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PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

April 1970

AEC RESEARCH &
DEVELOPMENT REPORT

BNWL-1358

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BNWL-1358
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Reactor Technology

PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

By

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

April 1970

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PACIFIC NORTHWEST LABORATORIES
RICHLAND, WASHINGTON 99352

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

PLUTONIUM UTILIZATION PROGRAM
TECHNICAL ACTIVITIES QUARTERLY REPORT
DECEMBER 1969, JANUARY, FEBRUARY 1970

FOREWORD

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

BNWL-1358

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March, April, May 1968	BNWL-828
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1.0 SUMMARY

REACTOR NEUTRONICS

Critical experiments were completed in the PRCF in a H₂O-moderated lattice (1.06 in. square pitch) of UO₂-2 wt% PuO₂ fuel rods.

A transverse buckling option and a "white" albedo boundary condition option have been added to the THERMOS/BATTELLE code. An error was corrected in the anisotropy correction option and the cell smear routine has been modified to make it more useful.

An option has been added to the slowing down spectrum code HRG to permit the calculation and punching of 2-table sets of S_N parameters based on an improved, "extended," transport approximation.

FUELS TECHNOLOGY

Dose rate measurements were made on BWR type fuel assemblies containing mixed-oxide rods at United Nuclear Corporation and General Electric Company. Although the dose rates were lower than expected, the data indicate that some shielding may be required in the final assembly stage. Measurements were also made at PNL on simulated arrays containing mixed-oxide fuel rods enriched with high exposure plutonium.

Dissolubility studies are being conducted on UO₂-PuO₂ fuel samples to investigate the effects of different fabrication parameters and plutonium compositions.

An instrumented experiment designed to monitor internal gas pressure during irradiation to peak burnups of 45,000 MWd/MTM

is being fabricated. Design of an experiment to measure centerline fuel temperatures in different mixed-oxide fuel rods during irradiation is being developed.

The defect test assembly containing 12 fueled Zircaloy-clad $\text{UO}_2\text{-PuO}_2$ rods is being irradiated in the ETR M-3 loop for the first planned cycle of irradiation without an intentionally defected rod.

Postirradiation measurements indicate that dimensional changes are more pronounced in cold-pressed-sintered rods than in rods containing hot-pressed pellet or vibrationally compacted fuel.

The defect in a cold-pressed-sintered $\text{UO}_2\text{-PuO}_2$ rod is characteristic of a Zircaloy cladding failure originating from within the rod caused by contaminants.

There is excellent consistency in the structural characteristics of fuel specimens extracted from the instrumented vibrationally compacted PRTR rods.

Comparison of fuel structures formed in vibrationally compacted rods irradiated under the same conditions to low burnup indicates that the extent of the equiaxed grain growth boundary is the most reliable indicator of fuel operating conditions.

REACTOR SAFETY

A joint PNL/GE/INC program was initiated in SPERT to investigate the possible effects on the transient behavior of adding PuO_2 enrichment to UO_2 fuels. Also, confirming experiments to compare the transient behavior of pelleted and packed particle fuels are being conducted in TREAT.

The draft copy of BNWL-SA-3143, Fracture Behavior of Flawed Zircaloy-2 Pressure Tubes, has been completed and submitted to NUCLEAR ENGINEERING AND DESIGN for publication.

This paper shows the fracture testing of Zircaloy-2 pressure tubes for simulating the effect of a sharp crack should involve the use of fatigue cracks rather than machined notches or slots, since the latter do not represent as severe a stress concentration as an actual crack. It has also been shown that an empirical failure criteria may be used to treat the failure of pressure tubes containing axial through-wall cracks.

PLUTONIUM RECYCLE TEST REACTOR

Surveillance activities have disclosed no problem areas during the first three months of deactivation. Fuels not being retained for possible future reactor operation and R&D are being shipped to ARCHO for processing.

