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**PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER, 1967, JANUARY, FEBRUARY, 1968**

STAFF OF BATTELLE-NORTHWEST

APRIL 1968

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**AEC RESEARCH &
DEVELOPMENT REPORT**

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PACIFIC NORTHWEST LABORATORY

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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT

December, 1967, January, February, 1968

By

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

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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER, 1967, JANUARY, FEBRUARY, 1968

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 present peak burnup of HPD elements in the PRTR is 6400 MWd/tonne. Other pre-batch core elements have attained peak burnups of over 16,000 MWd/tonne. The entire PRTR loading of 81 elements was recently examined, and their condition is excellent. Modest amounts of crud are forming on the element surfaces in the neutral pH coolant. Post-irradiation examination results indicate that the irradiation performance of both hot-pressed and cold-pressed and sintered $\text{UO}_2\text{-PuO}_2$ fuel rods is excellent.

Measurements on high exposure plutonium with portable laboratory instruments show considerable variation in dose rate with different geometric arrays, particularly at short distances from small surface area sources. Personnel exposure rates were significantly reduced during Phoenix fuel plate fabrication with high-exposure plutonium by the use of local shielding and modified handling procedures.

Capabilities at the Los Alamos Scientific Laboratory are being utilized to study the radial fission product distribution in irradiated $\text{UO}_2\text{-PuO}_2$ PRTR fuel rods. A continuous precision gamma scanning technique will be used to determine the

diametral distribution of various fission products and fissile elements in a $\text{UO}_2\text{-PuO}_2$ fuel rod irradiated in PRTR.

Eight fuel rods containing UO_2 (4.8% ^{235}U) pellets were fabricated and shipped to ANL for irradiation testing in TREAT. Seven similar vibrationally compacted fuel rods are being fabricated. These rods will be irradiated in TREAT to compare their behavior.

An instrumented $\text{ThO}_2\text{-5 wt% PuO}_2$ fuel element was irradiated in PRTR at peak powers of 7.4 kW/ft to a peak burnup of 450 MWd/tonne.

Fission product migration studies show that ruthenium migrated extensively during irradiation and that the high and low concentrations correspond with the dark and light regions on the beta-gamma autoradiograph, respectively.

Irradiation of an experimental fuel element containing low density ($\sim 71\%$ TD) mixed-oxide pellets showed that initially nonsintered pellets recrystallized into a coherent, relatively high-density fuel column having a large central void. Calculations indicate that in-reactor modification of the fuel improves the thermal characteristics so that heat ratings as high as 25 kW/ft could be attained without fuel melting.

The results of $\text{UO}_2\text{-PuO}_2$ capsule irradiations are being analyzed. Measured $^{243}\text{Am}/^{241}\text{Am}$ ratios used in burnup analysis are lower than calculated values. Possible sources of error are under study.

Irradiation of four vibrationally compacted mixed-oxide fuel rods instrumented to measure internal gas pressure continues in PRTR. Peak burnups attained in each set of two rods are 6000 and 3600 MWd/tonne. The pressure in the high and low burn-up fuel rods continues to increase with exposure at approximately constant rates.

The first irradiation of the 8-rod FERTF Test Element was conducted in the FERTF at a peak power generation of 18 kW/ft. Three fuel rods were removed from the assembly for examination without difficulty.

Excess CVTR fuel assemblies are being considered for driver fuel for continued irradiation of 12 to 19 PRTR-HPD fuel assemblies to peak burnups in the range of 40,000 to 50,000 MWd/tonne. One CVTR assembly was modified for irradiation testing in a PRTR fringe position.

Information from laboratory experiments with irradiated vipac, incrementally loaded, and hot-pressed pellet $\text{UO}_2\text{-PuO}_2$ fuel material in an oxalic-peroxide-gluconic (OPG) decontamination solution revealed that about 80% of the fuel was dissolved in 5 hr. Analytical measurements are being made to determine if there was a preferential dissolution of the UO_2 .

REACTOR PHYSICS

Approach-to-critical experiments are in progress in the Critical Approach Facility (CAF) using $\text{UO}_2\text{-4 wt% PuO}_2$ fuel rods in an H_2O moderator. Measurements have been completed in three hexagonal lattices (0.80, 1.386, and 1.60 in.) which cover the range of fuel-to-moderator ratios of interest. Agreement between the pre-experiment calculations and the experimental results is excellent.

Irradiated fuel elements from three different reactors have been examined during the past quarter using the PNL Gamma Scan Facility. The fuels studied included: $\text{UO}_2\text{-1.5 wt% PuO}_2$ irradiated in EBWR; $\text{UO}_2\text{-0.5 wt% PuO}_2$ low-density test fuel rods irradiated in ETR; and 19-rod clusters of $\text{UO}_2\text{-2 wt% PuO}_2$ irradiated in the Batch Core Experiment in PRTR.

The second burnup step in the Batch Core Experiment in PRTR has been completed, with an average exposure on the 55-element batch loading of 3630 MWd/MTU. An extrapolation of the available data predicts an average exposure on the 55-element batch loading of approximately 12,000 MWd/MTU at the end of the reactivity lifetime of the core.

Experiments are continuing at the Plutonium Recycle Critical Facility under the cooperative program between the USAEC and the Italian CNEN to study the problems of placing plutonium-enriched fuels in boiling

water reactors. The measurements have included power sharing and power distributions in a variety of heterogeneous core configurations that contained both enriched UO_2 fuel and UO_2 - PuO_2 mixtures. Analytical studies are being conducted concurrent with the experiments so that the degree of complexity needed in the calculational models to obtain adequate agreement between calculation and experiment can be determined.

A method has been developed to compare the results from measurements of large negative reactivities made using the endogenous pulsed source technique with the results from two-dimensional diffusion theory calculations which include control rod models. The analysis method will be applied to measurements planned for the PRCF.

A systematic study has been made of the importance of various assumptions in calculated and measured temperature coefficients.

Analysis of measurements performed in the PCTR light water tank utilizing a uniform lattice of UO_2 -0.86 wt% PuO_2 rods in a water moderated 1.0 in. square pitch has been completed. Results of the determination of k_∞ by (a) the null reactivity technique, and (b) measurement of the adjoint weighted excess neutron production cross section $\langle \Sigma_p \rangle$ in unpoisoned lattice are presented. Values of k_∞ determined by the two independent methods mentioned above were 1.120 ± 0.005 and 1.127 ± 0.006 , respectively. Relative reaction rate measurements in the unpoisoned lattice indicate that the thermal disadvantage factor is 1.249 ± 0.009 compared to the calculated THERMOS

value of 1.282. The measured adjoint weighted excess neutron production cross section for this lattice is $7.8 \pm 0.3 \times 10^{-3} \text{ cm}^{-1}$.

The third generation one-dimensional burnup code ZODIAC G is conditionally operable on the Univac 1108.

A method of representing resonance parameters averaged over an energy range has been derived. This averaged resonance peak height is shown to be a function of concentration and not constant. A number of simplified cases have been investigated to gain insight into the behavior of these averaged resonances in more complicated cases. This has immediate application to the ALTHAEA⁽¹⁾ code where heretofore some of these parameters were derived by interference from comparing calculated burnup data to experimental data.

PLUTONIUM UTILIZATION STUDIES

A study has been made of the costs of fabricating fuel assemblies for pressurized water reactors and boiling water reactors, using both enriched UO_2 and UO_2 - PuO_2 mixtures. The study included the case of fuel assemblies that contained rods of different enrichments. An abstract of a paper describing this work and the effect on plutonium values has been submitted for presentation at the May 1968 IAEA meeting on Economics of Reactor Fuels.

PRTR OPERATION

A continuing satisfactory condition of aluminum surfaces in the PRTR moderator system is indicated by

examinations of corrosion coupons. Boron deposition appears to have decreased and remains at an acceptably low value.

Shutdown radiation measurements on the PRTR primary system show that radiation levels with "neutral" pH coolant operation are substantially the same as those found after comparable periods of operation with pH 10 coolant. Additional data and time are required to establish long term radioactivity buildup rates with "neutral" pH reactor operation.

The PRTR operated to 63 MW to produce 2074 MWd during December, 1967, and January and February, 1968.

The PRTR has accumulated 7300 MWd since beginning the batch core experiment with the central 55 fuel elements (batch core) accumulating 5870 MWd.

The interim critical tests for the second burnup step of the batch core were conducted when the reactor achieved 7300 MWd. The preliminary results showed that the excess reactivity of the core decreased from an initial 263 mk to a value of 191 mk. Batch core life was extrapolated to

23,000 MWd, confirming previous forecasts.

The first test of an 8-rod element in the Fuel Element Rupture Test Facility was concluded. A specific rod power of 18 kW/ft was obtained.

All fuel elements were removed from the PRTR on February 7 to conduct the annual containment vessel leak rate check. The first check of the leak rate from the containment vessel was greater than the allowable 0.678% at 10 psig. Leak repair and retesting are in progress at month-end. During this long outage, the safety system modifications were completed to satisfy new reactor safety criteria.

All batch core fuel elements and 38 process tubes were examined visually. Process tube examination did not show any significant increase in the depth or number of fret marks since the previous examination. A few fuel elements required rebanding, but all were in satisfactory condition for continued irradiation.

Corrosion surveillance and radiation surveillance programs continued during the quarter. No significant changes were observed.

2.0 FUELS DEVELOPMENT

HIGH POWER DENSITY FUEL IRRADIATIONS IN PRTR

M. D. Freshley and T. B. Burley

Present Core Status

High Power Density (HPD) fuel elements are being irradiated in the PRTR as part of the Batch Core Experiment - a controlled large-scale burnup experiment. The experiment involves operation of the PRTR loaded with 66, 19-rod cluster UO_2 - PuO_2 HPD fuel elements. Statistically significant fuel performance data are being obtained on UO_2 - PuO_2 fuels operating at peak linear rod powers in the range of 18 to 20 kW/ft with maximum fuel temperatures near melting. The burnup status of the present core loading is summarized in Table 2.1.

The present peak burnup of HPD elements is the batch core is approximately 6400 MWd/tonne. Nondestructive and destructive examination of selected fuel rods continues to indicate excellent performance. No fuel failures have occurred thus far in the program. A core average fuel burnup of approximately 10,000 MWd/tonne with peak burnups in the range of 20,000 MWd/tonne will be achieved during the Batch Core Experiment. Because it is essential that UO_2 - PuO_2 fuels meet the performance criteria of presently designed PWR-BWR fuels, irradiation of 12 to 19 selected UO_2 - PuO_2 HPD fuel assemblies will be continued after the Batch Core Experiment in PRTR to peak burnups in the range of 40,000 to 45,000

MWd/tonne under thermal conditions at least comparable to projected light water reactors now being committed.

Irradiation of selected elements of various kinds is continuing in fringe positions of the PRTR (Table 2.1). These elements are being irradiated to evaluate their performance at higher burnups. The fringe elements contain both pneumatically impacted and mechanically mixed (incrementally loaded) UO_2 - PuO_2 type fuel. The highest peak burnup of over 16,000 MWd/tonne is being obtained on one of the first mechanically mixed UO_2 - PuO_2 elements charged into the reactor. Irradiation of these elements will continue in the PRTR.

The entire PRTR fuel loading of 81 elements was examined in the basin during the current extended outage. The fuel has attained about one-third of the goal exposure to be attained during the Batch Core Experiment. In general, the condition of the elements is excellent. Minor damage to the elements caused by normal charge-discharge operations was repaired. Seventeen loose or damaged circumferential strip bands were replaced. Six additional bands were removed from various elements for experimental purposes. Thirteen damaged hanger pins were also replaced. No rod exchanges were made on the HPD elements.

Crud deposition caused by the neutral pH coolant was noted on the fuel element surfaces. The nonuniform crud deposit is thinner on elements from the central regions of the core

TABLE 2.I. PRTR Core Loading Status (1/28/68)

Element Type ^(a)	Number of Elements	Peak Rod Power Generation, kW/ft	Peak Burnup, MWd/tonne
<u>Batch Core Experiment:</u>			
Vipac PI ^(b) UO ₂ -2 wt% PuO ₂ (Batch Core Experiment Elements)	64	20	6,400
Hot-Pressed Pellet UO ₂ -1.94 wt% PuO ₂	2	21.5	5,700
8-Rod FERTF Test Element	1	18	560
<u>Fringe Position Tests:</u>			
Vipac PI UO ₂ -2 wt% PuO ₂ (Pilot HPD Elements)	5	20	9,800
Swaged PI UO ₂ -2 wt% PuO ₂	1	20	6,100
Swaged PI UO ₂ -1 wt% PuO ₂	1	17	12,700
Vipac PI UO ₂ -1 wt% PuO ₂	2	15	9,400
Vipac MM ^(c) UO ₂ -0.5 wt% PuO ₂	1	16	16,300
Swaged MM UO ₂ -0.5 wt% PuO ₂	2	13	15,600
Swaged UO ₂	1	14	14,000
Vipac PI ThO ₂ -5 wt% PuO ₂	1	8	260

(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.

than on elements being irradiated in the peripheral regions. The amount of crud currently is judged not to be thick enough to cause any problems; however, the rate of further crud buildup will be closely followed.

Postirradiation Examination Results:

Among the first irradiations performed in the reactor during the Batch Core Experiment were tests to determine the maximum allowable reactor power level based on a maximum

fuel temperature limit in HPD fuels. The limiting PRTR power level is established by the linear rod power that produces fuel temperatures just below melting. Twenty fuel rods were metallographically examined to ascertain fuel structures formed during irradiation and fuel performance. The HPD fuel rods examined included vibrationally compacted UO₂-PuO₂ rods, rods containing UO₂-PuO₂ pellets fabricated by hot pressing, and rods containing UO₂-PuO₂ pellet fuel fabricated by cold pressing and sintering. The PuO₂

concentration in the three types of HPD mixed-oxide fuels evaluated is nominally 2 wt% PuO_2 . Although most of the results of these examinations are preliminary and the burnups are relatively low, some interesting comparative observations on fuel performance have been made, some of which were reported previously.⁽¹⁾

Postirradiation examination indicates that the onset of fuel melting occurred in one of the UO_2 - PuO_2 hot-pressed pellet fuel rods after irradiation at a peak rod power of about 21.2 kW/ft (Figure 2.1a). Structures typical of pellet fuel were formed during irradiation. The structure of this specimen irradiated to a peak burnup of 1270 MWd/tonne is characterized by an unaffected region adjacent to the cladding, an equiaxed grain growth region, a columnar grain growth region, and a centrally located region composed of high density pore-free grains. Ceramographic examination and the centrally located high fission product concentration regions on the beta-gamma autoradiograph that indicate once-molten fuel show that once-molten fuel was extruded into radial cracks that apparently existed during irradiation. Examination of a companion rod (3700 MWd/tonne peak) containing UO_2 - PuO_2 pellets fabricated by hot pressing (Figure 2.1b) shows that all evidence of the once-molten fuel structure is eradicated in less than 2430 MWd/tonne of subsequent irradiation under nonmolten conditions. Evidence of once-molten fuel has also been essentially eradicated from the beta-gamma autoradiograph. This sam-

ple was irradiated under the same maximum power conditions as the specimen shown in Figure 2.1a. A sample from another rod irradiated to a peak burnup of 3950 MWd/tonne under the same postmolten conditions as the sample shown in Figure 2.1b indicates that although the fuel structure is similar, i.e., the type and extent of columnar grain growth, a central void has formed (Figure 2.1c). A central void formed in the sample shown in Figure 2.1c because fuel melting did not occur at any time during its irradiation, and no axial fuel relocation resulted.

