Pu , continues to increase for several years and is the principal contributor to the increase in dose rate after the first 2 months.

The measured gamma dose rate from the Yankee plutonium sample increased 23% (4.8 rad/hr to 5.9 rad/hr) during the first 57 days. Continued measurements will be made to determine the increase in dose rate resulting from the build-in of ^{241}Am with time. It is estimated that the total surface dose rate value will double during the first year after chemical purification.

A plutonium sample containing 95% ^{240}Pu was obtained and analyzed by gamma energy spectrometry. The reported ^{240}Pu photopeaks were identified and confirmed. No additional photopeaks attributable to ^{240}Pu were observed. The L X-ray yield from ^{240}Pu with energies in the range of 13.6 to 20.1 keV was measured to be 234 times greater than the intensity of the 45 keV gamma peak from ^{240}Pu . Although expected, these data illustrate the significant contribution of the low energy radiation to the total surface dose rate from

high exposure plutonium and indicate the magnitude of the exposure problem associated with direct contact with such material. The data also illustrate the significant reduction in dose rate that is easily achieved by shielding out the low energy contribution.

The computer program PUSHLD for gamma dose rate and shielding calculations was modified to incorporate recent experimentally determined values of gamma absorption coefficients for plutonium. Calculated values now obtained agree with the experimental measurements. The code calculates surface dose rates from plutonium and dose rates through a variety of shielding materials as a function of plutonium isotopic content and time after chemical purification. The revised code and several shielding-dose rate curves will be published.

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 peak burnup levels for the high and low power fuel rods are 9,900 and 6,000 MWd/MTM, respectively. Attempts to repair the low burnup rods which were damaged as a result of non-related reactor maintenance activities were unsuccessful.

Preparation of additional experiments is underway in which fission gas release and internal pressure buildup characteristics in two vibrationally compacted and four pellet rods will be monitored and compared. Vibrational compaction of two of the rods was completed. Fabrication of the other four rods, which will contain pelleted $\text{UO}_2\text{-PuO}_2$, was delayed because of problems encountered during pellet processing.

* *Exchange Scientist from PNC, Japan*

IN-REACTOR FUEL TEMPERATURE MEASURING EXPERIMENT

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

The thermal behavior of four vibrationally compacted and four pellet-containing UO_2 - PuO_2 fuel rods during irradiation will be determined by the use of thermocouples positioned in the fuel material. The eight thermocouples containing fuel rods will be assembled into an eight-rod cluster for irradiation in the FERTF (Figure 2.1).

Fabrication of the four rods containing vibrationally compacted UO_2 -2 wt% PuO_2 was completed. Fabrication of the other four rods, which will contain UO_2 - PuO_2 pellets, has been delayed due to problems encountered during pellet processing. Efforts are being made to resolve these difficulties.

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 the elements in the core is summarized in Table 2.1.

Postirradiation examination of seven vibrationally compacted, mixed-oxide, fuel rods irradiated in PRTR to estimated peak burnups covering the range of 3400 to 10,200 MWd/MTM is being conducted as part of the Batch Core surveillance program. Analysis of the gas collected from the rods shows that other than the fission gases Xe and Kr, the major constituents are nitrogen and helium as expected (Table 2.2). A preliminary analysis of the data indicates good agreement between the actual and expected fission gas release fractions based upon

* *Exchange Scientist from PNC, Japan*

Neg. 0691975-1

FIGURE 2.1. FERTF Thermocoupled Fuel Element

TABLE 2.1. *PRTR Core Loading Status as of June 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 ² 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.*

TABLE 2.2. *Analysis of the Gas Collected from
Vibrationally Compacted UO_2 - PuO_2
Fuel Rods Irradiated in PRTR*

<u>Constituent</u>	<u>Mole Percent</u>						
	<u>FE-77</u>	<u>FO-79</u>	<u>FR-64</u>	<u>FS-16</u>	<u>FS-08</u>	<u>FS-23</u>	<u>FS-01</u>
Xe-136	5.25	4.90	22.80	24.40	27.10	27.50	31.8
134	3.82	3.46	15.40	16.10	18.00	18.20	20.6
132	2.64	2.59	10.80	11.20	12.40	12.80	14.4
131	1.80	1.77	7.31	7.53	8.31	8.43	9.40
130	--	--	0.04	0.03	0.03	0.07	0.05
129	--	--	0.10	0.06	0.01	0.25	0.01
Kr-86	0.36	0.36	1.75	1.86	1.88	2.19	2.28
85	0.05	0.05	0.25	0.28	0.27	0.30	0.34
84	0.22	0.23	1.09	1.12	1.13	1.36	1.36
83	0.14	0.14	0.62	0.63	0.62	0.74	0.74
CO ₂	0.08	0.06	<0.01	<0.01	<0.01	<0.01	<0.01
Ar	0.03	0.02	0.42	0.04	0.02	0.02	0.02
O ₂	0.09	0.06	<0.01	<0.01	0.05	<0.01	<0.01
N ₂	2.78	18.10	5.84	3.92	0.81	4.66	5.06
CO	<0.10	<0.10	<0.10	<0.10	<0.10	<0.10	<0.10
He	82.7	68.20	33.4	32.80	29.30	23.4	13.90
H ₂	0.07	0.06	0.06	0.09	0.05	<0.01	0.02
CH ₄	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01

estimated burnup values. The preliminary fission gas release data are summarized in Table 2.3.

Detailed metallography is in progress including electron microscopy of vibrationally compacted, cold-pressed-sintered, and hot-pressed UO_2 - PuO_2 fuels irradiated in PRTR to burnups as high as 7800 MWd/MTM.

TABLE 2.3. *Summary of Fission Gas Release Data*

<u>Fuel Rod No.</u>	<u>Est. Peak Burnup, MWd/MTM</u>	<u>Est. Peak Rod Power, kW/ft</u>	<u>Vol. of Gas Collected, ml (STP)</u>	<u>F. G. Release, %^(a)</u>	<u>Expected F. G. Release, %</u>
FS-01	10,200	18	179	36	25-35
FS-23	9,500	16	158	30	23-30
FS-08	8,500	15	118	25	20-25
FS-16	6,300	16	109	27	23-30
FR-64	6,300	11	83	20	15-20
FO-79	4,100	10	53	5	10-15
FE-77	3,400	8	39	4	8-13

a. *Fission gas release fraction based on the estimated burnup. This value may be changed when more accurate burnup data are obtained.*

TRANSIENT TESTING

M. D. Freshley and T. B. Burley

Detailed postirradiation examination was conducted on two Zircaloy-clad, vibrationally compacted UO_2 (5.01% ^{235}U) fuel capsules that were irradiated in the Transient Reactor Test Facility (TREAT). The transient irradiations were performed at reactor periods of 71 and 75 msec with energy inputs of 236 and 223 cal/g UO_2 , respectively. These irradiation conditions caused localized clad bulging near the bottom of the fuel columns but the cladding did not rupture. Measured cladding temperatures were as high as 1350 °C.

Detailed metallographic examination of the Zircaloy cladding around the circumference of the transverse sections substantiates the nonuniform heating effects observed during the irradiation of these capsules in the transparent autoclave⁽¹⁾. Recrystallized and transformed structure indicative of clad temperatures in excess of 1000 °C formed in two areas covering approximately 1/2 of the circumference of one specimen and in one area covering approximately 1/8 of the circumference of the other specimen (Figures 2.2 through 2.4). The non-uniform heating in vibrationally-compacted fuels is undoubtedly the result of intermittent core-clad contact. Since similar effects have been observed in pellet-containing capsules, intermittent contact appears to be a characteristic of fuel behavior under transient conditions.

Examination of the fuel substantiates that the irradiation conditions were similar for both rods. Recrystallization and equiaxed grain growth accompanied by some cracking of the fuel have occurred with no evidence of a significant temperature gradient. Although there is no indication of gross sintering, localized particle-to-particle sintering did occur (Figure 2.5). Inclusions in the fuel were identified by microprobe analysis as stainless steel probably introduced during processing.