2.0 REACTOR NEUTRONICS

CRITICAL EXPERIMENTS IN UO_2 -2 wt% PuO_2 FUELED- H_2O MODERATED ASSEMBLIES

W. P. Stinson

Critical experiments were conducted in the Plutonium Recycle Critical Facility (PRCF) to obtain data for verifying the accuracy of neutronic design methods. The fuel rods contained UO_2 -2 wt% PuO_2 with 8% ^{240}Pu ⁽¹⁾ and the moderator was H_2O . The core consisted of the fuel (1/2 in. diameter rods) in a 1.06 in. square lattice pitch loaded to form a cylinder 36 in. in height and a radius equivalent to 150 fuel rods when critical. The measurements conducted on this core were:

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

Approach-to-critical data were taken as a check on the techniques used in similar type measurements made in the Critical Approach Facility (CAF). The critical loading inferred from the subcritical approach was 150.1 ± 0.03 rods which agrees well with the actual critical loading of 150.1 ± 0.05 rods.

The kinetics measurements were made by recording and analyzing the reactor noise. The preliminary value of β/ℓ obtained is $76 \pm 3 \text{ sec}^{-1}$. The flux distribution was measured by scanning the fuel rods for fission product decay gamma rays. Empty thin wall aluminum tubes were placed in interstitial positions in the central part of the core to obtain data from which the void coefficient can be determined. Analysis of the data is continuing to obtain final values.

Reference

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

CODE MODIFICATIONS

C. L. Bennett

The following modifications to THERMOS/BATTELLE have been completed.

- 1) A transverse buckling option of the simple DB^2 form has been added. The radial, axial, or planer buckling is calculated from the input dimensions which may or may not include the extrapolation distances. The extrapolation distances would normally be calculated internally in this option.
- 2) A "white" albedo boundary condition option has been added to the cylindrical-geometry ray-tracing method. The boundary albedo can be specified for either the whole thermal range or each speed.
- 3) An error in the anisotropic scattering correction option (neglecting isotopes used in the cell not on the library) has been corrected.
- 4) The cell smear routine has been modified to correctly produce a complete spatially-collapsed-cell macroscopic-cross-section set which can be punched out on cards in a format compatable with its use in a succeeding THERMOS case as an added material used in the cell but not provided in the library.

CALCULATION OF 2-TABLE SETS OF S_N PARAMETERS IN HRG

J. L. Carter

For some time, the epithermal spectrum code HRG has had available an option of punching a 2-table set of parameters

for use in S_N codes such as DTF-IV. In this option, the entry for group g in the σ_g position of the first table is $\sigma_{t,g}(0)$, the flux weighted total cross section. Bell et al.⁽¹⁾ show that generally it is better to use the extended transport approximation, which gives

$$\sigma_g = \sigma_{t,g}(2) - \sigma_{2g}(2) \quad (1)$$

for a 2-table set. Here the subscript 2 refers to the second Legendre moment of the scattering cross section in the laboratory system and the 2 in parentheses indicates the cross sections should be weighted over the energies of group g by the second Legendre moment of the angular flux. An option has now been added to HRG to permit punching 2-table sets of S_N parameters based on this extended transport approximation.

Because evaluation of Equation (1) requires knowledge of the second Legendre moments of both the angular flux and the scattering cross section (neither of which is directly available in HRG), some auxiliary calculations must be done. Bell et al.⁽¹⁾ show that, for the unknown flux moment, an adequate approximation is

$$\phi_2(E) \propto \phi_1(E)/\Sigma_{tr}(E) \quad (2)$$

where $\phi_1(E)$ is the first Legendre moment of the flux, i.e., the current, and $\Sigma_{tr}(E)$ is the macroscopic transport cross section. These latter quantities are available in HRG for each fine group, so that the fine group analog of Equation (2) can readily be found.