A commercially fabricated PRTR fuel rod containing cold-pressed and sintered UO_2 - PuO_2 pellets was recently examined (Figure 2.2a). The irradiation performance of this rod is being compared with the performance of a companion rod containing UO_2 - PuO_2 pellets fabricated by hot pressing (Figure 2.2b). Both rods were presumably irradiated at the same peak power condition of 17.8 kW/ft to a peak burnup of about 1400 MWd/tonne. Coolant temperature was 40 to 65 °C. Recrystallization and equiaxed grain growth occurred in the central region of the cold-pressed and sintered pellet fuel which are characteristic of maximum fuel temperatures in the range of 1500 to 1600 °C (Figures 2.2a and 2.3). Maximum fuel temperatures in the range of 1900 to 2000 °C are expected under these operating conditions. Possible reasons for this apparent anomaly are being investigated. Because of the indicated lower effective thermal conductivity

FIGURE 2.1. Transverse Sections of UO_2 -1.94 wt% PuO_2 Hot-Pressed Pellet Fuel Rods Irradiated in PRTR. Rod (a) was irradiated at a peak power of 21.1 kW/ft to 1270 MWd/tonne. Fuel melting is evident. Rod (b) was irradiated at the same peak power as (a) but for an additional 2430 MWd/tonne under nonmolten conditions. Rod (c) was irradiated to 3950 MWd/tonne under the same post-melting conditions as rod (b).

Photomicrographs

β - γ Autoradiographs α Autoradiographs

(a)

(b)

FIGURE 2.2. Transverse Sections of (a) UO_2 -1.94 wt% PuO_2 Cold-Pressed and Sintered Pellet and (b) UO_2 -1.94 wt% PuO_2 Hot-Pressed Pellet Fuel Rods Irradiated in PRTR

of the hot-pressed pellet type fuel, maximum estimated fuel temperatures in this rod were above 2000 °C as expected (Figure 2.2b). A larger amount of gas was collected from the hot-pressed pellet fuel rod than from the cold-pressed and sintered pellet fuel rod (198 ml versus 29 ml) as a result of the higher fuel temperatures.

Discrete PuO_2 particles, similar to those present in irradiated vibrationally compacted, pneumatically impacted UO_2 - PuO_2 fuel, are present in the cooler peripheral regions of irradiated UO_2 - PuO_2 hot-pressed pellet fuel rods (Figure 2.4). Mechanically mixed UO_2 - PuO_2 feed material is used in both processes and no preirradiation homogenization or

solid-solution formation occurs.

During the irradiation of both types of fuel, however, homogenization of the UO_2 and PuO_2 takes place in the higher temperature regions, i.e., at temperatures sufficient to cause recrystallization and equiaxed grain growth. Discrete PuO_2 particles are not present in the mechanically mixed cold-pressed and sintered pellet fuel because homogenization occurs during the sintering step (Figure 2.3). Unexplained localized regions, which are sintered to a greater degree than the surrounding fuel material, are present in the cooler peripheral regions of irradiated cold-pressed and sintered UO_2 - PuO_2 pellet fuel (Figure 2.5). The small areas

FIGURE 2.3. Photomicrographs from a Transverse Section of an Irradiated UO_2 -1.94 wt% PuO_2 Cold-Pressed and Sintered Pellet Fuel Rod Irradiated in PRTF (Rod 101). The O/M ratio for this irradiated specimen, as determined by the oxidation method, is 2.008.

E-4549

E-4550

200X

FIGURE 2.4. Photomicrographs Showing Discrete PuO_2 Particles in the Cooler Regions of Irradiated UO_2 -1.94 wt% PuO_2 Pressed Pellet PRTR Fuel.

(approximately 0.6 mm diameter) apparently operated at higher localized temperatures although there is no evidence of localized PuO_2 agglomeration on the autoradiographs.

A large amount of second phase U_4O_9 is present in the microstructure of irradiated UO_2 - PuO_2 hot-pressed pellet fuel (Figure 2.6). The high O/M of approximately 2.15 accounts for the presence of the second phase. The shape of the U_4O_9 precipitate and the mode of its formation depends upon the oxygen concentration and the thermal treatment.⁽²⁾ The second phase exists as a continuous interconnected network of U_4O_9 plus considerable amounts of the precipitate in the form of a Widmanstätten structure within the large columnar grains near the thermal center of the fuel. This suggests that the oxygen

concentration may be greater near the thermal center of the fuel than in the cooler peripheral regions. Larger amounts of the precipitated second phase are also present in those rods that were slowly cooled during their last shutdown in the reactor as expected.

Gas was collected from three irradiated PRTR fuel rods containing hot-pressed UO_2 - PuO_2 pellets and from one rod containing cold-pressed and sintered UO_2 - PuO_2 pellets (Table 2.II). The larger amount of total gas collected from the hot-pressed pellet fuel rods, partially as a result of higher fuel temperatures, poses no immediate internal pressure problem. Higher fuel temperatures in the hot-pressed pellet rods also resulted in higher fission gas release values.

E-4613

FIGURE 2.5. Localized High Temperature Area in the Peripheral Region of an Irradiated UO_2 -1.94 wt% PuO_2 Cold-Pressed and Sintered Pellet PRT Fuel Rod.

FIGURE 2.6. Second Phase U_4O_9 is Present in the Equiaxed Grain Growth Region (Left) and Columnar Grain Growth Region (Right) of Irradiated Hot-Pressed UO_2 -1.94 wt% PuO_2 Pellet Fuel. The O/M ratio for this irradiated specimen, as determined by the oxidation method, is 2.145.

Mass spectrographic analyses indicate that the gas released from the hot-pressed pellet fuel contains a high percentage of CO_2 (Table 2.III). The CO_2 is probably formed by the reaction between carbon and UO_2 at temperatures above 1200°C . Carbon is introduced as an impurity during the hot pressing operation which is performed in a graphite die.

References

1. Staff of Battelle-Northwest. *Plutonium Utilization Program Technical Activities Quarterly Report-October, November, December 1967*, BNWL-654, Pacific Northwest Laboratory, Richland, Washington. February 1968.
2. B. E. Schaner. "Metallographic Determination of the $\text{UO}_2\text{-U}_4\text{O}_9$ Phase Diagram," *Journal of Nuclear Materials*, vol. 2, no. 2, pp. 110-120. 1960.

TABLE 2.II. Gas Release from $\text{UO}_2\text{-PuO}_2$ Pellet Fuel Rods Irradiated in PRTR

Rod Number	Fabrication Method	Peak Power, kW/ft	Rod Burnup, MWd/tonne		Total Gas Collected, ml	Calculated Xe+Kr Release, %
			Average	Peak		
A-13	Hot Press	21	2860	3700	217	46.5
B-13	Hot Press	18	2080	2680	666	45
B-15	Hot Press	17.8	990	1280	198	47
101	Cold Press and Sintered	17.8	990	1280	29	1.4

TABLE 2.III. Analysis of Gas Released from Irradiated PRTR $\text{UO}_2\text{-PuO}_2$ Pellet Fuel Rods

	Rod Number and Mole Percent			
	A-13	B-13	B-15	101
Xe-136	14.2	3.31	5.55	0.92
134	9.49	2.16	3.65	0.58
133	--	--	--	--
132	6.32	1.41	2.19	0.82
131	4.24	0.94	1.43	0.37
130	--	--	0.19	0.01
Kr-86	1.19	0.27	0.49	0.07
85	0.19	0.04	0.08	<0.01
84	0.69	0.16	0.32	0.04
83	0.41	0.10	0.16	<0.01
CO_2	38.0	81.5	42.8	0.40
Ar	<0.01	<0.01	<0.01	0.01
O_2	<0.01	0.02	0.27	0.06
N_2	9.21	4.73	22.2	0.35
CO	2.85	0.68	<0.1	<0.1
He	13.0	4.52	16.9	96.0
H_2	0.21	0.13	3.68	0.42
CH_4	<0.01	<0.01	<0.01	<0.01

HIGH EXPOSURE PLUTONIUM STUDIES

R. C. Smith, L. G. Faust,
D. D. Lanning

A series of measurements were made on high exposure plutonium in the form of Al-Pu alloy fuel plates to determine dose rate variations due to geometry effects. The metal alloy contained 20 wt% plutonium and was clad in 0.020 in. of aluminum. The isotopic composition of the plutonium was approximately 0.5% ^{238}Pu , 69% ^{239}Pu , 21% ^{240}Pu , 6.6% ^{241}Pu , and 2.9% ^{242}Pu . Average plate dimensions were about 3 1/2 in. wide by 12 in. long.

The dose rates from the high exposure plutonium fuel plates as measured by a portable "Cutie Pie" (CP) dosimetry instrument are summarized in Table 2.IV. Variations in dose rate from the plates are shown as a function of geometrical array and distance from the plate surface.

There was no measurable increase in dose rate when more than seven plates were added to the stack (as measured from the top). It should be pointed out that the reading obtained with a portable CP instrument held against the surface of any source is not the true surface dose rate from the source. The actual dose rates from these fuel plates ranged from 2.24 to 2.37 rad/hr at the surface as measured with a special research laboratory system. The rather large difference (2.3 rad/hr versus 0.8 rad/hr maximum) between the actual surface dose rate and the CP reading is due mainly to three factors:

(1) the sensing unit on a CP is about 2 in. away from the instrument window, (2) a major portion of the dose rate from high exposure plutonium is from low energy photons which are appreciably attenuated by the window and even 2 in. distance and, (3) the

TABLE 2.IV. Measured Dose Rate Versus Source Geometry and Distance for High Exposure Fuel Plates

Source Geometric Array	Dose Rate, mR/hr ^(a)			
	Surface	1 ft	2 ft	3 ft
Single plate (3 1/2 by 12 in.)	360 to 400	28	10	5
3 1/2 by 36 in.	--	--	--	11
10 by 12 in.	--	100	30	13
12 by 31 in.	--	120	45	28
15 by 21 in.	800	140	80	60
3 plates stacked on top of one another	480			
7 plates stacked on top of one another	600			

(a) The measurements are taken from the center of the array or directly above the center.

efficiency of the CP is necessarily lower than that of the elaborate laboratory system used for surface dose rate determinations. The dose rate at 1 ft as measured by the research system averaged about 100 mR/hr.

Detailed records of personnel exposure are being maintained during fabrication of Phoenix Al-Pu alloy fuel plates. The accumulated hand exposure is followed by weekly readings of two different types of finger rings and total body exposure is measured on a monthly basis by film badges. (Shorter time periods were used early in the process.) In addition, self-reading pencils are worn and checked daily as an early warning device.

The high surface dose rate from the Al-Pu alloy dictates that direct glove to metal contact be held to a minimum, and several modifications to normal handling procedures (such as the use of tongs, local shielding, and lead aprons) are necessary. Hand exposure as high as 350 mrem/hr and total body exposure rates in excess of 20 mrem/hr were recorded at the start of the fabrication program. Some work stations and process units still require fuel handling modifications; however, latest measurements indicate that the rate of personnel exposure for the overall process has been reduced by a factor of 10.

FISSION PRODUCT DISTRIBUTION IN IRRADIATED UO_2 -PuO₂ FUEL RODS

J. A. Basmajian and M. D. Freshley

A knowledge of the radial distribution of fission products and fis-

side elements is important for the complete analysis of the irradiation behavior of mixed-oxide fuels. The techniques presently being used at PNL are alpha and beta-gamma autoradiography and radiochemical analysis of microdrilled samples. There are inherent problems associated with these techniques. Autoradiography is not quantitative and does not distinguish particular species. Analysis of microdrilled samples does not provide a continuous distribution and is subject to analytical errors. The use of precision continuous gamma scanning to accurately determine the radial distribution of fission products and fissile elements in irradiated UO_2 -PuO₂ fuel rods is being investigated. Capabilities of the Los Alamos Scientific Laboratory (LASL) are being used to determine the diametral distribution of various fission products and fissile elements by precision gamma scanning. Precision gamma scanning techniques have been developed at LASL utilizing high resolution Ge (Li) anticoincidence gamma spectroscopy.

The first studies will be conducted on a transverse section from a vibrationally compacted UO_2 -2 wt% PuO₂ fuel rod that was irradiated in PRTR to about 430 MWd/tonne. Fission product and fissile element distributions in this sample have been extensively investigated by autoradiographic and microdrilling techniques and the gamma scanning results will be compared with these data. A polished surface immediately adjacent to the surface that was intensively examined by autoradiographic and

microdrilling techniques was prepared. The sample was enclosed in a special aluminum container and shipped to LASL for gamma scanning studies.

The distribution of several elements will be studied by scanning across the diameter of the specimen along two axes 90° apart. The isotopes that will be studied are ^{238}Pu , ^{239}Pu , ^{235}U , ^{95}Zr , ^{103}Ru , ^{105}Pu , ^{137}Cs , ^{140}Ba , ^{144}Ca and ^{242}Cm . Gamma ray spectra analyses will be made prior to the scans.

TRANSIENT TESTING

R. L. Gulley and D. E. Blahnik

Eight pellet-containing fuel rods were fabricated and shipped to Argonne National Laboratory (ANL) for transient irradiation in the Transient Reactor Test Facility (TREAT). These rods will be tested as part of a joint PNL-ANL program* to investigate the transient behavior of thermal reactor fuel rods. Each Zircaloy-clad rod is 1.44 cm (0.565 in.) in diameter and contains 13.93 cm (5.48 in.) of UO_2 (4.8% ^{235}U) pellets with a 1.82 cm (0.72 in.) long fission gas plenum. A depleted UO_2 pellet is located at each end of the fuel column.

Seven vibrationally compacted, pneumatically impacted, UO_2 rods are

* Contributors to the cooperative program include staff members of the Chemical Engineering Division of ANL, the TREAT Facility, and PNL.

being fabricated with fuel from the same batch of material used to fabricate the pellet-containing rods. Except for the fuel form, these rods will be similar in design to the pelleted rods and will be irradiated to compare the transient behavior and performance limitations of pelleted and vibrationally compacted UO_2 fuel rods.

THORIA-PLUTONIA IRRADIATIONS

T. B. Burley and M. D. Freshley

A Zircaloy-clad 19-rod cluster containing vibrationally compacted, pneumatically impacted, ThO_2 -5 wt% PuO_2 element attained a peak burnup of 450 MWd/tonne at a peak rod power generation of 7.4 kW/ft. Maximum fuel temperatures were approximately 1200 °C. The element that contains two fuel rods instrumented to measure internal gas pressure and plenum gas temperature was discharged after approximately six weeks of operation. The maximum fuel rod pressure measured during this irradiation was approximately 3 atm. This pressure is higher (approximately 1.5 times) than the pressure measured in vibrationally compacted UO_2 - PuO_2 fuel rods operating under similar conditions. The higher pressure in the ThO_2 - PuO_2 fuel rods is believed to be a consequence of the higher sorbed gas ($0.09 \text{ cm}^3/\text{g}$) and moisture (28 ppm) content in the ThO_2 - PuO_2 fuel material.

FISSION PRODUCT MIGRATION IN MIXED-OXIDE FUEL

R. Shimanuki* and M. D. Freshley

Fission product migration in vibrationally compacted $\text{UO}_2\text{-PuO}_2$ PRTR fuel rods is being studied. A transverse section from a $\text{UO}_2\text{-2 wt\% PuO}_2$ fuel rod irradiated at a peak power of about 20 kW/ft with the onset of fuel melting is being extensively analyzed by the microdrilling technique. This rod was irradiated in the 12-rod ring of a 19-rod HPD element to a peak burnup of 0.9×10^{19} fission/cm³.

Analytically determined fission product distributions are compared with calculated diametral distributions. Self-shielding effects within the 19-rod cluster fuel elements result in nonsymmetric fission product formation within rods from the outer 12-rod ring along an axis away from the center of the element. Preliminary analytical results for ¹⁰³Ru are compared with the calculated distribution across the diameter of the sample in Figure 2.7. The results show that ruthenium migrates extensively during irradiation and that the high and low concentrations correspond with the dark and light regions on the beta-gamma autoradiograph, respectively.

Similar studies of other fission products and uranium and plutonium isotopes are in progress. A sample has been sent to the Los Alamos Scientific Laboratory for continuous

gamma scanning studies to compare results from the two experimental techniques.