Additional transient irradiations of vibrationally compacted and pelleted UO_2 fuel capsules will be conducted in the transparent autoclave facility in TREAT to confirm the comparative results on vibrationally compacted powder and pellet-containing capsules obtained from previous experiments. The previous results indicated that flux peaking at the ends of the fuel column and excessive internal pressures may have influenced failure threshold energies and transient behavior. The capsule design was modified to alleviate these possible problems by including hafnium neutron absorber

a. Capsule A-3
 Energy Input - 236 cal/g UO_2
 Reactor Period - 71 msec

b. Capsule A-4
 Energy Input - 223 cal/g UO_2
 Reactor Period - 75 msec

FIGURE 2.2. Irradiated TREAT Capsules Showing Heat Affected Zircaloy-2 Cladding.

Heat Affected
Neg 469-356E

Unaffected
100X

a. Zircaloy Clad

Neg 469-356F

100X

b. Heat Affected Zircaloy-2 Cladding

FIGURE 2.3. *Photomicrograph of TREAT Capsule A-3
Zircaloy-2 Cladding.*

Unaffected Zone Heat Affected
 Neg 469-357E 100X
 a. Zircaloy-2 Clad

Neg 469-357F 100X
 b. Heated Affected Zircaloy-2 Cladding

FIGURE 2.4. Photomicrographs of TREAT Capsule A-4
 Zircaloy-2 Cladding (polished and etched)

Neg 469-356C

100X

a. Capsule A-3
(central region)

Neg 469-357C

100X

b. Capsule A-4

FIGURE 2.5. Transient Irradiated Capsule Fuel Showing
a) Recrystallization and b) Particle-to-Particle
Sintering.

discs at the ends of the fuel column, autoclaving the capsules as the last fabrication step, and centering the fuel column in the neutron flux.

A calibration capsule of the modified design was fabricated and shipped to TREAT for low power testing. After irradiation, the capsule will be disassembled and the activity of each pellet determined to evaluate the axial flux profile and assess the effectiveness of the hafnium discs in reducing flux peaking. Fabrication of five capsules (two vibrationally compacted and three pellet capsules) of the modified design for testing in TREAT is in progress.

References

1. R. L. Gulley, "Transient Tests on Thermal Reactor Oxide Fuels," Plutonium Utilization Program Technical Activities Quarterly Report, September, October, November, 1967, BNWL-654, Pacific Northwest Laboratory, Richland, Washington, pp. 2.19-2.20. February 1968.

3.0 REACTOR PHYSICS

THEORETICAL CORRELATION OF THE PRTR BATCH CORE EXPERIMENT

S. Matsuura* 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.

Results of the analysis of the PRCF and PRTR critical experiments were reported earlier.⁽¹¹⁾ This report contains results of a theory-experiment correlation of the worths of peripheral and internal fuel elements in the PRTR. It further contains an analysis of the effects of an inaccurate representation of PRCF core leakage and nonasymptotic spectra on k_{eff} and β/λ .

Correlation of PRCF Critical Experiments

A series of analytical correlations of the PRCF critical experiments to determine the feasibility of the present analytical methods have been reported.⁽¹¹⁾ The results showed

* On assignment to PNL from JAERI, Japan.

that the calculated effective multiplication factors were definitely larger than the measured values by amounts of 0.3 to 1.9% $\Delta k/k$. A study has been conducted to find the main causes of this trend, as well as to define the size of inevitable error (e.g., the error caused by manufacturing tolerances) in the correlation. The effects on effective multiplication factors due to the approximate prescriptions representing the measured system in the calculation, and the experimental uncertainties have been additionally studied. The most essential and difficult problem among them seemed to be the validity of cylindricalization and homogenization of the reactor cores, since the cores were actually highly heterogeneous assemblies with rod cluster fuels. There were two main problems to be considered for these processes of homogenization and cylindricalization:

- A. The cell calculation as the base of homogenization was made under the assumption of an infinitely repeated lattice and did not take into account the spatially transient spectrum at the core-reflector boundary. The fuel cells which were close to the reflector should have had a softer spectrum than the interior fuel cells because their cell spectrum was affected by the surrounding reflector. Similar considerations could be applied inversely for the peripheral reflector. The problem was to determine the transient spectrum effect on k_{eff} .
- B. In order to perform 1D calculations in the horizontal direction, the cores were approximated by cylinders. The equivalent core radii were determined in such a way that the cylindricalized cores had same volumes with the original assemblies of hexagonal cells. This cylindricalization of the core would have underestimated the effective area of the core surface because of the neglect of the unevenness of core surface. This would in turn, have caused an underestimation of the leakage rate.

Determining the size of the leakage correction for k_{eff} was another problem. These transient effects at core-reflector boundaries were considered by two methods. The first method concerned only A) and did not include B). It treated the core as consisting of three regions, e.g., the standard fuel cell region, the boundary region, and the asymptotic reflector region. The boundary region was a mixed region of peripheral fuel cells and a part of reflector. Since it was not obvious how much additional moderator should have been added to the fuel cell to form the boundary region, a set of parametric cell calculations was performed as a function of the moderator-to-fuel volume ratio. The results were used for the cylindricalized 1D-three-region reactor calculation with a varying boundary region thickness. These calculations were made for the 13-element core with HAMMER. The moderator-fuel volume ratios of the boundary region were 18.4, 24.6 and 30.1 in addition to 12.6 (standard cell). From the results of these calculations, the changes of spectrum, microscopic and macroscopic cross section and other cell constants due to the change of moderator-to-fuel volume ratio were observed. The calculated values of k_{eff} were 1.0175, 1.0220, 1.0229 and 1.0206 for the 12.6, 18.4, 24.6, and 30.1 moderator-to-fuel volume ratio cores respectively, while the measured k_{eff} value was 1.066. Although this method did not improve the calculated results, it indicated how the microscopic cross-sections in the fuel cell changed with the addition of moderator, and the results were used in the second method.

The second method considered problems A) and B). For problem A, the cores were considered to consist of four regions: the standard fuel cell region, the boundary fuel cell region, the boundary reflector region, and the asymptotic reflector region. The boundary fuel cell was a peripheral fuel cell adjacent to the reflector. On the other hand, the boundary

reflector region was a part of the reflector adjacent to the fuel cells. The boundary region had some intermediate spectrum and microscopic cross sections between those of standard fuel cell and those of a well-moderated cell. After several trials to determine the proper mixture of the microscopic cross-sections of the boundary regions, the simple average of the macroscopic cross sections of the asymptotic regions on either side of the boundary regions was taken as the microscopic cross sections of the boundary region. Because the maximum change of microscopic cross section between the two extreme spectrums was about 10%, and the trend of changes was monotonic, the difference of final results in k_{eff} due to the difference of microscopic cross-section estimating method was less than 2.5 mk, even in the most sensitive case of the 7-element core. The effect on k_{eff} due to the change in the microscopic cross sections was estimated by perturbation theory and the results are shown in Table 3.1. It shows that the transient spectrum effect on k_{eff} is less than about +5 mk, and it decreases with the increase of core size.

The effects on k_{eff} due to the additional leakage from the uneven surface of the core were estimated by considering the effective surface area for the assembly of hexagonal cells and by equating the effective surface-cylindricalized surface ratio and the ratio of the geometrical bucklings. The effective surface area of a core was obtained from the calculation of the average probability of neutron leakage from each surface of an assembly of hexagonal cells, and the critical equation of the point reactor model was used to obtain the effect on k_{eff} due to the change of geometrical buckling. The final results are shown in Table 3.1. It shows that the leakage effects are much larger than the spectrum effects, and the amount of the effect depends on the degree of unevenness of core surface and the size of the core. The preceding discussion suggests the following conclusions concerning the inherent uncertainties in the correlation.

TABLE 3.1. *Spectrum and Leakage Correction to k_{eff}*

PRCF Core	Calculation Method(a)	Calculated, k _{cal} (not corrected)	Experimental, k _{exp} ^(b)	Difference, Δk = k _{cal} - k _{exp}	Boundary Spectrum Correction, Δk_s	Surface Leakage Correction, Δk_g	Corrected, k' _{cal}	Difference, $\Delta k'$ = k' _{cal} - k _{exp}
7	1	1.0248		+0.014			1.0186	+0.008
	2	1.0226	1.011	+0.012	+0.0049	-0.0111	1.0164	+0.005
	3	1.0297		+0.019			1.0235	+0.013
13	1	1.0208		+0.015			0.9989	-0.007
	2	1.0193	1.006	+0.013	+0.0035	-0.0254	0.9974	-0.009
	3	1.0249		+0.019			1.0030	-0.003
28	1	1.0093		+0.002			1.0045	-0.003
	2	1.0010	1.007	+0.004	+0.0010	-0.0058	1.0062	-0.001
	3	1.0133		+0.006			1.0085	+0.002

(a) Calculation Method 1 = HAMMER 1D-4 groups, 2: HTH 1D-6 groups,
3 = EXTERMINATOR 2D-6 groups.