Finding the second Legendre moment of the scattering cross section is more difficult, but sufficient information is available so that a reasonable approximation can be made if we can assume that the only anisotropy is that of elastic scattering and that this is linearly anisotropic in the center

of mass system. In this case, the microscopic cross section in energy and solid angle for a particular isotope is, in usual notation,

$$\sigma_S(E, \mu_C) = \frac{\sigma_S(E)}{4\pi} [(1 + \omega_1^C(E) \mu_C)]. \quad (3)$$

In the laboratory system, the cross section for this isotope is

$$\sigma_S(E, \mu_L) = \frac{\sigma_S(E)}{4\pi} \sum_{\ell=0}^{\infty} \omega_{\ell}^L(E) P_{\ell}(\mu_L). \quad (4)$$

The transformation matrix relating these two sets of ω 's is well-known;⁽²⁾ we only need values for $\ell=0,1,2$. For $\gamma = 1/A \ll 1$, these are

$$\omega_0^L(E) = 1 \quad (5a)$$

$$\omega_1^L(E) = 2\gamma + (1 - \frac{3}{5} \gamma^2) \omega_1^C(E) \quad (5b)$$

$$\omega_2^L(E) = \gamma^2 [1 + 0(\gamma^2)] + 2\gamma [1 + 0(\gamma^2)] \omega_1^C(E) \quad (5c)$$

where only terms to order γ^2 are given explicitly for $\omega_2^L(E)$. The angular moments we seek are not the ω_{ℓ}^L themselves, but instead the coefficients of the expansion

$$\sigma_S(E, \mu_2^L) = \sum_{\ell=0}^{\infty} \frac{2\ell + 1}{4\pi} \sigma_{\ell}(E) P_{\ell}(\mu_2^L). \quad (6)$$

From Equation (4) it follows that

$$\sigma_{\ell}(E) = \frac{\omega_{\ell}^L(E) \sigma_S(E)}{2\ell + 1}. \quad (7)$$

Now $\omega_1^C(E)$ can be eliminated between Equations (5b) and (5c) and (7) can be used to express $\sigma_2(E)$ in terms of $\sigma_1(E)$ and $\sigma_0(E)$.

In fine group form, the result can be written

$$\begin{aligned} \sigma_{2,n} = & \frac{6}{5} \frac{\gamma[1 + \frac{0(\gamma^2)}{3\gamma^2}]}{1 - \frac{3}{5}\gamma^2} (\sigma_{1,n} - \frac{2}{3}\gamma\sigma_{0,n}) \\ & + \frac{1}{5}\gamma^2 [1 + 0(\gamma^2)]\sigma_{0,n} \quad \gamma \ll 1. \end{aligned} \quad (8)$$

For an individual isotope, Equation (8) gives the second Legendre moment we seek.

Many HRG nuclides are individual isotopes and $\sigma_{0,n}$ and $\sigma_{1,n}$ are available for them, so that $\sigma_{2,n}$ can be found if γ is known. We can readily find γ from the knowledge that the scattering is isotropic in the center of mass system at sufficiently low energies. Thus, we can choose a low energy fine group m for which scattering is isotropic and find γ from

$$\frac{\sigma_{1,m}}{\sigma_{0,m}} = \frac{2}{3} \gamma. \quad (9)$$

For these nuclides, σ_g in Equation (1) can be found by averaging $\sigma_{t,n} - \sigma_{2,n}$ over all fine groups n in g , where $\sigma_{2,n}$ is found from Equation (8) and (9), and where, from (2), the weighting function is $J_n/\Sigma_{tr,n}$.

The option which has been added assumes that this same procedure can also be used for the remaining HRG nuclides and for the macroscopic cross section. This extension of the application of Equation (8) and (9) needs justification. What is obtained from (9) is an average value $\bar{\gamma}$, where

$$\bar{\gamma} = \frac{\sum_i \gamma_i N_i \sigma_{0,m}^i}{\sum_i N_i \sigma_{0,m}^i} \quad (10)$$

and the index i refers to the component isotopes.