IN-REACTOR SINTERED FUEL

G. R. Horn, R. L. Gulley, and R. J. Anicetti

An experimental fuel element containing low density (~71% TD) mixed-oxide pellets was irradiated to investigate the in-reactor performance of cold-pressed, nonsintered pellet fuel. During irradiation the nonsintered pellets recrystallized into a coherent, relatively high density fuel column having a large central void. In-reactor modification of the fuel improved the thermal characteristics so that no central melting occurred when the rods were operated at greater than 16 kW/ft.

The 7-rod cluster element was irradiated in the Engineering Test Reactor P-7 loop at a peak linear rod power of 16.2 kW/ft to a peak burnup of approximately 1200 MWd/tonne. The Zircaloy-clad rods contained 0.50 in. diameter $\text{UO}_2\text{-0.5 wt\% PuO}_2$ pellets compacted to 71% TD. The fuel column length in each rod was approximately 28 in. In anticipation that gross axial fuel relocation (slumping) might occur during the irradiation of this low density fuel we included three types of fuel restrictors in the element to study their effectiveness in preventing slumping. These included 0.400 in. long depleted UO_2 sintered pellets, 0.038 in. thick tungsten wafers, and 0.058 in. thick tungsten wafers.

* Visiting scientist from Japan Atomic Energy Research Institute.

FIGURE 2.7. Ruthenium-103 Distribution in a Vibrationally Compacted UO_2 -2 wt% PuO_2 Fuel Rod (FR-47) Irradiated in PRTR at a Peak Power of about 20 kW/ft. Peak burnup is 0.9×10^{19} fissions/cm³. The direction across the specimen is radial with respect to the center of the element.

Autoradiography and gamma scanning of the irradiated rods showed no evidence of axial fuel movement except in the rod that was subjected to the maximum power conditions. Although some axial fuel movement apparently occurred in this rod, gross relocation was effectively controlled by the depleted UO_2 pellet restrictors.

Destructive postirradiation examination of fuel rods from the cluster showed that extensive fuel sintering occurred during irradiation. In a rod that contained no axial fuel movement restrictors (the center rod of the cluster), a large central void and radially oriented high density columnar grains formed along essentially the entire length of the fuel column, but there was no evidence of fuel slumping or melting (Figures 2.8 and 2.9). In a rod containing 0.038 in. thick tungsten axial fuel movement restrictors, central voids formed in the fuel column between restrictors (Figure 2.10). Again, there was no evidence of fuel slumping or melting. Analyses of the gases collected from these rods indicated that approximately 60% of the fission gases formed were released from the fuel during irradiation.

The initial thermal conductivity of as-fabricated, nonsintered pellets is considerably lower than that of presintered pellets. However, as this experiment demonstrated, fuel temperatures greater than melting were not attained because in-reactor fuel sintering and attendant densification improved the thermal conduc-

tivity and decreased the length of the heat path. Calculations indicate that heat ratings as high as 25 kW/ft can be achieved in reactor-sintered pellet fuels of this type without fuel melting.

Preliminary fabrication studies have been aimed at establishing a source of relatively inexpensive UO_2 powder that can be compacted to 70% TD. Heat-treated ceramic grade UO_2 attained 60% TD after dry compaction at 40 tsi. Three batches of material were heat treated in hydrogen for 4 hr at 1000 °C, 1100 °C and 1200 °C. Sterotex was used as a die wall lubricant and no damage occurred to the hardened steel die. Two new die wall lubricants, butyl stearate and butyl oleate, may permit higher compaction pressures. Other forms of UO_2 to be investigated are as-calcined and hammermilled UO_3 (calcined UNH) and metal production grade UO_2 .

Preliminary fabrication cost studies indicate that considerable savings in fabricating plutonium-bearing oxide fuels may be realized by eliminating the sintering step. Costs associated with powder preparation, sintering, grinding, and scrap recovery can be significantly reduced or eliminated.

These encouraging preliminary results have prompted further experiments with this fuel concept. Further fabrication studies are planned to optimize fabrication parameters, and additional irradiation tests have been initiated to determine in-reactor performance at higher heat ratings and exposures.

E-4424

Photomicrograph

E-1305

 β - γ Autoradiograph

E-1304

 α Autoradiograph

FIGURE 2.8. In-Reactor Sintered $\text{UO}_2\text{-PuO}_2$ Fuel. Transverse Section Approximately 9 in. from the Bottom of the Rod having no Restrictors.

17239

Photomicrograph

1318

β - γ Autoradiograph

1319

α Autoradiograph

FIGURE 2.9. In-Reactor Sintered $\text{UO}_2\text{-PuO}_2$ Fuel. Longitudinal Section from the Peak Heat Flux Region of the Rod having no Restrictors.

17242 ↗
0.038 in. Thick
Tungsten Restrictor

Photomicrograph

1337

β - γ Autoradiograph

1338

α Autoradiograph

FIGURE 2.10. In-Reactor Sintered UO_2 - PuO_2 Fuel. Longitudinal Section from the Peak Heat Flux Region of the Rod Containing Thin Tungsten Restrictors.

UO₂-PuO₂ HIGH BURNUP IRRADIATIONS

W. J. Bailey

Irradiation testing of vibrationally compacted UO₂-1.5 wt% PuO₂ capsules was completed earlier and data from the highest burnup capsules and rods are being analyzed. Burnups to approximately 30,000

MWd/tonne (7.4×10^{20} fissions/cm³) were attained with the capsule specimens. Ratios of measured isotope concentrations of the fuel from the UO₂-PuO₂ capsules irradiated in the MTR are summarized in Table 2.V. A transverse section from one of the specimens is shown in Figure 2.11.

Neg 0670414

FIGURE 2.11. Photomosaic (Originally 75X) of Transverse Section at Midplane of Irradiated, Zircaloy-Clad (1.07 cm OD), UO₂-1.5 wt% PuO₂ Fuel Capsule. The vibrationally compacted fuel rod contains pneumatically impacted mixed-oxide which achieved a burnup of 16,930 MWd/tonne and operated with an initial maximum linear rod power of 630 W/cm (19.2 kW/ft).

TABLE 2.V. Measured Isotopic Ratios from
 UO_2 - PuO_2 Capsules Irradiated in MTR

Estimated Burnup		Isotope Ratios			
MWd/tonne	10^{20} fiss/cm ³	$^{239}Pu/^{242}Pu$	$^{243}Am/^{241}Am$	$^{244}Cm/^{243}Am$	$^{244}Cm/^{242}Cm$
12,560	3.1	7.85	0.33	0.088	5.03
16,930	4.2	4.12	0.54	0.116	9.89
18,300	4.6	3.62	1.15	0.116	2.66
~26,400	~6.5	2.16	2.79	0.153	4.84
~30,000	~7.4	1.79	4.33	0.220	8.06

The specimen burnup is estimated from isotopic ratios. The measured and calculated $^{239}Pu/^{242}Pu$ ratios are in reasonably good agreement, but the measured $^{243}Am/^{241}Am$ ratios are much lower than the calculated values. Possible sources of error are being studied. Calculated values for the curium isotopes are not yet available for comparison purposes.

GAS PRESSURE MEASURING EXPERIMENTS

T. B. Burley

Irradiation of two vibrationally compacted UO_2 -2 wt% PuO_2 High Power Density fuel elements, each containing two instrumented fuel rods, continues in PRTR. The two sets of instrumented fuel rods have operated to peak burnups of 6000 and 3600 MWd/tonne. The internal pressure in both sets of fuel rods continues to increase with burnup at approximately constant rates (Figures 2.12 and 2.13). The fission gas release rates are directly proportional to volumetric averaged fuel temperatures as indicated by the calcu-

lated pressure buildup curves in Figures 2.12 and 2.13. Internal gas pressure increases in a step manner during reactor power cycles or shutdown-startup sequences. Close observation of the gas pressure in the high-burnup fuel rods before and after numerous reactor shutdown-startup cycles indicates that most of the fission gas is released during shutdown.

The power generation in the low burnup fuel rods was recently increased from a nominal peak of approximately 9.5 kW/ft to 11.4 kW/ft. This power increase raised the volumetric averaged fuel temperature from approximately 700 °C to 900 °C. As a result, the measured internal gas pressure increased noticeably during this operation.

In the high burnup fuel rods, the power generation was decreased from a nominal peak of 16.5 kW/ft to approximately 14 kW/ft. This decrease in power lowered the volumetric averaged fuel temperature from approximately 1250 °C to 1100 °C. The measured internal pressure did not decrease during the lower power

FIGURE 2.12. Internal Pressure Buildup in High Burnup PRTR Fuel Rods

FIGURE 2.13. Internal Pressure Buildup in Low Burnup PRTR Fuel Rods

operation, but in fact increased at approximately the same rate as prior to the power decrease. This is explained by the fact that changes in internal gas pressure are relatively insensitive to small temperature changes at high temperature.

FERTF TESTING

M. D. Freshley and D. D. Lanning

The effects of specific power and burnup on the defect behavior of pellet and vibrationally compacted fuel will be studied by irradiating rods in the PRTR Fuel Element Rupture Test Facility (FERTF).⁽¹⁾ The 8-rod FERTF Test Element will be utilized for these experiments.

Two low power irradiations were performed with the FERTF Test Element in PRTR to determine the relative power generation between rods operating in the test element and rods operating in the surrounding elements in Ring 1 of the batch core. These experiments confirmed that the present 8-rod FERTF Test Element design will provide the power generation flexibility needed to optimize the power match between the FERTF and Ring 1 within the desired range of operating conditions.

The first irradiation of the 8-rod FERTF Test Element, which was the first of the power calibration experiments, was successfully completed. This test was conducted at a peak rod power generation of 18 kW/ft with maximum fuel temperatures of about 2600 °C. Three of the vibrationally compacted rods were removed from the assembly without difficulty for postirradiation examin-

ation and replaced with unirradiated rods. The next test will be performed at a peak power generation of about 18.7 kW/ft.

Two carefully selected and characterized circumferential strip bands irradiated on the FERTF Test Element were gamma scanned in an effort to determine the relative power generation among the 8 rods. Prior to irradiation, it was determined that the mass variation along the length of each band was approximately $\pm 1\%$ from a mean value. Counting was done with a NaI detector (>0.5 MeV) at 0.2 cm intervals along the bands. Local variations in the gamma activity of the bands relative to rod position were detected as expected. Variations in peak heights, however, indicate the possibility of a flux tilt across the assembly and therefore, a power generation variation among the 8 rods. However, the pattern of the suggested variation does not conclusively indicate a flux tilt. Analysis of these data is continuing.

Reference

1. M. D. Freshley. "Proposal to Perform Irradiation Experiments in the FERTF at Maximum Power Generation of About 20 kW/ft," BNWL-CC-1265, December 1, 1967.

CVTR DRIVER FUEL FOR PRTR

R. L. Gulley and D. E. Blahnik

Excess Carolinas-Virginia Tube Reactor (CVTR) fuel assemblies are being considered as possible driver fuel for continuing irradiation of 12 to 19 selected UO_2 - PuO_2 High Power

Density (HPD) fuel assemblies in PRTR.⁽¹⁾ The HPD fuel assemblies will be irradiated to peak burnups in the range of 40,000 to 45,000 MWd/tonne.

Utilization of the CVTR fuel rods in PRTR requires design modifications to provide an assembly that is compatible with the HPD fuel assembly design. The CVTR fuel contains cold-pressed and sintered enriched UO_2 (~2% ^{235}U) pellets clad in Zircaloy.

One CVTR assembly was modified for irradiation testing in the PRTR to evaluate the performance of the modified fuel element assemblies as driver fuel. This assembly was modified by (1) removing the rods from the CVTR assembly, (2) reducing the overall length of each rod by shortening the plenum and fuel column, (3) welding PRTR type end plugs in each rod, (4) attaching wire spacers on 12 of the rods, and (5) assembling the rods into a 19-rod element similar to a PRTR fuel assembly.

Reference

1. M. D. Freshley and S. Goldsmith. High Power Density Fuel Studies in PRTR, Pacific Northwest Laboratory, Richland, Washington. BNWL-SA-1623, January 29, 1968.

CERAMIC FUEL DISSOLUTION STUDIES

L. D. Perrigo

Dissolution processes are being developed to chemically decontaminate reactors that use ceramic fuel materials. Work during the past quarter has been centered on determining the effects of irradiation history on ceramic dissolution.

The first of a series of laboratory dissolution experiments with PRTR-exposed ceramic fuels was completed in late February. In these tests, hot-pressed pellet and vipac, incrementally loaded, UO_2/PuO_2 fuel materials were treated with oxalic-peroxide-gluconic (OPG) decontamination solutions. Preliminary data show about 80% dissolution in five hours; analytical measurements have not yet been made to determine if there was a preferential dissolution of the UO_2 .

Additional experiments are planned to further define the behavior of these irradiated materials. The amount of UO_2/PuO_2 solid solution present; the number, size, and location of discrete PuO_2 particles in UO_2 matrices; variations in fuel operating temperature radially and in length; and mode of manufacture and attendant properties such as crack frequency and depths are expected to be important variables in the dissolution process. Their effects will be observed during future experimentation.

3.0 REACTOR PHYSICS

APPROACH-TO-CRITICAL EXPERIMENTS WITH

UO₂-4 wt% PuO₂-H₂O LATTICES

W. P. Stinson, J. H. Lauby

A series of subcritical experiments have been initiated to determine critical masses and bucklings of UO₂-4 wt% PuO₂-H₂O lattices.⁽¹⁾ These experiments are part of a series of experiments that utilize fuels having different plutonium enrichments and various concentrations of the isotope ²⁴⁰Pu. These data provide a rigorous test of the analytical methods used in calculating parameters of light water moderated systems which contain plutonium.

The experiments are being conducted in the Critical Approach Facility (CAF),⁽²⁾ using hexagonal lattices which are reflected with an effectively infinite thickness of H₂O in all directions. The lattice plates are constructed from 0.75 in. thick lucite. A pair of identical lattice plates, spaced vertically one fourth the length of a fuel rod from each end of the core, is used to support the fuel rods in a vertical position. The weight of the fuel rods is supported by a 0.75 in. thick lucite plate. A Be-Pu neutron source containing 240 g of plutonium is positioned on the vertical axis of the core, 2 1/2 in. beneath the lower end of the core. Three BF₃ proportional counters which are placed in the reflector and spaced at approximately 120° intervals around the core are used to collect the experimental data. A BF₃ proportional counter placed in the

geometrical center of the core (replacing a fuel rod) is used in a few of the experiments for correlation purposes.

The UO₂-4 wt% PuO₂ fuel is vibrationally compacted into tubes of Zircaloy-2 whose wall thicknesses are 0.0335 in. and whose inside diameters are 0.4975 in. Each tube contains 1084.5 g of UO₂-PuO₂ in a height of 36.0 in. The resulting average fuel density is 9.46 g/cm³. The isotopic composition of the fuel is natural uranium and plutonium with 0.28/75.38/18.10/5.08/1.15 at% of ²³⁸Pu/²³⁹Pu/²⁴⁰Pu/²⁴¹Pu/²⁴²Pu, respectively.

The experiments were conducted using critical approach and exponential measurement techniques. The relative multiplication of the loadings were measured as the fuel is added up to 96% of the critical mass with an infinite reflector on the top and also with the moderator level 7 in. below the top of the fuel. Data for a clean core is obtained by removing the control and safety rod thimbles, replacing the center counter by a fuel rod, and unloading the fuel in appropriate increments. The critical number of fuel rods is obtained by fitting a straight line, by the method of least squares, to $I_0(\text{AR})/\phi$ plotted against the number of fuel rods, where

I_0 = A modified Bessel Function
of the first kind

A = Normalizing factor

$$R = \text{Core radius} = \ell \sqrt{\frac{N\sqrt{3}}{2\pi}}$$

ϕ = Neutron flux as detected by the 3 BF_3 proportional counters and averaged

ℓ = Lattice spacing

N = Number of fuel rods in the core.

Data for the exponential experiments were taken by removing the central fuel rod and obtaining a neutron flux traverse with a BF_3 proportional counter along the vertical axis. Similar traverses with indium foils were taken on some of the cores for comparison purposes.