(b) These values were revised from previously reported values by the result of rechecking the experimental data of control rod worths.

- The present 1D method with measured vertical buckling is suitable to analyze a large core within an accuracy of 5 mk.
- For a small core, agreement within 10 mk between measurement and calculation cannot be obtained without appropriate correction for the core-reflector transient spectrum effect and the leakage effect due to the unevenness of core surface.
- No clear cut superiority is evident between HTH and HAMMER, and between energy group schemes.
- A number of problem areas in the calculational model remain; they are: the joining region of the thermal spectrum, the feasibility of the cylindrical approximation of the 19-rod cluster fuel element, and the relation between thermalization and the low lying plutonium resonance absorption.

The following results were found regarding error contributions due to unavoidable errors in the correlation.

- The effect of manufacturing tolerance (especially with the tolerance of fuel density $9.45 \pm 0.09 \text{ g/cm}^3$) is ± 0.53 mk.
- The effect of uncertainty of measured vertical buckling (0.4% in B^2) is about ± 1 mk.
- The effects of uncertainties of measured B^{10} concentration (± 0.05 wppm) are ± 1.6 , ± 1.0 and ± 0.6 mk for 7, 13, and 28 element cores respectively.
- The effect of uncertainties of measured moderator level: negligible.
- The uncertainties in the residual control rods worth are ± 0.4 , ± 0.2 , and ± 0.2 mk for 7, 13, and 28 element cores respectively.

From these results, it could be concluded that the final limitation of the accuracy of the analytical correlation with measured values are ± 1.3 to ± 2.0 mk for the PRCF cores.

The Effect in $\beta_{\text{eff}}/\lambda$ Due to the Transient Spectrum and the Change of Leakage Rate

The theoretically predicted values of $\beta_{\text{eff}}/\lambda$ which were previously reported differed significantly with the measured values. The theoretical values of β/λ were obtained by KINPAR.⁽¹⁰⁾ The error in the calculated value of β/λ due to the transient spectrum effect, was estimated by applying the changes in Σ_f to the expression for calculating β/λ used in KINPAR. The result for the 13-element core showed that the relative change of β/λ due to the transient spectrum was about 0.34%.

The neutron lifetime λ is essentially determined by the average neutron speed, the average cross-section and the nonleakage probability. The effect on λ due to the change of leakage rate of neutrons was estimated by the simple point reactor model. The resultant effect for the difference of leakage between the cylindricalized core and the assembly of the hexagonal cells was estimated to be about 0.3% to 1.0%. Therefore, the discrepancies between the measured and calculated values of β/λ cannot be explained in terms of the spectrum and leakage effects alone. These studies also indicate that the results of measurements of β/λ in small critical assemblies are much more applicable for evaluating calculational methods used for large reactors than are results of measurements of k_{eff} . This occurs because the spectrum and leakage effects in small critical loadings are large compared to the measurement inaccuracies in k_{eff} but small compared to the measurement inaccuracies in β/λ .

Correlation of the PRTR Critical Experiment

Worth of a Peripheral Fuel Element

The worth of a peripheral fuel element has been calculated using the 2DB code with the hexagonal geometry option. This

option performs an estimate of the peripheral fuel element worth for the 55-element core by representing the added fuel element as a homogenized, hexagonal fuel cell, rather than an added annulus as is done by HAMMER. The calculated worth is 1.45 mk. The experimental worth is 1.75 mk. The difference is -0.30 mk. (-17%)

Worth of an Interior Fuel Element

The calculated worth of fuel element 6069 in PC-1748 is 8.75 mk; the experimental worth is 7.4 mk. The difference is +1.35 mk (+15%).

Worth of the Creep Capsule

The calculated creep capsule worth is 0.3015 mk. The creep capsule worth has not been measured, however, such measurements are planned. The error in the calculation of fuel element worth falls in the range of 10-20% for most correlations so far performed: HAMMER (PRCF 8 el, 14 el, PRTR 31 el) and 2DB hex. (PRTR 55 el). The hexagonal description seems to have no defined superiority over the annular description for this correlation.

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EXPERIMENT-THEORY CORRELATIONS OF $\hat{\beta}/\ell$

V. O. Uotinen and S. R. Dwivedi

The ratio of effective delayed-neutron fraction to mean neutron lifetime ($\hat{\beta}/\ell$) has been calculated for several H_2O -moderated and reflected lattices using the computer code KINPAR.⁽¹⁾ The calculated ratios compare well with measured values.

Over the last few years the ratio $\hat{\beta}/\ell$ has been measured, using reactor noise techniques, in several light water

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lattices⁽²⁻⁵⁾ in the PRCF. The experimental technique consisted of obtaining a magnetic tape recording of the reactor noise, frequency-analyzing the tape with a multichannel frequency-analyzer, and least squares fitting an equation of the form:

$$y = A_1 + \frac{A_2}{\omega^2} \frac{\omega^2 + A_3}{\omega^2 + A_4}$$

to the data points. In the above equation y is the power per unit bandwidth, and ω is 2π times the frequency; A_1 is the white noise background, A_2 characterizes the amplitude of the plateau, $(A_3)^{1/2}$ is 2π times the low frequency breakpoint, and $(A_4)^{1/2}$ is $\hat{\beta}/\ell$.

Another method⁽⁶⁾ of deducing $\hat{\beta}/\ell$ has been to extrapolate to critical from subcritical measurements of the prompt neutron decay constant, α .

The computer code KINPAR⁽¹⁾ has been used to calculate values of $\hat{\beta}/\ell$ for these lattices. The code uses standard perturbation theory expressions for β and ℓ . KINPAR is linked to the one-dimensional, multigroup, diffusion theory code HFN, from which it obtains direct and adjoint fluxes for the perturbation theory expressions. The isotopic delayed neutron fractions used in KINPAR are derived from absolute yields (delayed neutrons/fission) reported by Keepin et al.⁽⁷⁾ and Cox⁽⁸⁾ and are consistent with values reported in ANL-5800. Velocities, prompt neutron spectra, and values of $v\sigma_f$ were obtained from the computer codes HRG and THERMOS for six neutron energy groups, with group boundaries at 10 MeV, 0.498 MeV, 0.235 MeV, 11.7 keV, 2.38 eV, 0.683 eV and 0.0 eV. All of the delayed neutrons were assumed to be in the energy group between 0.498 MeV and 0.235 MeV. For two of the lattices a four-group calculation was also done in which there was only one group between 10 MeV and 11.7 keV. This four-group scheme made no differentiation between the spectra of prompt and delayed neutrons.

Calculated and measured values of $\hat{\beta}/\lambda$ are compared in Table 3.2. Calculated values (6 G/sec) are consistently larger than measured values, but agree well within experimental uncertainties in all cases but one. (In this isolated case the measured value was deduced from an extrapolation to critical of subcritical measurements of α .) Calculated values of $\hat{\beta}/\lambda$, obtained with four energy groups, are approximately 10% smaller than values obtained with six groups. This indicates that the difference in age, τ , (and thus also the leakage, τB^2) between the prompt and delayed neutrons is significant.

Some improvement in the calculated $\hat{\beta}$ could be expected with a more exact representation of the delayed neutron spectrum, and some improvement in the calculation of λ could be expected with the use of multiple thermal groups. Nevertheless, for the uniform light water lattices investigated here, the six-group calculational scheme that has been used gives good agreement between measured and calculated values of $\hat{\beta}/\lambda$.