The material bucklings are computed from the relaxation lengths obtained from the exponential data and are equated to the geometrical bucklings for the critical loadings using the equation

$$B_m^2 = B_g^2 = \left(\frac{J_0}{R_e + \lambda} \right)^2 - \gamma^2$$

$$= \left(\frac{J_0}{R_c + \lambda} \right)^2 + \left(\frac{\pi}{H + 2\lambda} \right)^2,$$

where

B_m^2 = Material buckling

B_g^2 = Geometrical buckling

γ = Relaxation length of neutron flux on the vertical axis

H = Length of fissionable material in the fuel rods

R_e = Radius of the bare core during exponential measurement

R_c = Radius of the bare core when critical

Values of reflector savings and extrapolation length (λ) are calculated by an iteration process using the preceding equation. It is assumed that the value of λ is the same in the axial and radial directions.

The results obtained are summarized in Table 3.1. The error on the critical number of rods results from the goodness of fit of the approach data fitted to a straight line by the method of least squares. The error on λ is caused by the errors on the critical number of rods and on the relaxation length. The latter error is calculated by a generalized least squares computer program (LEARN) which fits the data from the exponential experiments to a hyperbolic sine function.

The agreement between the pre-experiment calculations and the measurements is fairly good, and, with the exception of the 0.80 in. lattice, the calculations were conservative. However, the approach in the 0.80 in. lattice was limited to approximately 81% of the extrapolated critical mass by the number of rods available, and previous approach-to-critical experiments have shown that extrapolations made for cores which contain less than 96% of the critical mass tend to underestimate the number of rods required for critical.

TABLE 3.I. Measurements With UO_2 -4 wt% PuO_2 FuelApproach-to-Critical

Hexagonal Lattice Pitch, in.	0.80	1.3856	1.60
H ₂ O/Fuel Volume Ratio	1.564	7.271	10.874
Hydrogen/Plutonium Atom Ratio	123.5	578.1	804.8
Number of Rods for a Critical Core:			
Extrapolated from			
measurements	325.8 ± 1.2	123.8 ± 0.4	175.7 ± 0.4
Calculated ⁽¹⁾	330.19	117.72	169.80

Exponential

Number of Rods Used in

Exponential Measurements	169	61	91
Buckling, B^2 (m ⁻²)	84.8 ± 0.5	87.6 ± 0.5	61.6 ± 0.4
Reflector Savings (λ), cm	8.27 ± 0.05	6.57 ± 0.04	5.06 ± 0.08
Relaxation Length (γ) cm ⁻¹	0.0576 ±	0.0660 ±	0.0529 ±
	0.0002	0.0002	0.0004

References

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2. F. Swanberg, Jr., and H. L. Henry. *Critical Approach Facility Final Safeguards Analysis Report*, BNWL-CC-1319, Pacific Northwest Laboratory, Richland, Washington. November 19, 1967.

NONDESTRUCTIVE ANALYSIS OF IRRADIATED FUEL

D. E. Christensen and G. Manca*

Irradiated fuel elements from three different reactors have been examined during the past quarter us-

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ing the PNL Irradiated Fuel Gamma Scan Facility. The fuels studied included: UO_2 -1.5 wt% PuO_2 irradiated in EBWR; UO_2 -0.5 wt% PuO_2 low density test fuel rods, irradiated in ETR; and 19-rod clusters of UO_2 -2 wt% PuO_2 , irradiated in PRTR.

EBWR Fuel

The third set of rods which was removed from EBWR on June 29, 1967, was received from Argonne National Laboratory, and 36 of the 72 rods have been scanned. The 36 rods scanned came from various regions of the core: 18 from the plutonium zone, 14 from the enriched uranium zone, and 4 from the natural uranium zone. The remaining 36 rods are being scanned as an intact element, Pu-59. The rods

will be scanned individually after the study of the inert element is completed.

It was not possible to keep the collimator directly over any given rod in the bundle because of alignment problems. Therefore, a scan along the length of the inert element was not completed successfully. A transverse scan across the bundle was more successful. However, due to interferences between the bundle holder and the underwater tray assembly, a continuous scan across all 6 columns of rods could not be made. The results of a transverse scan, taken at the region of maximum activity, is shown in Figure 3.1. Rod EX-62 contained natural UO_2 , and the rest of the rods contained UO_2 -1.5 wt% PuO_2 . The presence of the natural UO_2 rod is denoted by a decrease in gamma activity as compared with the UO_2 - PuO_2 rods. The correlation between the measured gamma activities and the rod positions in the bundle was referenced to the position of the peak of the activity on Rod EX-62. The spatial resolution was good using an 0.25 in. diameter aperture. Improvements in the spatial resolution could be realized with either a smaller diameter aperture or with a rectangular slit aperture.

It must be noted that these measurements do not represent a measure of the activity from the outer rods alone. Assuming no contribution from rods in vertical columns other than the one being scanned, the measured activities can be represented by the following expression:

$$A_i = \sum_{j=1}^N \alpha_{ij} \gamma^{j-1},$$

where A_i is the activity measured above the i^{th} vertical column of rods in the bundle, α_{ij} is the activity of the j^{th} rod in the i^{th} column, and γ is the geometrical attenuation factor for the gamma activity of a rod in the i^{th} column. It is expected that the γ can be determined for this bundle by appropriate scanning of the rods after removal from the bundle.

Low Density Fuels

Scans were made of 7 rods of UO_2 -0.5 wt% PuO_2 fuel that were fabricated using cold-pressed, unsintered pellets. The fuel had an initial density of approximately 73% T.D. The rods were irradiated in the GEH-P7 Test Loop of the ETR during Cycle 92 (10/12/67 to 11/26/67) to approximately 600 MWd/T. Typical of the scans is that shown in Figure 3.2 of Rod 6-B. Restrictors of 0.038 in. and 0.058 in. thick tungsten and 0.4 in. pellets of high density depleted UO_2 were used in the rods, and one notes the position of the high density depleted UO_2 pellets which were spaced approximately 7 in. apart, used in Rod 6-B. The rods were scanned using a 0.25 in. diameter aperture that was positioned at intervals of 1 in. along the rods.

Two energy intervals were monitored; one covered the range from 0.660 MeV to 0.870 MeV to span the $^{95}\text{Zr}/^{95}\text{Nb}$ gammas, and the other was an integral count above 1.45 MeV to detect $^{140}\text{Ba}/$

FIGURE 3.1. Transverse Gamma Scan of EBWR 36-Rod Element, ^{59}Pu . The scan was done, after 209 days cooling time, with a NaI (TL) 2 in. by 2 in. detector monitoring the $^{95}\text{Zr}/^{95}\text{Nb}$ Gammas, (energy range, 0.662 MeV to 0.865 MeV).

FIGURE 3.2. Gamma Scan of Low Density Fuel Test Rod 6-B. The scan was done, after 53 days cooling time, with a NaI (TL) 2 in. by 2 in. detector monitoring both the $^{95}\text{Zr}/^{95}\text{Nb}$ and $^{140}\text{Ba}/^{140}\text{La}$ gammas.

^{140}La gammas. Included in Figure 3.2 is the ratio of these two activities at each position. These data are presently being analyzed.

PRTR Fuel

Scanning of intact 19-rod clusters of UO_2 -2 wt% PuO_2 which are part of the batch core in the PRTR has been completed. Seven clusters, one from each of the first 7 rings of channels in the reactor, have been examined. Measurements were made near the peak power position axially on 2 rods in each cluster. The rods were on the inside and on the outside edge of the cluster relative to the center of the reactor. The results of these measurements are shown in Figure 3.3.

The $^{95}\text{Zr}/^{95}\text{Nb}$ and $^{140}\text{Ba}/^{140}\text{La}$ gammas were monitored using energy ranges of 0.662 to 0.878 MeV and 1.48 to 1.72 MeV respectively. A 1/16 in. aperture positioned 27 9/16 in. from the top of the fuel was used for the scans. The elements examined and their radial positions in the core are indicated in Figure 3.3. The shape of the inferred radial power profile is in fair agreement with measured tube power data, and the measurements provide quantitative information about the power tilt across an element at various radial positions in the core. The data have been corrected for background only and are not in final form.

FIGURE 3.3. Gamma Scan Data from 19-Rod PRTR Fuel Elements. The data were taken after 25 days cooling time with a NaI (TL) 2 in. by 2 in. detector monitoring both the $^{95}\text{Zr}/^{95}\text{Nb}$ and $^{140}\text{Ba}/^{140}\text{La}$ gammas.

PRTR BATCH CORE EXPERIMENT

J. W. Kutcher and R. I. Smith

The second burnup step of the Batch Core Experiment in the PRTR has been completed, with a total core accumulated exposure of 7300 MWd, which corresponds to an average exposure of 3630 MWd/tonne on the basic 55-element batch loading. The second series of interim critical tests was performed, starting five days after shutdown from a reactor power level of 63 MW. The peripheral elements were discharged and the central channel (1550) was returned to D₂O coolant and charged with a standard batch core element to complete the basic 55-element core configuration. The measurements included the boron concentration for a 100 in. critical level at shutdown temperatures (cold) and xenon-free, boron sensitivity and reactivity worth of selected control rod material samples. The analysis of the measurements has not been completed, and the results reported here are preliminary values, subject to small changes in the final analysis.

A summary of the most pertinent results is presented in Table 3.II.

A linear extrapolation of the cold, xenon-free boron concentration data as a function of average batch core exposure predicts an average exposure of approximately 12,000 MWd/tonne for the 55-element batch loading. This prediction, coupled with the peak-to-average power profiles present in the core just prior to the second interim critical test period (radial ~ 1.24 , axial ~ 1.2 , within cluster ~ 1.2), results in a predicted average ex-

posure of the peak cluster of approximately 14,900 MWd/tonne. The exposure for the central 1/2 in. of the hottest rod in the peak element is predicted to be approximately 21,500 MWd/tonne in the 55-element batch core loading at the end of its reactivity lifetime.

A power map of the core has been determined by gamma scanning one irradiated cluster from each of the seven rings of channels in the batch loading. The details of these measurements are reported on page 3.3 of this Quarterly.

MEASUREMENT OF LARGE NEGATIVE REACTIVITIES WITH THE ENDOGENEOUS PULSE SOURCE METHOD

P. Loizzo* and N. Pacilio*

In the framework of the Joint USAEC-CNEN program in the PRCP, provision is made for the measurement of the worth of cruciform control rods in a UO₂ and in a UO₂-PuO₂ core. The problems related to these experiments were of two types: a problem due to the experimental conditions and one due to the interpretation of the experiments.

The very high spontaneous fission background makes it very difficult to detect correlated bursts of prompt neutrons in the critical α measurements. The ideal power for this kind of measurement is of the order of less than 1 mW. In a UO₂-PuO₂ core (8 at.% ²⁴⁰Pu in Pu), the power generated from spontaneous fissions

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TABLE 3.II. Data from Batch Core Experiment

Total Reactor Production, MWd	Average Batch Core Exposure MWd/Tonne	Core Condition	^{10}B Concentration, wppm	Moderator Temp Coeff of Reactivity, $\Delta k/^\circ\text{F}$	Inferred Excess Reactivity, mk
0	0	55 F.E. cold, xenon-free	20.95		263
10	5.2	full core, 6 MW equilib. xenon	22.47	$+ 1.07 \times 10^{-4}$	
3369	1680	55 F.E. cold, xenon-free	17.3		229
~6300	~3060	full core, 54 MW, equilib. xenon	13.0	$+ 0.72 \times 10^{-4}$	~180
~7050	~3410	full core, 63 MW, equilib. xenon	11.5		~174
7300	3630	55 F.E. cold, xenon-free	14.05		191

is of the order of $6 \mu\text{W}$. Therefore, to have a stable reactor power with a multiplication of 3×10^4 for the measurements of α_{crit} , it is necessary to operate at a power of 0.2 W. Hence, to avoid a large dead time correction in the BF_3 counters, the efficiency must be reduced to about 100 times less than the ideal conditions. In this situation, it is very difficult to distinguish the correlated bursts (deriving from successive generations of prompt neutron families) from the uncorrelated random noise.

Some preliminary tests have been made, and it appears that a compromise is still possible between the various experimental parameters affecting the measurements. The final measurements have been scheduled for

the next quarter with ameliorated electronics. In any case, with the method of interpretation that will be used, the measurement of α_{crit} will be desirable but not necessary. For the subcritical measurements, the addition of some neutron sources will eventually be necessary for the most subcritical cases.

The plans provide for the introduction into the cores of several types of cruciform control rods with different spans: boral slabs, B_4C slabs or tubes arranged in a slab, and Hf_2O_3 slabs or tubes. The worth of the rods is expected to be of the order of 2 to 4% in reactivity. The subcritical α (decay constant of the endogenous neutron bursts) must be converted into terms of reactivity to allow a comparison

of the theoretical model for control rod calculations with the experiments.

A technique has been developed to compare the α value obtained from 2-D diffusion calculations which include the control rod directly with the α value obtained from the measurements. The core is simulated in X-Y calculations with a proper description of the geometry and of the transparency of the control rods and the following α -eigenvalue equation is solved (for example, with Exterminator II):

$$\begin{aligned}
 D_i \nabla^2 \phi_i - (\Sigma_a^i + \Sigma_r^i) \phi_i \\
 + (1-\beta) \chi_i \sum_j v \Sigma_f^j \phi_j \\
 + \sum_j \Sigma_r(j \rightarrow i) \phi_j = \alpha \left\langle \frac{1}{v_i} \right\rangle \\
 (i = 1, 2, \dots, N_j)
 \end{aligned}$$

The inverse velocity, $\langle 1/v_i \rangle$, is averaged with respect to space and energy

in the same way as Σ_a or Σ_f are averaged in our model, namely:

$$\left\langle \frac{1}{v_i} \right\rangle = \frac{\int_E \int_V \frac{\phi(E, \underline{x})}{v} dE dV}{\int_E \int_V \phi(E, \underline{x}) dE dV}$$

With this formulation, a series of fictitious experiments has been defined to assess the relationship between α and k_{eff} . In a core with four control rods of variable span, both k_{eff} and α were calculated. The core simulates future UO_2 - PuO_2 configurations planned for the PRCF, but in the experiment only one rod will be used. The results are shown in Table 3.III.

The results of Table 3.III show that the deviation from the equation

$$\rho = \frac{1 - k_{eff}}{k_{eff}} = (\alpha_{crit} - \alpha) / \alpha_{crit}$$

TABLE 3.III. Assessment of a k_{eff} , (α) Relationship

Rods and Span	k_{eff}	Calculated α	Deviations from Linearity
Prompt Critical	1.00364	0.0	-
None	1.00000	112.5	-
4 x 1.905 cm	0.95615	1511	0.5%
4 x 5.715 cm	0.90465	2885	15.9%
4 x 13.335 cm	0.83870	(5235)	(14.8%)

is about 0.5% if the reactor is less than 12\$ subcritical, and greater than 15% if the reactor is more than 28\$ subcritical. When the reactor is 50\$ subcritical or more, the reflector eigenvalue ($1/\lambda_{\text{refl}}^{\text{th}}$) is smaller than the core eigenvalue, and EXTERMINATOR converges to the smaller eigenvalue, i.e., the one that is of no use for assessing the reactivity. This fact can be very easily recognized because the second eigenfunction is very high in the reflector, while the flux distribution of the first eigenvalue is not different from the static (i.e., delayed) neutron distribution.

USAEC-CNEN JOINT PROGRAM IN THE PRCF

P. Loizzo,* R. Martinelli,* and L. D. Williams

Experiments are continuing at the Plutonium Recycle Critical Facility under the cooperative program between the USAEC and Italian-CNEN to study the problems of placing plutonium-enriched fuel in a boiling water reactor. Reactivity measurements and power distributions were made for H₂O-moderated cores uniformly loaded with two types of fuel: UO₂-PuO₂ mixtures and enriched UO₂. The plutonium rods contain 1128 g of UO₂-2 wt% PuO₂ (8 at.% ²⁴⁰Pu). The fuel is 0.508 in. diameter, 36 in. long, and clad with 0.030 in. of Zircaloy-2. The

uranium elements contain 826 g of UO₂-2.35 at.% ²³⁵U. They are 0.440 in. diameter, 36 in. long, and clad in 0.030 in. of aluminum. The rods are loaded on a 0.75 in. square lattice pitch. Reactivity measurements for the mixed oxide loadings were reported earlier.⁽¹⁾

The same core configurations were investigated for both types of fuel. The loadings, as previously described,⁽¹⁾ are uniform array, water hole, water slab, water cross, basic 9x9, basic 7x7, and basic 9x9 with voids. Some zoned loadings were also studied, but the power distributions cannot be analyzed until final power ratios are determined by absolute intercalibration measurements.

Minimum critical loading for the UO₂ fuel was about 377 rods. The excess reactivity for the UO₂ cores investigated was determined by using moderator level for reactivity compensation. The excess reactivity that each loading would have with an infinite top reflector is tabulated in Table 3.IV. The reactivity values are for an effective delayed neutron fraction of 7.64×10^{-3} .

Some effort was devoted to measuring reactivities using the endogenous pulsed source method. The measurements were terminated after it was concluded that further equipment refinements were necessary to obtain good results. Solution of these problems is in progress, and the measurement will be repeated at a later date.

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TABLE 3.IV. *Excess Reactivities for the UO_2 Loadings*

<u>Configuration</u>	<u>Number of Fuel Rods</u>	<u>Excess Reactivity, mk</u>
Regular	385	4.1 ± 0.2
Water Hole	384	3.5 ± 0.2
Water Slab	408	2.6 ± 0.2
Water Cross	420	2.0 ± 0.2
Basic 9x9	453	3.4 ± 0.2
Basic 7x7	461	1.8 ± 0.2
Basic 9x9 (voids)	477	3.6 ± 0.2
Basic 9x9 ^(a) (zoned)	373	5.5 ± 0.2

(a) *The loading had 81 of the mixed-oxide fuel elements in the center 9x9 position. The 373 includes 81 mixed oxide.*

Relative power generation rates for various types of fuels were measured in the plutonium regular loading using calorimetric methods.⁽²⁾ The elements to be compared were placed in thermally insulated thimbles which were installed at core positions of equal flux. The heatup rate of the fuel was measured at a equilibrium power level of 1 kW by thermocouples attached to the fuel cladding. The procedure was repeated at least three times for each pair of fuel rods.

In addition to the two standard fuel types, UO_2 -4 wt% PuO_2 (16 at.% ^{240}Pu), UO_2 -0.9 wt% PuO_2 (8 at.% ^{240}Pu), and UO_2 -2 wt% PuO_2 (24 at.% ^{240}Pu) fuel elements were also compared. The relative cladding heatup rate (dT/dt) ratios determined by the calorimetric technique are summarized in Table 3.V. The large uncertainties on the ratios are due to the fact that the sensitivity of the readout equipment was low relative to

the voltage change resulting from cladding heatup. Further measurements are planned using the same technique, but with increased reactor power.

Following the calorimetric measurements, the relative fission product decay rates of the different fuels were determined during the time period from 3 to 7 hr following shutdown. Analysis of these results has not yet been completed.

After 15 days and up to about 20 days after irradiation, the rods were gamma analyzed for their relative ^{140}La activity. The integral counts under the ^{140}La peaks at half-maximum were compared for each fuel combination. The ratio of the integral counts coupled with the fission yields of ^{140}La is the fission rate ratio of the fuel rods. The ratio of the ^{140}La activity in each fuel rod pair is seen in Table 3.V for comparison with the results from calorimetric measurements.

TABLE 3.V. Ratios of Cladding Heat-Up Rates
and ^{140}La Activities for Compared Fuel Rods (a)

Fuels Compared	$(dT/dt)_{\text{clad}}$ Ratio	^{140}La Activity Ratio
UO_2 -2 wt% PuO_2 (8% ^{240}Pu) UO_2 -2.35 wt% ^{235}U	1.47 ± 0.05	1.354 ± 0.015
UO_2 -2 wt% PuO_2 (8% ^{240}Pu) UO_2 -2 wt% PuO_2 (24% ^{240}Pu)	1.17 ± 0.06	1.129 ± 0.008
UO_2 -2 wt% PuO_2 (24% ^{240}Pu) UO_2 -2.35 wt% ^{235}U	1.29 ± 0.06	1.207 ± 0.014
UO_2 -2 wt% PuO_2 (8% ^{240}Pu) UO_2 -4 wt% PuO_2 (16% ^{240}Pu)	$0.78 \pm 0.06^{(b)}$	0.906 ± 0.004
UO_2 -2 wt% PuO_2 (8% ^{240}Pu) UO_2 -0.9 wt% PuO_2 (8% ^{240}Pu)	$1.51 \pm 0.06^{(b)}$	1.501 ± 0.017

(a) These ratios have not been converted to actual power generation ratios, because the accuracy of the calorimetric measurements is too poor for intercalibration experiments. The measurements will be repeated at a higher power.

(b) Estimated uncertainties, since only one measurement was made.

Power distributions for each loading were measured by gamma scanning fuel rods following an irradiation of about 0.25 kW hr. Automatic decay correction was achieved using a dual counting system for the gamma scanning. In most cases, at least three symmetrical fuel rods were measured for each point of interest. The experimental uncertainty for a given location has been defined as one-half

of the maximum dispersion of the data from symmetric rods. The "Relative Accuracy of the Experiment" as listed in Tables 3.VI and 3.VII, has been defined as the average of the experimental uncertainties from all locations in a given irradiation:

$$\text{R.A.E.} = \frac{1}{N} \sum_{i=1}^N (x_{\text{max}} - x_{\text{min}})_i$$

TABLE 3.VI. Theoretical-Experimental Correlation for UO_2 -2 wt% PuO_2 (8% ^{240}Pu) Cores

Core Configuration	Calculational Model		Reactivity			Power Distribution			Figure of Merit $\sigma_{e-c}, \%$
	Number of Groups	Number of mesh/rod	k_{calc}	k_{meas}	$k_E - k_C, mk$	Number of Exp. Points	Relative Accuracy of the Experiment, %	Number of Points Compared	
Regular	4	4	1.0035	1.0002	-3.3	19	0.62	20	0.84
	5	4	1.0018		-1.6				
Water Hole	4	4	1.0048	1.0015	-3.3	14	0.80	20	1.29
	5	4	1.0031		-1.6			20	1.32
Water Cross	4	4	1.0108	1.0076	-3.2	13	1.25	25	1.71
	4	9	1.0092		-1.6			25	1.69
	5	4	1.0109		-3.3			25	2.01
Water Slab	4	4	1.0095	1.0066	-2.9	22	0.75	27	1.08
	5	4	1.0087		-2.1			27	1.35
Basic 7 x 7	4	4	1.0080	1.0031	-4.9	8	(0.35)	22	1.25
	5	4	1.0086		-4.5			22	0.78
Basic 9 x 9	4	4	1.0120	1.0082	-3.8	8	0.86	26	1.19
	4	9	1.0090		-0.8			26	1.12
	5	4	1.0122		-4.0			26	1.35
Basic 9 x 9 Voids	4	4	1.0055	0.9994	-6.1	10	1.30	26	2.04
	5	4	1.0061		-6.7			26	2.26

TABLE 3.VII. Theoretical-Experimental Correlation for UO_2 (2.35 at.% ^{235}U) Cores

Core Configuration	Calculational Model		Reactivity			Power Distribution			
	Number of Groups	Number of mesh/rod	k_{calc}	k_{meas}	$k_E - k_C, mk$	Number of Exp. Points	Relative Accuracy of the Experiment, %	Number of Points Compared	Figure of Merit $\sigma_{e-c}, \%$
Regular	4	4	0.9999	1.0041	-4.2	28	0.53	28	0.73
	5	4	0.9990		-5.1			28	1.11
Water Hole	4	4	0.9994	1.0035	-4.1	30	0.58	30	0.56
	5	4	-		-			-	-
Water Cross	4	4	0.9990	1.0020	-3.0	31	0.40	31	0.70
	5	4	0.9998		-2.2			31	1.34
Water Slab	4	4	0.9993	1.0026	-3.3	20	0.37	26	0.83
	5	4	0.9991		-3.5			26	1.29
Basic 7 x 7	4	4	1.0014	1.0018	-0.4	28	0.47	28	1.00
	5	4	-		-			28	-
Basic 9 x 9	4	4	1.0039	1.0034	+0.5	33	0.46	33	0.71
	5	4	1.0057		+2.3			33	1.32
Basic 9 x 9 Voids	4	4	-	1.0036	-	18	0.40	-	-
	5	4	-		-			-	-

Power distributions for the various core configurations were calculated using EXTERMINATOR II.⁽³⁾ Several parameters were varied to determine the optimum model for agreement with the experimental measurements. The energy groups were varied between 2 and 5 and the number of mesh points used to describe each fuel element was varied between 1 and 9. In a few cases, the cross sections from RIBOT-5⁽⁴⁾ were varied to check the sensitivity of the calculations to the cross-section values. In all cases, the latest version of RIBOT was used, i.e., library with $\eta_{2200} (^{239}\text{Pu}) = 2.079$, (Schmidt's KFK-120 library) and a fission resonance integral of 294 b for ^{239}Pu .

Each experimental observation, instead of the average value from symmetrical points, was compared to calculations. This placed more importance on the rods where symmetry

checks were made. In most cases, a few of the experimental points had to be discarded because they were out of the range of experimental error (set at about 2%) when compared to symmetrical rods. A few experimental points were also discarded when there were unreasonably large discrepancies and surrounding rods agreed with calculations. Typically, out of 25 to 35 experimentally measured rods, 3 or 4 are discarded for the first reason and 2 or 3 for the latter. The outliers in the irradiation measurements are believed to be caused by irregularities in the fuel rods (nonhomogeneous fuel material, rods not perfectly straight, etc.).

The sensitivity of the calculation-measurement comparison to the number of energy groups and the number of mesh points is shown in Table 3.VIII for a 9x9 rod simulated bundle of UO_2 -2 wt% PuO_2 fuel rods in the center of the core.

TABLE 3.VIII. Calculation Versus Experiments
for the 9x9 Basic UO_2 - PuO_2 Core, Experimental
 $k_{\text{eff}} = 1.0082 \pm 0.0005$

Number of Groups	Mesh/Rod	Calc. k_{eff}	$\sigma_{\text{e-c}}, \%$	Running Time on Univac 1108, sec
5 (EXTR.II) ⁽³⁾	9	1.0089	1.36	740
5 (EXTR.II)	4	1.0122	1.35	305
5 (EXTR.II)	1	1.0281	2.59	52
4 (EXTR.II)	9	1.0090	1.12	654
4 (EXTR.II)	4	1.0120	1.19	274
4 (EXTR.II)	1	1.0273	2.69	41
2 (EXTR.II)	9	1.0188	1.18	300
2 (BURNY) ⁽⁵⁾	9	1.0135	1.48	118
2 (BURNY)	4	1.0174	1.67	62
2 (BURNY)	1	1.0352	3.12	24

The figure of merit selected for the comparison of an experiment with a calculation was the standard deviation of the distribution of the fractional difference at each point between experiment and calculation:

$$\sigma_{e-c} = \left\{ \frac{\sum_{i=1}^N \left[\left(\frac{x_e - x_c}{x_e} \right)_i - \left(\frac{x_e - x_c}{x_e} \right) \right]^2}{N-1} \right\}^{1/2}$$

Also listed in Table 3.VIII are the running times on the Univac 1108 for each calculation.

It can be seen from Table 3.VIII that the parameter that increases the computer time most is the number of mesh points per rod. For both 4 and 5 energy groups, calculations for 1 mesh/rod over-estimates the reactivity by 2% in k and increases the figure of merit for the power distribution comparison by a factor of 2 over the 9 mesh/rod calculation. The calculation with 9 mesh/rod yields a difference of reactivity of only 0.4%, compared with the 4 mesh/rod calculation, and an insignificant difference is seen in the figure of merit for the power distribution comparison between the 4 mesh/rod and 9 mesh/rod.

It can be noted that the calculation with 1 thermal group (2 or 4 energy groups) yields a better power distribution than those with 2 thermal groups. The 2 thermal group model tends to over-estimate the rod power at the water gaps by approximately 2%, but the reason for this result is not

fully understood. It is believed that some unknown errors (i.e., the ratio of epithermal removal cross sections between H_2O and smeared fuel region, or a similar cause) are compensated for in the intrinsically less accurate 4 group model and are not as well compensated in the more precise 5 group model.

The larger figures of merit for the power distribution comparisons calculated from the code BURNY⁽⁵⁾ resulted because, at the time of calculation, an old RIBOT library was used instead of the new one used in the RIBOT-5 + EXTERMINATOR calculations.

The 4 mesh/rod model, as seen from Table 3.VIII, adequately fits the experimental results with reasonable computing cost. Calculations using this rod model and both 4 and 5 energy groups are compared against the UO_2 -2 wt% PuO_2 (8 wt% ^{240}Pu) experiments in Table 3.VI. Both energy group models adequately calculate reactivities (within about 4 mk) and reactivity changes (i.e., a difference of 0.5 mk on an 8 mk change between the regular core and the basic 9x9). Both the 4 and 5 energy group models adequately describe the power distributions.

The only case where it was felt that the model did not satisfy the core condition was in the basic 9x9 with voids where the figure of merit is consistently larger than the other measurements. Since the major discrepancies in the power shapes were in the voided region of the core, it is felt that the cause is inherent in

the description of the voids. Calculations are in progress to investigate this.

The results of some of the UO_2 loadings are shown in Table 3.VII. The same models were used, and generally the same conclusions found in the mixed-oxide loadings apply to the uranium loadings. The reactivity calculations for UO_2 configurations were adequate using the 4 group model. The reactivity differences between calculations and experiments varied from -4.2 mk to +0.5 mk. A comparison of the measured and calculated power distributions for a uniform core of UO_2 -2.35 at.% ^{235}U is shown in Figure 3.4.

The precision of the measurements for the 4 group model better fits the UO_2 configuration than the mixed-oxide configuration since the typical maximum figure of merit for the power distributions (95% confidence interval) is about 1.5% on the former and 2.5% for the latter. The increased dispersion in the mixed-oxide loadings may be due to the larger discrepancies in the fuel rod compositions as seen in symmetry checks. It also may be a systematic effect enhanced by the non-uniformity of the configurations. For example, compare the figures of merit for the water cross, the basic 9x9 with void, and the uniform loading, in Table 3.VI. Since the measurement-calculation dispersions are not grossly different from the statistical experimental error, a definite conclusion as to whether the differences are random or systematic cannot be reached.

A few of the plutonium loadings will be repeated later with selected rods to study this question further.

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ANALYSIS OF TEMPERATURE COEFFICIENTS

V. O. Uotinen

In recent years, temperature coefficients of reactivity have been measured in a number of PuO_2 - UO_2 - H_2O lattices. Experiments with UO_2 -1.5 wt% PuO_2 ⁽¹⁾ and UO_2 -2 wt% PuO_2 ⁽²⁾ fuel rods have been conducted by Battelle-Northwest in the Plutonium Recycle Critical Facility (PRCF). Experiments with UO_2 -6.6 wt% PuO_2 fuel rods⁽³⁾ have been conducted by Westinghouse at the Westinghouse Reactor Evaluation Center (WREC). A preliminary analysis of these temperature

FIGURE 3.4. Comparison of Experimental Power Distribution with Calculated Power Distribution for a Uniform Core of UO_2 -2.35 at.% ^{235}U

coefficients was presented at the November 1967 ANS meeting.⁽⁴⁾ This analysis has recently been extended to include a systematic study of the importance of various effects on the temperature coefficient.

The following results pertain to the average temperature coefficient as calculated for a temperature change from 20.4 to 72 °C.