TABLE 3.2. Measured and Calculated Values of $\hat{\beta}/\lambda$, sec^{-1}

	Lattice Description	Measured		Calculated	
		6 Groups		4 Groups	
1.	2.35% enriched UO_2 (Ref. 5) 0.75 in. sq. lattice	159 $\pm$ 10	162.0	---	
2.	1.5 wt% PuO_2 - UO_2 (depl.) (Ref. 3) 0.71 in. hex. lattice	71 $\pm$ 6 (75 $\pm$ 8) ^b	75.6	---	
3.	UO_2 -2 wt% PuO_2 (8% ^{240}Pu) 0.85 in. hex. lattice (Ref. 2)	128 $\pm$ 8	128.0	115.0	
4.	UO_2 -2 wt% PuO_2 (^{240}Pu) 0.85 in. hex. lattice (Ref. 2)	133 $\pm$ 8	139.7	129.6	
5.	UO_2 -2 wt% PuO_2 (8% ^{240}Pu) 0.75 in. sq. lattice (Ref. 6)	135 $\pm$ 1 ^a	144.7	---	
6.	Al-1.8 wt% Pu (Ref. 4) 0.80 in. hex lattice	33 $\pm$ 3 (26 $\pm$ 3) ^b	34.8	--	

a. Deduced from an extrapolation of subcritical measurements of α .

b. Deduced from measured flux decay data obtained after insertion of negative reactivity.

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BATTELLE REVISED FLARE (BRF-1)*

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The computer code, FLARE⁽¹⁾ which was imported from Argonne's Code Center,⁽²⁾ has undergone considerable revision in order to make it operational on the UNIVAC 1108 computer and to expand its range of applicability to commercial nuclear power plants.

* Partially funded by Division of Production

Modifications Required to Make Operational on UNIVAC 1108

- The input data to FLARE is read by means of a generalized input routine which uses free field format and repeat and skip counters. In order to unpack this free field data, a subroutine BITA⁽³⁾ was added in which the Field function is used for partial word manipulation.
- In the original version all variables used in FLARE appear in Equivalence statements. This was a source of confusion to the optimizer portion of the UNIVAC 1108 compiler; hence, all equivalence statements were removed and replaced by named common blocks.

Modifications and Corrections to Increase Flexibility

- The code was expanded to $17 \times 17 \times 16$ nodes. The INCLUDE and PARAMETER options on the 1108 are used to facilitate changing the dimensions of the problem to any desired by the user within the limits of $I \times J \times K \leq 4624$.
- Subroutines EDT1 and PCH1 were modified to handle problems as large as $34 \times 34 \times 34$. (This was in preparation for a larger version of BRF.)
- Subroutine CASE was revised to output more alphanumeric information concerning the problem being run. Also, an independent case is indicated by I and a dependent case by D on the pertinent input cards.
- Subroutine SETUP was changed so that alternating directions can be used in the source iteration. The direction of iteration is controlled by the variable NALT, which is an input parameter which follows CHI on card type 02.
 - NALT = 0, forward direction only
 - = 1, alternate direction
- Subroutine CLEAR was added to clear common.

- Two of the four required scratch tapes were removed. The arrays E, V are written to drum and the refuel information remains on Scratch Tape D. Tape A is not needed.
- The function EX was replaced by the library function EXP.
- The array FLARC was removed since it was not needed.
- Subroutine EDT1 was modified for the 48-line pages used on the 1108. An error in EDT1 relative to calculation of average quantities in the 1/4 core option was corrected. In addition several mechanical errors in formats, etc. were corrected.
- All tape operations were modified to correspond to 1108 hardware.
- The transport kernels were modified to increase their flexibility. The kernels now in BRF are

$$WV_{ijk} = \left[(1-g_V) \frac{\sqrt{M^2}}{2r} + g_V \frac{M^2}{r^2} \right] \cdot GVN$$

$$WH_{ijk} = \left[(1-g_H) \frac{\sqrt{M^2}}{2r} + g_H \frac{M^2}{r^2} \right] \cdot GHN$$

or

$$WV_{ijk} = \frac{e^{-g_V a/M^2}}{GVN}$$

$$WH_{ijk} = \frac{e^{-g_H a/M^2}}{GHN}$$

where g_V , g_H , GVN , and GHN follow the variable CRNR on card type 01, respectively.

- The refuel option was modified so that I*J fuel elements can be described; the previous limit was 84. In addition, various errors in subroutine REFUEL were corrected.
- The following subroutines were deleted:
 - WTBT
 - RDBT

- SBITF
- SBITB
- RWD
- RWDLO
- BLKSIZ
- FSUP
- ISUP
- EX

Current Developments (BRF-2)

A basic limitation on BRF is that it can accept a maximum of $17 \times 17 \times 16 = 4624$ nodes, which is not sufficient to describe each fuel element in the large power reactors currently being built. By using the drum storage capability of the UNIVAC 1108, the BRF code is being extended to handle $34 \times 34 \times 34 = 39304$ nodes.

A preliminary version is operational and indicates that problem running time is not excessive (less than double the non-drum running time). Work is progressing on optimization of this version.

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THE INTERMEDIATE RESONANCE APPROXIMATION IN HRG

J. L. Carter

According to the procedure^(1,2) previously used in HRG for calculating the resonance integral of a resolved

resonance of an absorber, the resonance is classified as either wide or narrow relative to the energy loss of a neutron in a scattering collision with the absorber. All resonances are considered narrow relative to the energy loss in collisions with either admixed or external moderator atoms. These restrictions lead to simplified basic expressions involving functions which are evaluated from tabulated values. Revisions are currently being made in HRG which relax the restrictions on the absorber and admixed moderator collisions, yet give the same basic expressions with modified parameters. The revisions are particularly important for fuels with an admixed moderator because of the effect of the improved moderator treatment on the wide strong resonances at 6.7 eV in ^{238}U and at 1.06 eV in ^{240}Pu . To illustrate, resonance absorption decreases of about 3% in ^{238}Pu and 4.5% in ^{240}Pu and a reactivity increase of about 18 mk were calculated for a representative light water lattice with mixed oxide fuel containing 2 wt% PuO_2 with 8% ^{240}Pu . For comparison, a previously reported revision⁽³⁾ to HRG which allocates the contribution of a resonance to several fine groups gave, for the same lattice and fuel combination, absorption increases of about 5% in ^{238}U and 10% in ^{240}Pu and a reactivity decrease of about 25 mk. The effects of these two revisions provide a good illustration of the cancellation of errors in the former HRG resonance treatment. There is a net change, however, which results in improved theory-experiment correlation for a range of critical experiments.

To obtain the expressions used in the current revisions, we begin by observing that, for nonmetallic fuels, the results of Reference 1 for the narrow resonance (NR) and infinite mass (IM) approximations can be combined in the single expression:

$$I = \frac{\sigma_o \Gamma_Y}{E_o} \left[\beta_\lambda J(\xi, \beta_\lambda) + L(t_\lambda, \xi, \beta_\lambda) \right], \quad (1)$$

where

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

$$t_{\lambda} = \left\{ (1 - \lambda)\sigma_m + \lambda\sigma_p \right\} N\bar{x} ;$$

and $\lambda = 0, 1$ for the IM and NR approximations respectively. The other parameters are in common use and are further defined in References 1 or 2. The two tabulated functions are

$$J(\xi, \beta_{\lambda}) = \int_0^{\infty} \frac{\psi}{\psi + \beta_{\lambda}} dx , \quad (2)$$

$$L(t_{\lambda}, \xi, \beta_{\lambda}) = \int_0^{\infty} \frac{\psi^2}{\psi + \beta_{\lambda}} P_o \left\{ \frac{t_{\lambda}}{\beta_{\lambda}} (\psi + \beta_{\lambda}) \right\} dx , \quad (3)$$

where $\psi = \psi(\xi, x)$ is the Doppler broadening function and P_o is the escape probability from the fuel. The form of β_{λ} immediately suggests that an improvement would result if a value of λ intermediate to 0 and 1 were used. An investigation of this intermediate resonance (IR) problem in a homogeneous system was first described by Goldstein and Cohen⁽⁴⁾ who developed the basic iterative procedure for determining λ . Further work by Goldstein^(5,6) and others^(7,8) has established the validity of the method, has introduced heterogeneity through the Wigner rational approximation, has extended the method to include both admixed and external moderators by inserting additional parameters, and has improved the procedure for determining the parameters.