Thermal Expansion of Lattice

The importance of the thermal expansion effect for any particular lattice depends on the material from which the lattice plates are constructed and the lattice pitch. This effect is especially important ($\sim +6 \times 10^{-5} \Delta k/k/^\circ\text{C}$) for the undermoderated PRCF experiments where the lattice plates are constructed from Lucite or Lexan; for the undermoderated WREC experiment, where the plates are of aluminum, the thermal expansion effect is somewhat smaller ($\sim +2 \times 10^{-5} \Delta k/k/^\circ\text{C}$).

Moderator Scattering Kernel

If one neglects the change in moderator scattering kernel with temperature and assumes that only the water density changes with temperature, calculated moderator temperature coefficients are significantly less negative. The change in moderator temperature coefficient varies from $\sim 1 \times 10^{-5} \Delta k/k/^\circ\text{C}$ for lattice of UO_2 -2 wt% PuO_2 rods to $\sim 13 \times 10^{-5} \Delta k/k/^\circ\text{C}$ for a lattice of UO_2 -6.6 wt% PuO_2 rods.

The few cores that have been studied do not constitute a systematic study of this effect, but it appears that in general the temperature-dependence of the moderator scattering kernel becomes more important as the amount of plutonium increases. Also, one would expect this effect to be more important at tight lattice pitches than at loose pitches.

Anisotropic Scattering

Making a correction to the hydrogen scattering kernel for anisotropic scattering changes calculated moderator temperature coefficients by $\sim 1-2 \times 10^{-5} \Delta k/k/^\circ\text{C}$ in the negative direction. This effect, although not as large as some of the other effects, is not negligible.

Temperature-Dependence of Reflector (Radial Buckling Effects)

When analyzing experiments where the reflector temperature is the same as the core temperature, it is very important to take into account the temperature dependence of the reflector. If one neglects this effect and assumes that the reflector remains at room temperature, calculated temperature coefficients are much more negative. This effect is roughly proportional to the buckling, ranging from $4.8 \times 10^{-5} \Delta k/k/^\circ\text{C}$ for a core whose buckling is about 70 m^{-2} to $9.5 \times 10^{-5} \Delta k/k/^\circ\text{C}$ for a core whose buckling is about 130 m^{-2} .

Temperature-Dependent Transverse Buckling

If one neglects to take into account the change in transverse buckling (or reflector savings) with temperature, the calculated temperature coefficients are more negative by $\sim 1 \times 10^{-5} \Delta k/k/^\circ\text{C}$. This effect, of course, depends on the magnitude of the transverse buckling, which was $\sim 8.8 \text{ m}^{-2}$ in the cases studied.

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ZODIAC G - D. D. Matsumoto and R. H. Holeman

The third generation one-dimensional burnup code ZODIAC G is conditionally operable on the Univac 1108. ZODIAC G was accomplished by blending the HTH and ZODIAC F overlay programs, in conjunction with new flow control and service routines.

In the usual ZODIAC G burnup case, epithermal and intermediate average cross sections are computed by HRG and combined with thermal cross sections computed by the transport code THERMOS to form a data library for the diffusion code HFN. Group-dependent average fluxes from HFN are used by CLERK to compute a one-group burnup cross section for the isotope transmutation code ALCHEMY. The new concentrations from ALCHEMY are used by REFIRE to update input for all the codes on the next orbit.

The code may be used in any one of four operating modes: (1) Fast reactor burnup mode, as presently available with ZODIAC F; (2) Burnup mode, with optional recalculation of cross sections by HRG and THERMOS at any orbit; (3) HTH (HRG-THERMOS-HFN) mode, including all options presently available with HTH; and (4) Non-burnup mode, in which stacked or mixed cases for the individual codes HRG, THERMOS, and HFN may be run in any order.

PCTR MEASUREMENT OF k_∞ FOR UO_2 -0.86 wt% PuO_2 RODS IN 1.0 in. SQUARE PITCH, WATER MODERATED LATTICE

D. F. Newman and D. R. Oden

Introduction and Summary

The infinite medium neutron multiplication factor, k_∞ , has been measured in the Physical Constants Testing Reactor (PCTR) for many graphite moderated lattices.^(1,2) Measurements performed in the PCTR light water tank utilizing a uniform lattice of UO_2 -0.86 wt% PuO_2 rods in a water moderated 1.0 in. square pitch have extended the usefulness of

the PCTR technique to determine neutron characteristics in light water moderated lattices.⁽³⁾ Analysis of the measurements made with this test lattice have been completed and results of the determination of k_{∞} by (a) the null reactivity technique, and (b) measurement of the adjoint weighted excess neutron production cross section $\langle \Sigma_p \rangle$ in the unpoisoned lattice are presented in this report.

The null reactivity technique is used to determine k_{∞} by accurately measuring the amount of neutron absorber required to absorb all the excess neutrons of a test sample in the equilibrium spectrum of the lattice having this concentration of neutron absorber throughout the lattice. The required amount of absorber is determined by a null reactivity condition on the test sample.

Information about the neutron multiplication properties of a medium without having to add absorber to the medium can be obtained from the measured adjoint weighted excess neutron production cross section, $\langle \Sigma_p \rangle$.⁽⁴⁾ A value of $k_{\infty} - 1$ can be obtained from the experimentally obtained quantity $\langle \Sigma_p \rangle$ by including a small calculated leakage-dependent term.

Values of k_{∞} determined by the two independent methods mentioned above were obtained for the same water-moderated test lattice with 1.0 in. square cells containing a 0.5 in. diameter rod of UO_2 -0.86 wt% PuO_2 fuel. The experimental results are summarized in Table 3.IX.

Experimental Techniques and Results

Neutron Energy Distribution in the Lattice

Determination of the condition of spectral equilibrium necessary for the proper measurement of the infinite medium neutron multiplication factor, k_{∞} , is usually made by measuring spatial variations in a suitable integral spectral index. Measurements of the cadmium ratio of 0.020 in. diameter gold pins, placed on the fuel cladding surface throughout the test cell and buffer regions, proved to be sufficiently sensitive for this purpose in this experiment.

The radial traverse data shown in Figure 3.5 indicate that the 108 fuel rods in the inner core container were more than adequate to achieve an equilibrium neutron spectrum across the center cell and that this condition was independent of the number of natural uranium flux modifiers loaded adjacent to the test cavity. Also shown in the figure is the traverse taken in the central core container with the water in this region containing 280 wppm boron. Borated polyethylene sheets in the radial buffer region helped to bring the spectrum into equilibrium. Cadmium ratio measurements were not made in the radial buffer region for this case since the localized poison would give false spectrum indications from pins on the fuel-clad surfaces near the poison.

Axial traverses were also made in the center cell which indicated a flux

TABLE 3.IX. *Summary of Results, Water-Moderated 1.0 in. Square Lattice*
[UO₂-0.86 wt% PuO₂ (7.24 at.% ²⁴⁰Pu) 1/2 in. Diameter Rods
(0.234 at.% ²³⁵U)]

Method of Determination	Boron Concentration	k _∞	<Σ _p > _o
Null reactivity measurement with boric acid added to the water as an absorber	276.3 ± 2.8 (ppm by weight)	1.120 ± 0.005	
Unpoisoned measurement using copper as a standard neutron absorber	0.0	1.127 ± 0.006	7.8 ± 0.3 x 10 ⁻³ cm ⁻¹
Unpoisoned measurement using Σ _a ^{cell} calculated by THERMOS	0.0	1.1275	
THERMOS-HRG calculation(a)	0.0	1.100	

Thermal disadvantage factor (TDF), $(\bar{\sigma}_{1/v} \bar{\phi})_M / (\bar{\sigma}_{1/v} \bar{\phi})_F$

	Boron Concentration	TDF
Experimental Measurement	0.0	1.249 ± 0.009
THERMOS Calculation	0.0	1.282
Experimental Measurement	276.3 ± 2.8 ppm	1.231 ± 0.012
THERMOS Calculation	276.3	1.2805

Cell Averaged Thermal Macroscopic Cross Section

(normalized to 2200 m/sec cross section at the surface of the cell)

$$\left(\bar{\Sigma}_a^{\text{cell}}\right)_o = \left(\frac{\bar{\Sigma}_a^{\text{cell}}}{\bar{\sigma}_{1/v}} \frac{\sigma_o 1/v}{\text{cell surface}} \frac{\bar{\phi}^{\text{cell}}}{\phi^{\text{cell surface}}} \right) = 6.626 \times 10^{-2} \text{ cm}^{-1}$$

THERMOS Calculation

$$\left(\bar{\Sigma}_a^{\text{cell}}\right)_o = \left(\frac{A_{2a}^{\text{cell}} v_a \Sigma_{oa}}{A_{2a}^s v^{\text{cell}}} \right) = 6.65 \pm 0.28 \times 10^{-2} \text{ cm}^{-1}$$

Experimental Measurement

(a) Private communication, C. L. Bennett

FIGURE 3.5. Radial Variation of Gold Cadmium Ratio

peak of 3.7% at the ends of the cell relative to that at the center of the cell (Figure 3.10). This effect is discussed further in another section, Reaction Rate Measurements. Methods of obtaining spectral equilibrium in the axial direction are discussed in Reference 3.

Cell Reactivity Measurements

The center 4 cells which were the test sample for the reactivity mea-

surements are shown in Figure 3.6. The central core container (12 cells) encloses the center cells radially, while the two short end buffer assemblies enclose the center cell along the long axis when the assemblies are loaded into the PCTR light water tank. The center 4 cells consist of four 0.5 in. diameter fuel rods 20.0 in. long assembled in the lutice center container of inside dimensions 2.0 in. x 2.0 in. x 20.0 in. A series of reactivity measurements were taken for

FIGURE 3.6. PCTR Light Water Tank
Central Container Assembly Components

each reactor loading and integral spectral indices were measured. Reactivity worths of the following quantities were measured:

1. The center cell, ($\Delta\rho_{\text{cell}}$), including any concentration (C_B) of neutron absorber dissolved in the water;
2. A small amount of copper in the center cell void box, ($\Delta\rho_{\text{Cu}}^{\text{void}}$), when the center cell is removed;
3. A small amount of copper on the center cell boundary, ($\Delta\rho_{\text{Cu}}^{\text{boundary}}$), when the center cell is loaded;
4. A small amount of borated polyethylene sheet on the center cell boundary, ($\Delta\rho_B^{\text{boundary}}$);
5. The center cell, ($\Delta\rho_{\text{cell}}^{\Delta B}$), including a small neutron absorber concentration increase ($C_B + \Delta C_B$) dissolved in the water.

Relative Reaction Rate Measurements

The relative reaction rates of the center cell components were determined in both poisoned and unpoisoned lattices when the proper neutron energy distribution existed in the test cell. The fission rates in the ^{239}Pu and ^{235}U in the fuel spectrum were measured relative to the PCTR thermal column spectrum⁽⁶⁾ by the activation of 1/2 in. diameter $^{239}\text{PuAl}$ and $^{235}\text{U-Al}$ foils both in the separable fuel element (see Figure 3.6) and on the thermal column foil rotator. The effective subcadmium absorption rates in the 1/v-like absorber components were monitored with copper foils placed throughout the cell. These activities were corrected for epica-

dium activity by measured cadmium ratios. Bare and cadmium-covered foils were activated in separate irradiations for both the lattice containing boric acid dissolved in the water and the unpoisoned lattice. Bare foils of each type activated simultaneously in the thermal column were used to normalize specific activities of the corresponding foils of each type in the center cell.

Separable fuel elements had 0.005 in. thick Zircaloy end caps in the center to allow placement of 1/2 in. diameter foils of PuAl , U-Al , Cu , and Au close to the fuel pellets. Special copper quadrant foils were used to integrate the flux radially over the area of the water moderator. One of these quadrant foils was dissected into concentric annular regions after being irradiated, and the specific activity of each region was measured separately. The specific activities of the foils of each type were calculated using a revised version of the computer code APDAC-I.⁽⁷⁾ This code corrects the counting data for activity decay during counting, room background, foil mass variations, and residual activity. Results of the relative reaction rate measurements, including corrections for flux dipping and peaking in the foils, are shown in Figure 3.7 for the unpoisoned lattice, and in Figure 3.8 for the poisoned lattice. Axial distributions of the relative reaction rate in copper pins placed on the fuel clad surface of the unpoisoned lattice are shown in Figure 3.9. The average axial reaction rate on the fuel clad surface is less than 0.1% different

FIGURE 3.7. Sub-Cadmium Cylindricalized $1/v$ Flux Traverse $UO_2-0.86 \text{ wt\% } PuO_2$ Rods in 1.0 in. Square Pitch Water-Moderated Lattice

FIGURE 3.8. Sub-Cadmium Cylindricalized $1/v$ Flux Traverse UO₂-0.86 wt% PuO₂ Rods in 1.0 in. Square Pitch, 280 ppm Boron in Water-Moderated Lattice

FIGURE 3.9. Relative Reaction Rate in Copper Pins Positioned on the Fuel Clad Surface Axially (Unpoisoned Lattice)

from the value at the fuel clad surface in the center of the cell. Since this end effect error is much smaller than the uncertainty in the foil activation measurements ($\sim 0.5\%$), the additional correction for end effects, which amounts to less than 0.1 mk in the final result, was neglected.

Determination of Boron Concentration in Water

Since the infinite medium excess neutron multiplication, $k_{\infty} - 1$, is directly proportional to the poison concentration, a precise determination of the latter is necessary for precise measurements of $k_{\infty} - 1$. To obtain a given boron concentration in solution the desired amount of boric acid crystals are weighed out and combined with an accurately determined volume of distilled water. The weight and volume determinations introduce only about a 0.1% error in the boron concentration. However, boric acid crystals are hygroscopic and the weight of the crystals, and hence the boron content, will depend on the amount of moisture contained in the crystals, which will vary from sample to sample.

In order to determine the boron content of solutions with these unknown combined uncertainties, a calibration curve has been obtained from a series of reactivity measurements in the Thermal Test Reactor (TTR). Reactivity worths of 14 samples of different nominal boron concentrations, supplied by Chemical Standards - BNW, were measured relative to a pure distilled water sample. A sample size of

50 cm³ contained in thin-walled aluminum cans was found to allow measurements over a concentration range of 0 to 2000 wppm (weight parts per million) without the necessity of control rod calibrations.

Measurements on the samples of a given nominal boron concentration were alternated with the water standard. Data for the determination of the water standard reactivity at the time of measurement of the boron-containing sample could thus be determined. This correction for reactor drift was obtained from a linear least squares fit of the water standard reactivity as a function of time during each half-day series of runs. The boron worth of a given sample was then the difference between the worth of the boron-containing sample and that of the water standard corrected to the time of measurement of the sample. The results of individual determinations were then used to get an average boron worth at a given concentration, and the uncertainty in this worth was found using standard statistical techniques.

A calibration curve was then made by least squares fitting the function $y = ax + bx^2$ to the data, using the multidimensional error analysis BNW program LIKELY.⁽⁵⁾ Using the average values of boron worths and uncertainties calculated therefrom, and nominal boron concentrations, percentage errors in boron concentrations were uniformly adjusted until the χ^2 of the fit agreed with the number of degrees of freedom associated with the problem. Thus, if the uncertainties in reactivities are assumed

to be known exactly, the above condition was met for an uncertainty in boron concentration of approximately 1%.

The best fit parameters for the function $y = ax + bx^2$ were found to be:

$$\begin{aligned} a &= 83.81 & \sigma_a^2 &= 0.4887 \\ b &= 0.8410 & \sigma_b^2 &= 0.00333 \\ & & \sigma_{ab} &= -0.03552 \end{aligned}$$

where,

y = wppm boron in sample;

x = reactivity worth of boron in cents.

The uncertainty in a boron concentration obtained using the calibration curve is by propagation of errors,

$$\sigma_x = \left[x^2 \sigma_a^2 + (a + 2bx)^2 \sigma_x^2 + x^4 \sigma_b^2 + 2x^3 \sigma_{ab} \right]^{1/2} \quad (1)$$

where x and σ_x^2 are determined from TTR reactivity measurements and the other quantities are as given above.

Boron concentrations used in this experiment were determined from the calibration curve to be 280.2 ± 1.8 and 249.3 ± 1.7 as compared to nominal values of 275 and 250 wppm, respectively.