These results can be readily adapted to HRG, but only selected ones have been included at this time. To include the effect of the admixed moderator in Equation (1) it is

only necessary to make a slight change in the definitions of β_λ and t_λ . These parameters become

$$\beta_{\kappa\lambda} = \frac{(\kappa - \lambda)\sigma_m + \lambda\sigma_p}{\sigma_o} \frac{\Gamma}{\Gamma_\gamma + \lambda\Gamma_n} \quad (4)$$

$$t_{\kappa\lambda} = \{(\kappa - \lambda)\sigma_m + \lambda\sigma_p\} N\bar{\ell} \quad (5)$$

In present notation, and neglecting the interference between resonance and potential scattering, the transcendental equations which are solved to find κ and λ are

$$\lambda = 1 - \frac{\tan^{-1} X_{\kappa\lambda}}{X_{\kappa\lambda}} \quad (6)$$

$$\kappa = 1 - \frac{\tan^{-1} Y_{\kappa\lambda}}{Y_{\kappa\lambda}}, \quad (7)$$

where

$$X_{\kappa\lambda} = \frac{4A}{(A+1)^2} \frac{E_o}{\Gamma} \left(\frac{\beta'_{\kappa\lambda}}{1 + \beta'_{\kappa\lambda}} \right)^{1/2}$$

$$Y_{\kappa\lambda} = \frac{4A_m}{(A_m+1)^2} \frac{E_o}{\Gamma} \left(\frac{\beta'_{\kappa\lambda}}{1 + \beta'_{\kappa\lambda}} \right)^{1/2}$$

$$\beta'_{\kappa\lambda} = \frac{(\kappa - \lambda)\sigma_m + \lambda\sigma_p + \sigma_e}{\sigma_o} \frac{\Gamma}{\Gamma_\gamma + \lambda\Gamma_n}$$

$$\sigma_e = 1/\bar{\ell}N, \quad ,$$

and A and A_m are, respectively, the masses of the absorber and admixed moderator in units of the neutron mass. The steps in the HRG procedure are first to find κ and λ from Equations (6) and (7), then use these values in Equation (4) and (5) to

obtain the parameters to evaluate Equation (1). The one additional input number required is A_m .

The preceding discussion was confined to nonmetallic fuels; that is, $\sigma_m \neq 0$. For metallic fuels, $\sigma_m = 0$. In the NR approximation ($\lambda = 1$), both $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ are greater than zero and Equation (1) is still used by Adler et al.⁽¹⁾ In the IM approximation ($\lambda = 0$), however, both $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ equal zero. Under these conditions,

$$\lim_{\beta \rightarrow 0} \beta J(\xi, \beta) = 0$$

$$\lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} L(t, \xi, \beta) = L'(t'', \xi) = \int_0^\infty \psi P_0(t'', \psi) dx, \quad (8)$$

where

$$t'' = \sigma_0 N \bar{\ell} \frac{\Gamma_Y}{\Gamma},$$

and the subscripts on $\beta_{\kappa\lambda}$ and $t_{\kappa\lambda}$ have been suppressed for notational convenience. Adler et al., argue that in this limit the "volume" term, $\sigma_0 \Gamma_Y / \xi_0 \beta J(\xi, \beta)$, should approach the slowing down density in the fuel $2\sigma_{\text{pot}}/A$, rather than zero. For metallic fuels in the IM approximation, they replace the form of Equation (1) by

$$I = \frac{2\sigma_{\text{pot}}}{A} + \frac{\sigma_0 \Gamma_Y}{E_0} L'(t'', \xi) \quad (9)$$

and tabulate the L' -function separately. In the current revisions the $2\sigma_{\text{pot}}/A$ term is omitted; its small effect is already included to some extent in the IR formulation. Since the L' term is a limiting form of the L -function, Equation (1) can thus be used in all cases. Numerical problems occur, however, whenever the values of β and t from Equations (4) and (5) lie sufficiently far below the

ranges covered by the L-tables that the extrapolation procedures normally used in HRG are inadequate. To cover this situation, two additional extrapolation formulas have been introduced, one for the L-tables and one for the L'-tables. Both are based on the Wigner rational approximation. It can be shown that, using this approximation, Equation (3) becomes

$$L(t, \xi, \beta) = (\beta + \beta/t) J(\xi, \beta + \beta/t) - \beta J(\xi, \beta) . \quad (10)$$

The identity

$$L(t, \xi, \beta) = L(t_m, \xi, \beta) \times \frac{L(t, \xi, \beta)}{L(t_m, \xi, \beta)}$$

can be converted into an extrapolation formula for L by using Equation (10) to evaluate the ratio on the right hand side.

The result is

$$L(t, \xi, \beta) = L(t_m, \xi, \beta) \times \frac{(\beta + \beta/t) J(\xi, \beta + \beta/t) - \beta J(\xi, \beta)}{(\beta + \beta/t_m) J(\xi, \beta + \beta/t_m) - \beta J(\xi, \beta)} , \quad (11)$$

where t_m is the smallest tabulated value of t . Similarly, the identity

$$\begin{aligned} \beta J(\xi, \beta) + L(t, \xi, \beta) &= \lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} \{ \beta J(\xi, \beta) + L(t, \xi, \beta) \} \\ &\times \left(\frac{\beta J(\xi, \beta) + L(t, \xi, \beta)}{\lim_{\substack{\beta \rightarrow 0 \\ t \rightarrow 0}} \{ \beta J(\xi, \beta) + L(t, \xi, \beta) \}} \right) \end{aligned}$$

can be converted into an extrapolation formula for L'; the ratio is evaluated from Equation (10) and its limiting form, and the first factor is given by Equation (8). The result is

$$\beta J(\xi, \beta) + L(t, \xi, \beta) = L'(t'', \xi) \frac{(\beta + \beta/t) J(\xi, \beta + \beta/t)}{1/t'' J(\xi, 1/t'')} . \quad (12)$$

Equation (12) is used whenever both β and t are well below the ranges covered by the L-tables; this situation naturally arises for those strong, broad resonances in metallic fuels previously evaluated by Equation (9). If t is too small, Equation (11) is used.

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ANALYTICAL CORRELATIONS OF MEASURED POWER DISTRIBUTIONS

V. O. Uotinen, G. L. Gelhaus and S. R. Dwivedi

In the analysis of H₂O-moderated and reflected, clean, critical experiments, one generally assumes the reactor to be composed of two regions, core and reflector. Few group

cross sections for core and reflector are calculated, assuming that an infinite-medium spectrum applies in each region.

This two-region, infinite-medium model is generally adequate for the calculation of k_{eff} , but does not satisfactorily predict the power peaking near the core-reflector boundary. The reason for this is that in reality the spectrum near the boundary is modified, i.e., the spectrum in the core adjacent to the reflector is softer than the spectrum in an infinite medium of core material, and the spectrum in the reflector adjacent to the core is harder than the spectrum in an infinite medium of water. The two-region, infinite-medium model does not take into account this modification of the spectrum near the core-reflector boundary.

Calculations of power distributions have been performed using a modified model. The modification involves use of two reflector regions. The inner reflector region (one lattice unit thick, adjacent to the core) is represented by cross sections that are averaged over a spectrum characteristic of the core; the outer reflector region is represented by cross sections characteristic of an infinite H_2O medium.

This modified model, which partially accounts for the spectrum change near the boundary, has been applied to several lattices. Theory-experiment correlations of power distributions are improved in all cases by the use of this modified model. The type of improvement obtained is illustrated in Figure 3.1. The results shown in Figure 3.1 were obtained with the one-dimensional, diffusion-theory code HFN using four-group cross sections from calculations with the codes THERMOS and HRG. Similar improvement was observed when a modified reflector region was used in two-dimensional (x-y) calculations, with the code EXTERMINATOR-2, of power distributions measured in the CNEN-BNW experimental program.

FIGURE 3.1. Effects of Core-Reflector Constants on Radial Power Ratio

Further improvement would be expected if one were to also modify the cross sections of the outermost core region. It appears that appropriate cross sections for both modified regions (i.e., outer core and inner reflector) may be obtained from a cell calculation that contains an extra thickness (e.g., one mean free path) of moderator. This approach is being investigated and will be tested against clean lattice experiments and against experiments involving water holes, slots and crosses. The latter experiments more closely resemble actual power reactor configurations and thus provide more severe and more meaningful tests of the adequacy of the calculational model.

The present preliminary investigation has also pointed out differences between calculations with several standard diffusion theory codes, namely HFN, EXTERMINATOR-2 and 2DB. Power distributions calculated with these three codes are significantly different (Figure 3.2). Calculated values of k_{eff} also differ slightly for the three codes, and it is interesting to note that the discrepancy in power distribution is correlated with the discrepancy in k_{eff} .

MEASUREMENT OF THE TEMPERATURE COEFFICIENT OF REACTIVITY USING THE ENDOGENOUS PULSED SOURCE METHOD

J. B. Edgar

Introduction

The reactivity status of a critical assembly can be determined by observing the decay of spontaneous bursts of prompt neutrons. The measured quantity is the decay constant α , in the equation $n(t) = n_0 e^{-\alpha t} + \langle n \rangle$, which describes the time decay $n(t)$ of a burst of amplitude n_0 above the long-time averaged counting rate $\langle n \rangle$. This technique, called the endogenous pulsed source method,^(1,2) has been applied to a variety of experiments which required measurement of the reactivity status of a critical assembly.⁽³⁾

FIGURE 3.2. Comparison of Radial Power Ratios

This method has been applied recently to the measurement of the temperature coefficient of reactivity for a small core of 1/2-in. diameter rods of UO_2 -2 wt% PuO_2 (8% ^{240}Pu), arranged in a square pitch lattice with a center-to-center spacing of 0.75 in. and a moderator-to-fuel volume ratio of 1.55. A comprehensive series of measurements had previously been made in this core under the joint USAEC-CNEN program⁽⁴⁾ in the Plutonium Recycle Critical Facility (PRCF). This work represents an extension of that effort.

Experimental Details

The value of the prompt neutron decay constant at delayed critical, α_c , was determined for the core fueled with UO_2 -2 wt% PuO_2 rods by measuring values of α at several known levels of subcriticality and extrapolating to the value at delayed critical. This procedure was necessitated by the relatively large spontaneous fission rate in the plutonium-enriched fuel. The reactor was made critical with 319 fuel rods at an ambient temperature of 32 °C. With all controls withdrawn and fully reflected with H_2O on all boundaries, the loading had a measured excess reactivity of 15.6¢. The reactor was made subcritical by insertion of a calibrated control sheet located in the radial reflector. A plot of the measured inverse multiplication versus the measured value of α at each step was extrapolated to $1/M = 0$ to obtain the value of α at delayed critical. The results of these measurements are presented in Table 3.3.

At the beginning of the measurements to determine the variation of α with temperature, four additional fuel rods were removed from the periphery of the core, leaving a total of 315 rods. Based on prior determinations of peripheral fuel rod worth, the core was approximately 40¢ subcritical with all control mechanisms withdrawn and infinitely reflected. Measurements of the reactivity status were made at intervals of

approximately 5 °C, over the range from 30 to 75 °C, by determining the values of α at each temperature step. The results of these measurements are presented in Table 3.4 and in Figures 3.3 and 3.4.

TABLE 3.3. *Measurements Used in Extrapolation to Critical α*

Control Sheet S-2 Position, % Withdrawn	α (seconds) ⁻¹	Multipli- cation	Reactivity, dollars
46%	204.2 $\pm$ 1.5 (0.7%)	0.216	-0.517
42.25%	222.2 $\pm$ 1.8 (0.8%)	0.272	-0.651
39%	240.9 $\pm$ 1.9 (0.8%)	0.326	-0.780
36%	253.7 $\pm$ 2.0 (0.8%)	0.380	-0.909
33%	274.7 $\pm$ 2.8 (1.0%)	0.430	-1.029
30%	295.7 $\pm$ 5.9 (2.0%)	0.482	-1.154

Extrapolated Critical $\alpha = 135.1 \pm 1.0$ (seconds)⁻¹

Reactivity Calibration: Delayed critical to prompt critical
interval = 1\$ = 0.418 inverse
multiplication units

TABLE 3.4. *Measured Values of Alpha and Reactivity
as a Function of Temperature*

Temperature, °C	α (seconds) ⁻¹	Reactivity, dollars
30.48	202.7 $\pm$ 1.7 (0.8%)	-0.500 $\pm$ 0.015 (3.0%)
37.42	217.8 $\pm$ 2.2 (1.0%)	-0.612 $\pm$ 0.018 (2.9%)
41.37	229.6 $\pm$ 1.8 (0.8%)	-0.700 $\pm$ 0.016 (2.3%)
47.18	247.2 $\pm$ 2.3 (0.9%)	-0.830 $\pm$ 0.020 (2.4%)
50.09	279.3 $\pm$ 2.4 (0.9%)	-1.067 $\pm$ 0.021 (2.0%)
52.74	275.0 $\pm$ 2.3 (0.8%)	-1.035 $\pm$ 0.022 (2.1%)
55.56	285.1 $\pm$ 2.6 (0.9%)	-1.100 $\pm$ 0.022 (2.0%)
58.32	296.2 $\pm$ 3.3 (1.1%)	-1.192 $\pm$ 0.026 (2.2%)
63.81	317.0 $\pm$ 3.8 (1.2%)	-1.346 $\pm$ 0.031 (2.3%)
66.57	343.6 $\pm$ 3.8 (1.6%)	-1.558 $\pm$ 0.031 (2.0%)
69.48	338.6 $\pm$ 4.6 (1.4%)	-1.506 $\pm$ 0.036 (2.4%)
71.82	358.4 $\pm$ 5.3 (1.5%)	-1.653 $\pm$ 0.042 (2.5%)
74.03	350.0 $\pm$ 5.3 (1.5%)	-1.591 $\pm$ 0.041 (2.6%)

FIGURE 3.3. Variation of Alpha with Moderator Temperature UO_2 -2 wt% (8% ^{240}Pu) Rods in 0.75 in. Square Pitch Lattice Moderator/Fuel Volume Ratio = 1.55

FIGURE 3.4. Variation of Reactivity with Moderator Temperature UO_2 -2 wt% (8% ^{240}Pu) Rods in 0.75 in. Square Pitch Lattice Moderator/Fuel Volume Ratio = 1.55

Due to the extended period of time (up to three hours) required to accumulate experimental data for alpha measurements, thermal stratification of the moderator was anticipated as a problem. Also, at the higher temperatures, the moderator temperature loss was expected to be significant over the period of the measurement. Modifications were made to the reactor hardware to permit continuous mixing and heating of the moderator to assure that the moderator temperature remained constant and uniform over the time span of measurement.

Discussion of Analysis

The value of α was determined for each measurement by fitting the observed prompt neutron die-away to the equation $n(t) = n_0 e^{-\alpha t} + \langle n \rangle$ using Program LEARN.⁽⁵⁾

Alpha critical (α_c) had been determined to be 135.1 ± 1.0 seconds⁻¹. It was used to convert the measured α values obtained during the temperature experiment to levels of subcriticality in dollars (\$) by the following relationship:

$$\text{Level of subcriticality (\$)} = \frac{\alpha_c - \alpha}{\alpha_c}$$

This alpha-reactivity relationship has been previously demonstrated by Pacilio⁽⁶⁾ to be linear up to about 7 dollars subcritical. This conversion relationship is not strictly valid for these measurements since the neutron lifetime in these cores increases slightly with increasing moderator temperature. This effect is estimated to reduce the measured values of reactivity by about 2%. No corrections have been made to the measured data for the change in neutron lifetime in the results reported here.

The α and reactivity versus temperature data listed in Table 3.4 have been fitted with a linear function, as shown in Figures 3.3 and 3.4, respectively. The coefficients of the fitted function and their one standard deviation uncertainties are listed in Table 3.5.

TABLE 3.5. *Coefficients of the Linear Relationship*

Equation	A	B
$\alpha(T) = A + BT$	82.5 ± 9.0	3.70 ± 0.18
$\rho(T) = A + BT$	$(38.5 \pm 6.5) \times 10^2$	$-(2.73 \pm 0.13) \times 10^{-2}$

Differentiation of the fitted relationships yield the temperature coefficients of α and reactivity, as shown below.

$$\frac{d\alpha}{dT} = 3.70 \pm 0.18 \text{ alpha units}/^\circ\text{C}$$

$$\frac{d\rho}{dT} = -(2.73 \pm 0.13) \times 10^{-2} \text{ } \$/^\circ\text{C}$$

Using a value of 3.74 mk per dollar, the temperature coefficient of reactivity is $-1.02 \times 10^{-4} \frac{\Delta k}{k}/^\circ\text{C}$.