Measurement of Concentration of Boron in Water to Achieve Null Reactivity

Each series of cell reactivity measurements gave an estimate of the concentration of boron in water to absorb the excess neutrons of the

test cell $[(C_B)_0]$. The worth of the center cell ($\Delta\rho_{cell}$) including any concentration (C_B) of boron dissolved in the water, normalized to the worth of a small strip of copper in the center cell void box per gram of copper ($S_{Cu}^{void} = \Delta\rho_{Cu}^{void}/M_{Cu}^{void}$), was measured both in the unpoisoned lattice and the poisoned lattice. Results of these measurements are shown in Figure 3.10 with a linear least squares fit to the data. The null reactivity condition is achieved when the cell worth is zero at the correct boron concentration $(C_B)_0$ in the poisoned lattice. As shown in Figure 3.11, the value for $(C_B)_0$ is 276.3 ± 2.8 wppm boron. Boron concentrations used in the water of the poisoned lattice were 280.2 ± 1.8 ppm and 249.3 ± 1.7 ppm determined from measurements in the TTR. Linear interpolation in this narrow concentration range, in Figure 3.11, was used to determine $(C_B)_0$.

Use of Borated Polyethylene in Outer Radial Buffer

In the radial buffer region outside of the central 16 cells that contained boric acid solutions, consideration of a neutron poison in place of boric acid led to the selection of 0.005 in. thick sheets of borated polyethylene (~4 wt% boron carbide). Advantages of the use of this material over boric acid are:

1. There is essentially no boron contamination of water pumps, piping, and storage drums as there might be with boric acid.

FIGURE 3.10. Concentration of Boron in Water as a Function of Cell Worth Normalized to the Worth of Copper in the Void

FIGURE 3.11. Concentration of Boron in Water to Achieve $k_{\infty} = 1.0$

2. Safety is improved because loss of water does not mean loss of boron.
3. Boron concentration can be changed more easily and economically relative to poisoning the whole tank with boric acid.

It is superior to most other solid neutron poisons (e.g., copper) due to its relatively high blackness. Thus there is less water displacement since the specific absorption per unit volume is high. In addition, the water is being replaced mostly by polyethylene, a hydrogenous material. Also, its affect on the neutron energy spectrum is the same as that of the boric

acid, since in both cases, the predominant absorber element is boron.

The amount of material required in the radial buffer for a given boron concentration in the center cell was determined by reactivity measurements. The worth of borated polyethylene on the cell boundaries relative to the worth of boric acid in the center cell was used to find the thickness of borated polyethylene equivalent to a given concentration of boron in the boric acid. Gold-cadmium ratios in the central core container with borated polyethylene placed uniformly at the cell boundaries throughout the radial buffer region indicated that spectral equilibrium was sufficiently approached in the center cell (see Figure 3.5).

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GROUP-AVERAGED RESONANCE PARAMETERS

J. H. Nail

A study of a method of calculating energy group-averaged resonance parameters is being conducted. The result has been the definition of a consistent method of accurately calculating effective resonance integrals from average parameters. This has immediate application to the ALTHAEA code where heretofore some of these parameters were derived by inference from comparing calculated burnup data to experimental data. It is intended that these numbers could be supplied by the GANDS⁽²⁾ code based on the parameters supplied from the BNW Master Library.⁽³⁾

The resonance model used in the derivation was the Narrow Resonance-Infinite Absorber (NRIA) model. The effective resonance integral for this model is written as

$$I = \int \frac{\sigma_a(u') du'}{1 + \sigma_a(u')/\sigma_m} \quad (1)$$

where

u' = lethargy

σ_a = lethargy dependent absorption cross section

$\sigma_m = (N_F \bar{S})^{-1}$ for heterogeneous configurations

N_F = fuel isotope concentration

$\bar{S}$ = average chord length of fuel element.

Use of the Breit-Wigner formulas for resonance peaks results in

$$I = \frac{I^\infty}{\left[1 + \frac{\sigma_o \Gamma_\gamma}{\sigma_m \Gamma}\right]^{1/2}} \quad (2)$$

where

I^∞ = infinitely dilute resonance integral

σ_o = resonance peak

Γ_γ = capture width

Γ = total width.

Next the total effective resonance integral is defined to be

$$I_{\text{total}} = \sum_{i=1}^n \frac{I^\infty_i}{\left[1 + \sigma_{\text{max}}/\sigma_m\right]^{1/2}_i} \quad (3)$$

where

I^∞_i = infinitely dilute resonance integrals for each resonance

$\sigma_{\text{max}} = \sigma_o \Gamma_\gamma / \Gamma$

n = number of resonances.

In order to define the total effective resonance integral using averaged parameters, it is necessary to define the form of the equation. This may be done by defining

$$I_{\text{total}} = \frac{(I^\infty)_{\text{total}}}{\left[1 + Z/\sigma_m\right]^{1/2}} \quad (4)$$

so that the form of the effective resonance integral equation is preserved. Next, it seems very desirable to preserve the infinite dilution behavior, i.e., the total infinitely dilute resonance integral is the sum for all of the resonances. This means that $(I^\infty)_{\text{total}}$ is defined as

$$(I^\infty)_{\text{total}} = \sum_{i=1}^n I^\infty_i \quad (5)$$

When Equations (3) and (4) are equated and Equation (5) is substituted, last unknown parameter, Z , is uniquely defined. This results in the following equation for Z :

$$Z = \left\{ \left[\sum_{i=1}^n I^\infty_i \right] \div \sum_{i=1}^n \left[\frac{I^\infty_i}{1 + (\sigma_{\text{max}}/\sigma_m)_i} \right]^{1/2} - 1 \right\} \sigma_m \quad (6)$$

Note that since σ_m is a function of fuel concentration, the value of Z is not independent of concentration.

In order to keep the number of variables manageable, a number of simple situations with two resonances have been investigated to see what general behavior may be expected and if any simpler equations are possible. The combinations included several combinations of sizes. These variations were on the relative size of resonance peaks and infinitely dilute resonance integrals. Table 3.X shows

the combinations considered, and while these are not all of the kinds of combinations one could think of, these seem likely. This is based on a consideration of the implications as to resonance sizes inherent in the Breit-Wigner equations.

TABLE 3.X. Some Types of Resonance Combinations

Combination	Combination of Peak Sizes	Combination Resonance Integrals
1	$X_1 \gg X_2^{(a)}$	$I_1^\infty \gg I_2^\infty$
2	$X_2 \gg X_1$	$I_2^\infty \gg I_1^\infty$
3	$X_1 \gg X_2$	$I_1^\infty \approx I_2^\infty$
4	$X_1 \approx X_2$	$I_1^\infty \approx I_2^\infty$
5	$X_2 \gg X_1$	$I_1^\infty \approx I_2^\infty$
6	$X_1 \approx X_2$	$I_1^\infty \gg I_2^\infty$
7	$X_1 \approx X_2$	$I_2^\infty \gg I_1^\infty$

(a) $X = \sigma_{\text{max}}/\sigma_m$

Table 3.XI shows some example calculations demonstrating these combinations. Some of these resonances are based on real resonances, and others were generated in order to conveniently demonstrate the point.

It turns out that resonance combinations 4, 6, and 7 are almost trivial as the peaks are about equal and Z is insensitive to the sizes of the infinitely dilute resonance integrals. This does not mean that the effective resonance integral is not sensitive to the sizes of the infinitely dilute resonance integrals as can be seen by examining Equations (4) and (5). The

TABLE 3.XI. Resonance Parameters for Seven Combinations of Resonance Sizes

Approximation	Energy, eV	$\sigma_{\max}$, b	I^∞ , b	Theoretical Z ^(a) Values, b		
				$1/\sigma_m = 1 \times 10^{-3}$	1×10^{-5}	1×10^{-7}
$I_1^\infty \gg I_2^\infty$	8.79	577.68	13.21	501.8	518.6	518.6
$X_1 \gg X_2$	9.28	68.23	1.732			
$I_1^\infty \ll I_2^\infty$	8.79	72	1.93	463.5	480.4	480.6
$X_1 \ll X_2$	9.28	547	11.86			
$I_1^\infty \approx I_2^\infty$	8.50	155.2	2.868	99.4	101.4	101.4
$X_1 \gg X_2$	8.70	47.6	2.869			
$I_1^\infty \approx I_2^\infty$	8.50	155.2	2.868	157.0	157.0	157.0
$X_1 \approx X_2$	8.70	158.9	2.869			
$I_1^\infty \approx I_2^\infty$	8.50	155.2	2.868	592.8	766.1	766.4
$X_1 \gg X_2$	8.70	1440.3	2.601			
$I_1^\infty \gg I_2^\infty$	8.50	155.2	2.868	155.2	155.2	155.2
$X_1 \approx X_2$	9.00	155.1	1.624			
$I_1^\infty \ll I_2^\infty$	8.50	155.2	2.868	155.1	155.1	155.1
$X_1 \approx X_2$	9.00	155.0	11.059			

(a) Z = Theoretical group averaged capture resonance peak heights.

value of Z in the other cases where the peaks are not equal is dependent on the infinitely dilute resonance integrals as can be seen by comparing combinations 1, 2, 3, and 5.

Many different approximations have been studied to see if any have general applicability. Two approaches have shown promise as forms that may simplify the application of this method. The first is based on an expansion of the term in parenthesis in a revised form of Equation (6).

When the first order in this expansion is carried, the resulting equation for Z is:

$$Z = (\sigma_{\max}) \left[1 - 2 \sum_{\substack{k=1 \\ k \neq \ell}}^n \frac{I_k^\infty \sqrt{1 + x_\ell}}{I_\ell^\infty \sqrt{1 + x_k}} + 2 \sum_{\substack{k=1 \\ k \neq \ell}}^n \frac{I_k^\infty}{I_\ell^\infty} \right] + 2 \left[\sum_{\substack{k=1 \\ k \neq \ell}}^n \frac{I_k^\infty}{I_\ell^\infty} - \sum_{\substack{k=1 \\ k \neq \ell}}^n \frac{I_k^\infty \sqrt{HX_2}}{I_\ell^\infty \sqrt{1 + x_k}} \right] \sigma_m, \quad (7)$$

where

ℓ = subscript of largest resonance

$X = \sigma_{\max}/\sigma_m$ for each resonance.

Essentially, this assumes a strong dominance of one resonance as the expansion is based on normalizing to this resonance.

Another form that has resulted is based on those cases where the infinitely dilute resonance integrals are about equal, but the resonance peaks are not. This equation is as follows:

$$Z \approx \{ [HM]^2 - 1 \} \sigma_m \quad (8)$$

where

HM = harmonic mean of all shielding factors

$$= \frac{n}{\frac{1}{\sqrt{1+X_1}} + \frac{1}{\sqrt{1+X_2}} + \dots} \quad (9)$$

This form can be further simplified for the case where X is always somewhat less than 1.0 (small peaks or very low nuclear densities), so that the harmonic mean can be reduced to

$$HM \approx 1 + 1/2 \frac{X_1 + X_2 + \dots}{n} \quad (10)$$

and

$$Z \approx \frac{(\sigma_{\max})^1 + (\sigma_{\max})^2 + \dots}{n} \quad (11)$$

Some conclusions so far are that (1) where there is one resonance significantly larger than all others, it will dominate and largely determine Z; (2) where there is no clear dominance the mean shielding factor ($\sqrt{1 + Z/\sigma_m}$) is close to being an average of all the shielding factors; and (3) most, though not all, of

these calculated values are significantly dependent on fuel concentration. Equations (7) and (8) are attempts to cover the gray area between these first two situations in simplified forms that have been shown to have intuitive meaning.

The next step is to apply this method to real isotopes. Tables 3.XII and 3.XIII show calculations of group-averaged capture resonance Z values (theoretical values) for ^{235}U and ^{240}Pu . These are currently based on the parameters from BNL-325, 2nd Edition.⁽⁴⁾ For ^{235}U neither of the approximate forms can describe the Z value accurately. However, the harmonic mean is much better than the expansion based on assuming a single large peak. This would be expected from examining the ^{235}U resonances which have a number of large peaks that have significant dilute resonance integrals mixed among a large number of smaller resonances. Thus, one might expect an average to be roughly correct but be somewhat off due to the mixture of sizes.

The situation for ^{240}Pu is just the reverse, as this isotope has one resonance at 1.056 eV, which is significantly larger than all others. Thus, the result that the expansion based on a single large resonance would be quite good is expected. Also, as expected, the harmonic mean is not an accurate approach for this isotope.

These are, of course, not all of the factors considered in this study but serve to demonstrate that one can calculate (energy) group-averaged values of resonance parameters. Also,

TABLE 3.XII. Group Averaged Resonance Peak Values for ^{235}U

$\frac{1}{\sigma_m}$, nuclei b	Group Averaged Capture Resonance Peak Heights, b		
	Z (a)	HM (b)	LP (c)
10^{-3}	425	217	2213
10^{-4}	512	256	3324
10^{-5}	526	262	3529
10^{-6}	527	263	3551
10^{-7}	527	263	3553
10^{-8}	527	263	3553

(a) Z = Theoretical group averaged capture resonance peak height.

(b) HM = Approximate Z value resulting from assuming the harmonic mean.

(c) LP = Approximate Z value resulting from assuming the expansion based on a single large resonance.

TABLE 3.XIII. Group Averaged Resonance Peak Values for ^{240}Pu

$\frac{1}{\sigma_m}$, nuclei b	Group Averaged Capture Resonance Peak Heights, b		
	Z (a)	HM (b)	LP (c)
10^{-3}	151568	8634	150709
10^{-4}	155181	10436	154755
10^{-5}	159954	16983	159870
10^{-6}	161412	23570	161382
10^{-7}	161603	24995	161578
10^{-8}	161622	25155	161598

(a) Z = Theoretical group averaged capture resonance peak height.

(b) HM = Approximate Z value resulting from assuming the harmonic mean.

(c) LP = Approximate Z value resulting from assuming the expansion based on a single large resonance.

based on the investigation of simple resonance combinations one can predict and interpret the values of the parameters calculated by this method, and these simplified forms have suggested some equations which have an intuitive basis and point to simpler forms of the theoretical equation for Z, the group-averaged resonance peak. Finally, this study has shown that for some isotopes frequently one cannot use a constant value for Z. Rather it may be a function of the fuel isotope concentration.

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4.0 PLUTONIUM UTILIZATION STUDIES

ANALYSES OF FUEL FABRICATION COSTS

J. B. Burnham, P. Loizzo,* and
G. L. Simmons

The cost of fabricating fuel assemblies for a PWR has been studied in detail⁽¹⁾ for both mixed-oxide ($\text{PuO}_2\text{-UO}_2$) and enriched uranium (UO_2) fuel prepared by the vibratory compaction and pelletized process. Burnham⁽²⁾ has used this work to obtain formulae for the cost components of the fabrication of the same PWR fuel assembly with vibratory-compacted high exposure mixed-oxide fuel. We have extended this work to include enriched uranium PWR fuel

assemblies as well as UO_2 and $\text{PuO}_2\text{-UO}_2$ BWR fuel assemblies. Also included in this correlation is a situation in which the fuel assembly must contain rods of more than one enrichment.

The derived equations allow one to obtain the cost of fuel element fabrication as a function of rod size, fuel density, plutonium content, enrichment, interest rates, use charge, and number of extra enrichments per assembly. A document is being prepared that gives the details of these equations.

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2. J. B. Burnham. *Private Communication*, Pacific Northwest Laboratory, Richland, Washington.

* Exchange Scientist from CNEN (Comitato Nazionale Energia Nucleare) Cassacia - Rome, Italy under cooperative between USAEC and CNEN.

5.0 PRTR OPERATION

CORROSION OF AND BORON DEPOSITION ON ALUMINUM SURFACES IN THE PRTR MODERATOR SYSTEM - C. W. Pollock

A continuing satisfactory condition of aluminum surfaces in the PRTR moderator system is indicated by data obtained from aluminum corrosion coupons which were removed on 12-14-67.