The ability of the endogenous pulsed source method to determine the reactivity status of a subcritical assembly has a distinct advantage over other techniques for measuring changes in reactivity status with changes in temperature. The advantage is that the core loading (number of rods) and geometry remains unchanged throughout the full temperature range of the experiments. Thus, these measurements are much more amenable to direct calculation than are measurements which require a series of stepwise changes in the core loading.

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THE SHAPE OF PEAKS IN GAMMA-RAY SPECTRA FROM Ge(Li) DETECTORS

D. A. Kottwitz

The accurate, quantitative analysis of complex gamma-ray spectra obtained from Ge(Li) detectors depends on a quantitative understanding of the basic shape of a "full-energy" peak and its immediate vicinity. This information is developed best from an empirical study of the intense, isolated peaks provided by standard sources. During this quarter, we have started work on this problem, using the nonlinear least-squares computer program LEARN-I-A.

The shape of a full-energy peak depends generally on processes occurring in the detector crystal, amplifier circuitry, and the analogue-to-digital converter. Since it also depends on the operating conditions, e.g., voltages and time constants, great variety is possible. A typical example obtained at BNW (with features exaggerated for clarity) is indicated below by the solid line. There are three notable characteristics: (1) the high-energy side is well represented by a Gaussian function, (2) the low-energy side has a broad tail, and (3) there is long, flat shelf at energies below the tail. The asymmetry generally cannot be removed by subtracting a resolution-broadened shelf (shown by a dashed line). The purpose of the present effort is to find a simple but accurate representation of this asymmetry, which will be suitable for

incorporation into a least-squares spectrum-analysis computer program. The results obtained so far are too preliminary to be reported.

The methods being used to study the gamma-peak shape problem are based on a more complete and correct treatment of the problem of holding fixed any subset of parameters in the non-linear least-squares program LEARN. The original version of LEARN⁽¹⁾ provided only limited capability of doing this directly. The present scheme is based on a systematic expansion and contraction of the set of free parameters (and associated quantities). It is implemented by adding four more-or-less standard batches of FORTRAN cards to subroutines PUTIN, THEORY and PUTOUT. The scheme has been designed for adoption in existing LEARN packages with a minimum of reprogramming. A write-up of the method together with sample batches of cards is being prepared for distribution to any interested users of program LEARN.

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THE EFFECT OF RECYCLED URANIUM ON THE ^{238}Pu AND ^{236}Pu CONTENT
IN RECYCLED PLUTONIUM FUEL

E. T. Merrill and E. A. Schnaible

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/ton and about 7% high at 23,000, well within experimental uncertainty. The composition of plutonium produced in PWR's (Diablo Canyon design) and BWR's (TVA design) was then computed and the results are presented in Table 3.6.

TABLE 3.6. ^{236}Pu and ^{238}Pu Content of LWR-Produced
Plutonium 6 Months After Discharge

Reactor	Fuel	Exposure	ppm ^{236}Pu	% ^{238}Pu
BWR	Enriched U	27,500	0.057	1.4
BWR	Enriched U	32,000	0.081	1.4
PWR	Pu + Natural U			
	1st recycle Pu	32,000	0.044	2.2
	2nd recycle Pu	32,000	0.036	2.7
	3rd recycle Pu	32,000	0.030	3.1
PWR	Pu + Recovered U			
	1st recycle	32,000	0.082	2.3
	2nd recycle	32,000	0.069	2.8
	3rd recycle	32,000	0.060	3.2

In the first three 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 6 months after discharge from the reactor. The plutonium isotopes are then assumed to decay for 6 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/ton. Plutonium for the second

recycle is obtained by reprocessing the fuel from the first recycle after an exposure of 32,000 MWD/ton 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 2nd recycle fuel and adding plutonium recovered from spent, slightly-enriched uranium fuel.

In the second set of three plutonium recycle cases, PuO_2 is mixed with recovered UO_2 as described above. The recovered UO_2 would be obtained from the chemical separation plant and was assumed to contain 0.92% ^{235}U and 0.4% ^{236}U . The higher ^{236}Pu and ^{238}Pu buildup is due to the ^{236}U recycled.

BWR VOID FRACTION FLUCTUATION STUDY

E. T. Merrill and E. A. Schnaible

The ALTHAEA burnup code was used to study the effects of void fluctuations in a boiling water reactor on the reactivity and isotopic concentrations during the burnup life of a fuel element. The simulation was accomplished by assuming a high, a medium, and a low void fraction where the medium void fraction is some arbitrary void fraction and the high and low void fractions are of equal incremental deviation from the medium. Each burnup contained a combination of all three void fractions in which each void fraction passed through 9000 MWd/MT for a total burnup of 27000 MWd/MT. Six combinations are possible which are related to a single run comparable to a steady medium void fraction burnup.

This study includes both a uranium fuel and a natural uranium-plutonium enriched fuel. The uranium fuel study includes a void fluctuation of 38% and 19%. The natural uranium-plutonium enriched study includes a void fluctuation of 19% and 10%. The data has been tabled for convenience of analysis.

Table 3.7 presents the percent error of the isotopic concentrations and reactivities relative to a steady medium void fraction burnup. The letters L, M, and H designate the low, medium, and high void fractions, respectively. The order of the letters from top to bottom is the order the void fractions appear during burnup each covering 9000 MWd/MT for a total burnup of 27000 MWd/MT. The last column represents the percent error of the average of the six void fraction combinations. The first column designates the uranium fuel or natural uranium plutonium enriched fuel. The next column indicates the percent void fraction deviation from the medium void fraction.

TABLE 3.7. *Percent Error^(a) for Concentrations and Reactivities Relative to a Constant Void Burnup*

MWd/MT			Void Fraction Combinations						Average
0-9000 9000-18000 18000-27000			M L H	L M H	H L M	L H M	H M L	M H L	
Fuel	% Void Dev.								
²³⁵ U	U	19	0.27	0.68	-2.77	-1.73	-4.10	-3.47	-1.89
	U	9.50	0.41	0.62	-0.67	-0.27	-1.34	-1.15	-0.40
	U-Pu	9.50	0.44	0.59	-0.40	-0.07	-0.91	-0.77	-0.18
	U-Pu	5	0.22	0.30	-0.11	-0.07	-0.33	-0.22	0.00
²³⁹ Pu	U	19	-10.45	-11.88	0.77	-4.06	6.73	3.31	-3.09
	U	9.50	-3.62	-4.40	0.66	-1.30	3.31	2.12	-0.61
	U-Pu	9.50	-2.24	-2.80	0.17	-1.25	1.46	0.60	-0.70
	U-Pu	5	-0.66	-0.84	0.17	-0.36	0.60	0.35	-0.13
²⁴⁰ Pu	U	19	4.83	4.75	-1.00	-1.47	-4.49	-4.71	-0.52
	U	9.50	1.86	1.76	-0.10	-0.38	-1.70	-2.30	-0.10
	U-Pu	9.50	1.27	1.65	-0.53	-0.44	-1.47	-0.95	0.05
	U-Pu	5	0.37	0.52	-0.22	0.14	-0.59	-0.37	-0.03
²⁴¹ Pu	U	19	-9.56	-10.11	-0.87	-2.57	5.04	3.86	-2.76
	U	9.50	-5.13	-3.49	0.00	-0.68	2.40	2.00	-0.58
	U-Pu	9.50	-2.33	-2.83	0.35	-0.87	2.02	1.33	-0.41
	U-Pu	5	-0.96	-1.10	-0.20	-0.55	0.29	0.12	-0.06
²⁴² Pu	U	19	4.50	5.45	-1.69	0.58	-4.40	-2.79	0.00
	U	9.50	1.75	2.35	-0.57	0.58	-1.69	-1.14	0.29
	U-Pu	9.50	0.93	1.22	-0.05	0.56	-0.55	-0.23	0.32
	U-Pu	5	0.28	0.37	0.00	0.18	-0.18	-0.09	0.09
²³⁹ Pu + ²⁴¹ Pu	U	19	-10.30	-11.58	0.48	-3.81	6.43	3.41	-3.03
	U	9.50	-3.54	-4.24	0.55	-1.19	3.15	2.10	-0.61
	U-Pu	9.50	-2.27	-2.81	0.23	-1.13	1.63	0.83	0.61
	U-Pu	5	-0.64	-0.81	0.16	-0.31	0.62	0.39	-0.11
k _∞	U	19	-3.22	-3.53	-0.03	-0.99	1.82	1.13	-0.84
	U	9.50	-1.07	-1.24	0.09	-0.30	0.91	0.68	-0.16
	U-Pu	9.50	-1.14	-1.38	0.17	-0.42	0.88	0.53	-0.23
	U-Pu	5	-0.64	-0.81	0.16	-0.31	0.62	0.39	-0.11

(a) Percent error = $\frac{\text{Steady void} - \text{Variable void}}{\text{Variable Void}} \times 100$

For a 38% fluctuation in void fraction the error for a uranium fueled reactor ranged from -4.1% to 0.7% for the ^{235}U concentration with an average error of -1.9%, and from -11.9% to 6.7% for the ^{239}Pu concentration with an average error of -3.1%. At the same time, the reactivity error ranged from -3.5% to 1.8% with an average error of -0.8%.

If the void fraction fluctuation is only half as much; i.e., 19%, the error of the ^{235}U concentration will be reduced to a range from -1.3% to 0.5% with an average error of -0.4% and the ^{239}Pu concentration will be reduced to a range from -4.4% to 3.3% with an average error of -0.6%. The reactivity error ranged from -1.2% to 0.9% with an average of 0.2%.

The interpretation of the average error can be misleading since it appears that the error can vary considerably depending on whether the burnup starts or ends with a low, medium, or high void fraction. The average here as determined from the six void fraction combinations could signify many and rapid void fluctuations during burnup.

For the natural uranium-plutonium enriched reactor in which the void fraction fluctuates 19%, the error of the ^{235}U concentration is obviously less. It ranged from -0.9% to 0.6% with an average error of -0.2%. The error for the ^{239}Pu concentration, also somewhat less, ranged from -2.8% to 1.5% with an average error of -0.7%. The error of the reactivity in this system ranged from -1.4% to 0.9% with an average reactivity error of 0.2%.

For a further study of the natural uranium-plutonium enriched system the void fraction fluctuation was cut in half to 10%; i.e., a plus or minus 5% from some arbitrary medium void fraction. This reduced the error of the major isotopes to less than 1%. The ^{235}U concentration ranged from -0.3% to 0.3% and the ^{239}Pu concentration ranged from -0.8% to 0.6% with an average error of -0.1%.

The FLARE code used for reactor fuel cycle analysis calculations uses the average void history to determine the isotopic concentrations and reactivity at any point in time. The ALTHAEA burnup study here indicates that the void fluctuations would have to be less than 10% in order to maintain an error of less than 1% in reactivity and fuel composition. The method of study simulates the slow variations of the void that a particular module experiences as the control rods are shifted during the burnup period. The high or low void depends, of course, on the location of the control rods and the distribution of the heat; the order of the high and low void fraction during burnup will depend upon the location of the module in relation to the position of the control rods. The errors seem to be strongly dependent upon the void fraction history of a particular module. Whether averaging can adequately represent the void fluctuations is debatable. However, if the FLARE method is acceptable for analysis of enriched uranium systems, it should also be acceptable for plutonium-fueled systems in that the percent error is of the same magnitude in the two systems.

4.0 OPERATIONS

PLUTONIUM RECYCLE TEST REACTOR OPERATION

Operating Experience

K. L. Young

The reactor remained shut down for the report period in a continuation of the outage started on July 14, 1968. The lengthy outage was prompted by 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.

Resumption of reactor operation was further delayed by discovery of casting defects in several of the primary system valves. Corrective action consisted of repairing the defect in the primary pump no. 1 check valve (P-1) and replacing the 14-in. gate valve (P-4) on the reactor outlet piping. Current activities are directed at resumption of reactor operation.

Plant Engineering

D. L. Cornell

Radiographic examinations of the 8-in. 304 stainless steel, cast, check valve (P-1) had revealed a defect on the outside surface of the inlet transition to the valve body. Dye penetrant examination also revealed two more indications in the same general area. The defects were removed by grinding and repaired by welding, using a gas tungsten arc welding procedure. Dimensions of the ground-out defect areas were as follows: defect A, 3-7/8 in. long x 9/16 in. deep x 13/16 in. wide, tapered to 3/16 in. at the bottom of the groove; defect B, 4-3/16 in. long x 1-1/32 in. deep x 15/16 in. wide, tapered to 1/8 in. at the bottom of the groove; defect C, 2-1/2 in. long x 5/32 in. deep x 3/8 in. wide, tapered to 3/16 in. at the bottom of the groove.

Final measurements of the valve seat showed no distortion resulting from the weld repairs. Radiographic and liquid penetrant testing techniques were used throughout the weld repairs, and final test results met applicable code requirements.

Replacement of the original P-4 valve with a new 14 in., 304 stainless steel, forged, valve (WP-4) was completed except for final acceptance tests. All certifications were reviewed during the quarter. Pipe stubs were welded to the valve body to meet the length dimension of the original valve, and the valve was then welded into the system between the top ring header and the primary heat exchanger (HX-1). Some difficulty was experienced in moving HX-1 the required amount to compensate for weld shrinkage. However, this difficulty was overcome and final piping measurements, taken after welding was completed, showed the pipe length between HX-1 and the top ring header to be within 1/32 in. of its original length. Rigid quality assurance and maintenance procedures were followed throughout the replacement program and the craft work was judged to be excellent. Final radiograph and liquid penetrant tests have been completed successfully and the valve was reassembled in preparation for hydrostatic tests, flow tests, and other final acceptance testing.

Process Technology

R. H. Purcell

Evaluations of Reactor Systems and Components

The inside surfaces of the primary system main 14-in. diameter line and the inlet baffle plate to the steam generator (HX-1) were inspected after the P-4 valve had been removed from the system. Areas inspected included about 9 ft of upstream line and a small part of the inside surface of the upper ring header, about 2-1/2 ft of downstream line, and the HX-1 inlet baffle plate. The entire baffle plate surface and adjacent inlet plenum surfaces were inspected with a mirror. Part of the ends of about 10 tubes and the adjacent tube sheet surface were visible

through holes in the baffle plate. All surfaces were either a dull metallic lustre or a smooth bluish-red color, typical of clean stainless steel exposed to high temperature water. The areas visible for inspection were clean and free of any debris. Maximum radiation level inside the piping was 240 mrad/hr. Maximum radioactivity of material removed from the inside pipe walls was 48 mrad/hr. No alpha (i.e., no plutonium) radioactivity was detected.

General Safeguards Activities

- Criticality safety specifications were issued to cover miscellaneous fuel additions to the storage basin area during the quarter. These specifications allow coverage for fuel types which contain ≤ 230 g total Pu + ^{235}U of irradiated and unirradiated fuel, respectively.
- Revision of the operating standards spot check audit system was completed.
- PRTR reactor technician qualification examinations were reviewed and revised to be compatible with the new reactor technician training manual.
- A series of 3 hr sessions on reactor physics were held as a part of the reactor operations retraining program. General aspects of reactor physics were covered with emphasis on problems peculiar to the PRTR and the Batch Core Experiment.
- A study of possibilities of and techniques for leakage rate testing of the air lock doors was completed and transmitted to the AEC. The study concluded that testing was possible but unwarranted due to favorable past experience with the air lock doors, the ability to monitor air-lock door-seal pressure during reactor operation, and the established preventive maintenance program on the air-lock door seals.
- The annual criticality safety appraisal was conducted by Nuclear Safety during May. The appraisal included discussions of Criticality Safety Specifications, training

programs, auditing, and compliance with posting and labeling requirements. A visual inspection of the storage basin and records was also made. No violations of existing Criticality Safety Specifications or Criticality Safety Procedures were noted.

5.0 PUBLICATIONS AND PRESENTATIONS

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