A decrease in boron held by aluminum surfaces is apparent. Boron deposition on fourteen specimens that were subjected to intermittent immersion in the 20 ppm boron, D₂O moderator fluid varied from a low value of 59 $\mu\text{g B/dm}^2$ for a 5050 alloy coupon (30 months in test) to a high value of 140 $\mu\text{g B/dm}^2$ for a 6061 alloy coupon (15 months exposure). The average concentration found on the fourteen coupons was 96 $\mu\text{g B/dm}^2$. These values are less than those observed on eight comparable coupons withdrawn 6-8-67, where the lowest value was 108 $\mu\text{g B/dm}^2$ for a 5050 alloy after 23 months in test and the highest was 183 $\mu\text{g B/dm}^2$ for a 6063 alloy, also after 23 months. The average value in this group was 140 $\mu\text{g B/dm}^2$. Alloys in the group were 5050, 5052, 6061, and 6063, representing various moderator surfaces.

Deposits of less than 300 $\mu\text{g B/dm}^2$ are considered within tolerable reactivity control limits for the PRTR.*

* G. R. Bloom and T. F. Demmitt. *Corrosion and Boron Deposition in Boric Acid Solutions*, BNWL-520, Pacific Northwest Laboratory, Richland, Washington. December 1967.

Boron deposition on nine coupons above the reactor fluid in the helium gas blanket showed extreme variability, ranging from 22 $\mu\text{g B/dm}^2$ for a 6061 alloy after 6 months in test to 305 $\mu\text{g B/dm}^2$ for a 6061 alloy at 19 months. An average for nine coupons was 111 $\mu\text{g B/dm}^2$, four being in test 6 months, four for 30 months, and one for 19 months. Only two coupons from the gas blanket on 6-8-67 are available for comparison. A 5052 alloy on that date showed 519 $\mu\text{g B/dm}^2$ at 13 months and a 5050 alloy specimen also at 13 months showed 630 $\mu\text{g B/dm}^2$. No clear correlation as to time in test or alloy type can be made.

One generalization that appears justifiable is that the topmost coupons, those in positions 1 and 2 on the shim web coupon holder, showed the highest values noted above. This may tie in with the fact that temperatures of more than 200 °C have been observed in the gas blanket above the moderator fluid which is itself nominally at 140 °F. Heat from the primary coolant (500 °F) in-process tubes in the gas blanket region and heat from radiation absorption are thought to be the cause of the high gas blanket temperature.

Corrosion penetration values for seven specimens in the gas blanket were low, varying from 0.0065 mils for a 6061 specimen at 6 months in test to 0.079 mils for a 6063 coupon at 30 months. These values correspond to rates of 0.001 mils/month and 0.0026 mils/month. Penetration

measures on coupons from the moderator fluid region are in progress.

Gamma activity was measured on two coupons (both 6061 alloy) in the moderator fluid. Values lay between 3.8×10^6 d/m/cm² for ¹¹⁰Ag and 1.7×10^5 d/m/cm² for ⁴⁶Sc. Silver-110, ⁶⁰Co, ⁴⁶Sc, ⁹⁵Zr, ¹³⁷Cs, ⁵¹Cr, and ⁵⁹Fe were found. All are considered to be due to activation of corrosion products and other trace impurities in the moderator, and were not due to debris from the fuel element rupture of 9-29-65. Cobalt-58 is notably absent, whereas it is important in the primary coolant.

Coupons in the moderator fluid had even gray oxide coats with a few very small pits, while those in the moderator gas blanket were mottled, with colors varying from between bright metallic, white, and black. This appears to be a consequence of the introduction of new, bright coupons into the gas blanket, which do not achieve even coats in comparatively short exposure times. No mottled surfaces on the 6063 aluminum shroud tubes were observed with the borescope on 6-8-67.

Rough radiation readings from single coupons were taken for controlling personnel exposure. The maximum observed was 240 mR, 2300 mrad (uncorrected), and 10,000 d/m alpha (one surface) for a 6061 alloy coupon located near the center of the flux after 15 months in test. The coupons used for gamma scan noted above read 95 mR, 550 mrad in one case 150 mR, 900 mrad in the other. Averages for 22 coupons were 72 mR, 550 mrad.

SHUTDOWN RADIATION MEASUREMENTS

L. D. Perrigo

Systematic shutdown radiation measurements are made on the PRTR to determine radioactivity buildup rates. Current buildup rates in the PRTR primary system operating with a "neutral" pH coolant are being compared to those encountered during earlier reactor operation with pH 10 coolant.

Figure 5.1 shows radioactivity buildup data for the PRTR primary system typical locations from initial startup in 1961 to the present. Operation prior to 614 equivalent full power days (EFPD) was with pH 10 coolant; subsequent operation has been with "neutral" pH coolant. Of particular interest at present are the two periods of operation following decontamination at 121 and 614 EFPD. Although it is too early to establish long-term buildup rates for operation with "neutral" pH coolant, examination of radiation data from the two post decontamination periods show substantially the same levels of activity at comparable times. The similarity in levels after about 100 EFPD of post decontamination operation was found for all nine positions routinely monitored on the PRTR primary system. As reported for the last quarter operation, "neutral" pH coolant does not appear to increase the buildup of radioactivity in the PRTR. Recent data have been normalized to 70 MW even though lower operating power levels are in use, so that current data may be compared with earlier measurements on the PRTR.

FIGURE 5.1. Primary Pump Number 2
 Volute Radiation Levels

OPERATING EXPERIENCE

J. R. Fishbaugher

Pertinent data for the quarter are as follows:

1. Production
 - a. Reactor. 2074 MWd
 - b. 55 HPD Fuel Elements . 1550 MWd
2. Hours Critical. 935.6 hr
3. Critical Efficiency 41.9%
4. Total Experimental
Time Efficiency. 54.5%
5. D₂O Losses
 - a. Indicated Stack Losses
(11-17-67 - 2-29-68) . .2185 lb
 - b. Physical Inventory . . (no inventory taken at end of period)
6. Helium Loss 278,462 scf
7. Nitrogen Loss 131,000 scf

The maximum power level of the Plutonium Recycle Test Reactor was 63 MW during December, January, and February. January experimental efficiency was 72% for a record month since beginning the Batch Core Experiment. In February, critical tests for the second burnup step of the batch core experiment were conducted. All fuel elements were discharged from the reactor, and extensive preparations were completed for a determination of the leak rate from the containment vessel. The first check of the leak rate was greater than the allowable 0.678% per day at 10 psig. Leak repair and retesting were in progress at the end of the quarter.

Total production since the start of the Batch Core Experiment is 7300 MWd or 36.5% of the batch core goal exposure. The 55-element core has accumulated 5870 MWd. Operating

performance for the PRTR is shown in Figure 5.2.

The PRTR operation was interrupted by 13 shutdowns, of which 7 were intentional and 6 were scrams. A brief description of the major outages occurring during the report period follows:

December 10 - shut down for 204.3 hr on schedule to perform critical tests of an 8-rod fuel element test in the Fuel Element Rupture Test Facility. Preventive maintenance and repairs were also completed.

December 27 - shut down for 50.2 hr to repair D₂O leaks.

January 7 - shut down for 158.5 hr to complete modification work and testing which was required to begin irradiation of an 8-rod fuel element test in the FERTF.

January 28 - shut down for 778.3 hr through the report period to conduct a leak rate test of the containment vessel and critical tests after the second interim batch core exposure.

Other operating and equipment experience is as follows:

The fuel element rupture test facility continued to be restricted to a maximum coolant temperature of 150 °F. Excellent control of dissolved oxygen in the coolant was obtained by converting the RLT-1 cover gas from helium to nitrogen, use of an ammonia-based ion exchanger to replace a lithium-based ion exchanger, and ammonia addition for pH control.

All modification work that was required to begin the irradiation of the 8-rod fuel element in the FERTF was completed. This included coolant flow instrumentation, duplicate emergency coolant valves, duplicate blow-

FIGURE 5.2. PRTR Operating Experience

down values, differential pressure tap relocation, temperature measuring instrumentation and RTD's, basin equipment to examine FERTF fuel elements, and fueling vehicle discharge hook backup water supply.

Critical tests were conducted for the 8-rod fuel element to determine the power match between the FERTF (PC-1550) and the adjacent 19-rod HPD fuel elements in ring one. Results showed that the 8-rod element must contain the stainless steel sleeve, insert, and absorber rod to match the 18 kW/ft power generation of adjacent fuel elements. A satisfactory test

of an 8-rod fuel element was conducted at a specific rod power of 18 kW/ft.

Excessive helium losses resulted from a 4 in. long crack in the expansion bellows below the D dump valve. A similar leak was repaired in September 1966.

A second fog spray control valve was installed in parallel with the existing system. A second safety circuit has been installed to duplicate the original safety circuit. Additional ventilation containment valves and controls were 96% complete. These safety systems modifications provide increased redundancy and protection against "live shorts."

A leak developed in the gas loop shroud system whereby the water backed up from the pressurized shield coolant drain (PSCD) through the seal on gas blowers 4 and 5 into the shroud system. The leak was stopped by removing and capping the PSCD line from the gas blowers.

Ninety fuel elements were examined. Damaged, loose, or missing bands and/or hanger pins were found for 13 fuel elements. One missing band, one loose band, and 15 damaged bands were replaced. All fuel elements were approved for continued irradiation.

Thirty-eight process tubes were examined visually for a total of 78 since April 1, 1967. No significant changes were observed since the last examination.

Ninety-five irradiations were obtained in the pneumatic accessory in support of programs requiring activation analyses.

Corrosion surveillance continued throughout the quarter for the helium systems and all PRTR cooling systems. Results from coupon samples removed from the primary system and the moderator calandria this quarter showed corrosion rates of less than 0.005 mils/month for stainless steel, Zircaloy, nickel and/or aluminum in these systems. Electrical resistance probes continue to show uniform corrosion rates of less than 0.005 mils/month for the moderator system, and also for the reflector, top and bottom shield, and core blanket helium systems. Similar probes show moderate corrosion rates (0.10 to 0.19 mils/month) in the biological and thermal shield systems.

Further descriptions of the corrosion status of the various PRTR systems are contained in a report prepared this quarter.*

Visual surveillance of crud deposition, involving an average of 12 fuel element inspections per month, indicated a gradual increase in crud accumulations during the quarter. Thin smooth crud layers are present on central zone fuel surfaces, with somewhat thicker deposits present on the ends and in fringe-zone regions. The crud is relatively loose and easily removed.

Filters of 5 and 40 μ pore size, located in reactor process channels, continued to provide crud surveillance and control throughout the quarter. These filters are replaced and cleaned as they become loaded with fine crud particles.

No increases in radiation levels were observed at the primary and reflector systems' monitoring points during the quarter.

Fire Protection Improvements

Physical work has begun on project AEC-193 which will provide sprinklers throughout the 309 Building except for the containment vessel and in certain areas which cannot be sprinkled because of nuclear safety considerations.

* R. B. Richman and C. W. Pollock, Corrosion and Wear Surveillance in the Plutonium Recycle Test Reactor, BNWL-726, Pacific Northwest Laboratory, Richland, Washington. February 28, 1968

Waste Handling Project

Project BCP-013 was 62% complete on March 1, 1968. This project will provide a number of services which will enable faster recovery from a reactor incident of moderate severity. Included are: 1) additional charcoal filters, 2) a submersible sump pump for C-cell, 3) high range

liquid level indicators for sumps, 4) outfall monitor for manhole No. 3, 5) a vessel air sampler, and 6) a pumpout connection for manhole No. 2.

PROCESS TECHNOLOGY - J. R. Fishbaugher

The irradiation status of the fuel elements in the PRTR on February 29 is as follows:

<u>Type of Fuel Element</u>	<u>No.</u>	<u>Peak FE Exposure, MWd</u>	<u>Accumulated Exposure, MWd</u>
Batch Core Fuel Elements (New) UO_2 -2 wt% PuO_2	71	150 (4500 MWd/T)	6840
Batch Core Fuel Elements (Old) UO_2 -2 wt% PuO_2	3	226 (6770 MWd/T)	492
UO_2 -1 wt% PuO_2	3	447 (8940 MWd/T)	1133
UO_2 -0.5 wt% PuO_2	3	570 (11,400 MWd/T)	1529
UO_2	1	493 (9860 MWd/T)	493
ThO_2 -5.0 wt% PuO_2	1	8	8

Interim Critical Tests

The interim critical tests (reactivity worth of a ring three HPD element; moderator boron worth and cold, clean critical measurement; reactivity oscillator measurement; and reactivity worth of poison rods in moderator) for the second burnup step of the Batch Core Experiment were completed in February at a batch core exposure of 7300 MWd. The results showed that the reactivity of the core decreased from an initial excess reactivity of 263 mk to a value of 191 mk. The life of the batch core was extrapolated to be 23,000 MWd, confirming previous forecasts.

Batch Core Power Tests

A measurement of the temperature coefficients of reactivity was made under equilibrium conditions to indicate that the positive moderator temperature coefficient of reactivity had decreased about 25% from its initial value due to the decrease in boron concentration.

PRTR Test 72-1 - Irradiation of Zircaloy Corrosion Coupons

An assembly of Zircaloy-2 and 4 corrosion samples was installed in the place of one fuel rod on fuel element No. 6512 in September 1967. Irradiation of this element continued

throughout the quarter. Inspection in early February indicated that the fuel element, samples, and spacers were in good condition.

PRTR Test 99-2

A new assembly was installed in process channel 2049 to determine the in-reactor creep rate of Zircaloy-2. The irradiated assembly (T-99) was removed and shipped to the Radiometallurgy Laboratory after attaining an exposure of about 7.4×10^{20} nvt (greater than 1 meV).

PRTR Test 124 and 131 - Fission Gas Pressure Measurement in HPD Fuel Elements During Irradiation

In December, fuel element No. 6521 (T-131), a plutonium-enriched, thorium oxide fuel element containing two fission gas pressure-measuring fuel rods was charged into process channel 1342. Twenty temperature and 15 pressure readings were obtained for rod No. 3; two temperature readings were obtained for rod No. 6. The maximum temperature and pressure observed were 374 °F and 15 psig in rod No. 3.

The two UO_2 -2 wt% PuO_2 fission gas pressure measuring fuel elements continue to perform satisfactorily (PRTR T-124). One hundred and seventy-one (171) temperature readings and 177 pressure measurements were made on each of the four test rods in FE-6519 and 6520. The maximum pressure observed was 12.5 psig and a maximum temperature of 500 °F was observed

for FE-6519; 106 psig and a maximum temperature of 410 °F was observed for FE-6520.

PRTR Test 127-1 - Irradiation of Four 15 in. Long Zircaloy-2, 2.5 wt% Niobium Burst Test Specimens

An assembly of four burst test samples of Zircaloy-2, 2.5 wt% niobium was irradiated for a limited time in a reactor process channel. The assembly was discharged for evaluation of repeated low flow conditions.

PRTR Test 136 - Determination of Maximum Allowable Fuel Rod Specific Power

The evaluation of the test data was completed and a summary letter issued. Revised tube power limits were placed in effect following approval of the controlling Operating Standards and Process Specifications. Specific rod power limits in the primary system are 19.3 and 21.2 kW/ft for vipac and pellet rods respectively. The factors of conservatism are 1.06 for instrument error and 1.01 for accuracy of axial flux peaking measurement. Preparation of the final report was begun.

PRTR Test 140 - Calibration of Process Tube Outlet RTD's

The four primary system inlet RTD's were removed, modified, calibrated, and reinstalled during the quarter. The difference in the inlet temperature values as indicated by these four RTD's was reduced to less than

1 °F. This completes the inlet RTD upgrading program.

PRTR Test 142 - Irradiation of Pel-
leted Fuel Elements in the Main Core

One pelleted fuel element was irradiated in the main core throughout

the quarter. The second pellet element was moved from the FERTF to the main core in January. The maximum accumulated exposure of 138 MWd (4140 MWd/T) was obtained for the second pelleted element (FE-6700).

6.0 PUBLICATIONS AND PRESENTATIONS

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