It is not immediately clear that $\bar{\gamma}$ is appropriate for use in Equation (8). For one thing, other averages such as $\overline{\gamma^2}$ are needed in (8), and $\overline{\gamma^2} = \bar{\gamma}^2$ may be a poor approximation even for group m. In addition, if the ratios of the component σ_0 's vary from group to group, $\bar{\gamma}_m = \bar{\gamma}_n$ may also be a poor approximation. If the γ_i are nearly equal, however, as in those HRG nuclides which represent elements, it is clear from Equation (10) and its analogs for higher powers of γ_i that $\overline{\gamma^P} \approx \bar{\gamma}^P$ and $\bar{\gamma}$ can be used in (8) with little error. Where the γ_i are widely different (as in the HRG use of H_2O and D_2O as nuclides and as is usually the case for the components of macroscopic cross sections for a cell), the accuracy of the approximation can best be determined by testing actual data. Because $\gamma = \bar{\gamma} \ll 1$ is not true for these cases, another problem arises in using (8), i.e., the leading terms in γ do not reduce to the correct limit as $\gamma \rightarrow 1$. This deficiency can be largely overcome by adding terms in γ^2 in (8), with adjustable coefficients chosen to give the correct values as $\gamma \rightarrow 1$. When this is done, (8) is replaced by

$$\begin{aligned} \sigma_{2,n} = & \frac{6}{5} \gamma \left(\frac{1 - \frac{3}{8} \gamma^2}{1 - \frac{3}{5} \gamma^2} \right) \left(\sigma_{1,n} - \frac{2}{3} \gamma \sigma_{0,n} \right) \\ & + \frac{1}{5} \gamma^2 \left(1 + \frac{1}{4} \gamma^2 \right) \sigma_{0,n}. \end{aligned} \quad (11)$$

A test case has been run to compare the macroscopic value, the σ_g of (1), with the sum of the microscopic values of the individual components. In the test case Equations (9) and (11) were used to find each fine group $\sigma_{2,n}$ and the weighting function of $J_n/\Sigma_{tr,n}$ was used in averaging over the fine groups in g. In addition, a 0.5 in. fuel rod containing a mixture of plutonium and uranium oxide clad with zirconium in a geometry with a fuel/water ratio of about 1 was assumed for

the calculation. Three broad groups were used, with lower energy limits of 1.17×10^4 , 2.38, and 0.683 eV. The percent differences between the macroscopic and sum of microscopic values of Equation (1) were -1.8, 9.0, and 3.4 for the three groups. The H_2O was treated as a nuclide in this case. A separate estimate of the error in treating H_2O as a nuclide rather than as $2H + O$ is about 3% in the lower energy fine groups and 6% in the higher energy fine groups.

Although these results indicate that an uncertainty of the order of 10% is introduced by extending Equations (9) and (11) to macroscopic cross sections, the accuracy is usually better than that of the basic assumption of linearly anisotropic scattering of the center-of-mass system. Therefore a more elaborate treatment of the available data is not warranted at this time.

References

1. G. I. Bell, G. E. Hansen, and H. A. Sandmeier. "Multi-table Treatments of Anisotropic Scattering in S_N Multi-group Transport Calculations," Nuc. Sci. Eng., vol. 28, pp. 376-383. 1967.
2. Reactor Physics Constants, 2nd Ed., ANL-5800, pp. 50-51. Argonne National Laboratory, Argonne, Illinois, July 1963.

3.0 FUELS TECHNOLOGY

HIGH EXPOSURE PLUTONIUM STUDIES

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

Dose rate measurements were made on several arrays of plutonium enriched fuel rods at United Nuclear Corporation, General Electric Company, and PNL. The measurements were made on fuel rod arrays similar to BWR assemblies.

At United Nuclear Corporation, dose rate measurements were made on arrays of the type which may be used as transitional fuel in a change from a UO_2 to a $\text{UO}_2\text{-PuO}_2$ loading for a Dresden type BWR. The fuel rods were 0.55 in. OD and 48 in. long with 0.035 in. thick Zircaloy-2 cladding. High exposure plutonium recovered from Dresden fuel containing about 18 wt% ^{240}Pu was used in these pins at an enriched level of 2.3 wt% PuO_2 in natural UO_2 . Up to four, 36 pin arrays were measured at one time. The gamma dose rate data are summarized in Table 3.1.

TABLE 3.1. Measured Gamma Dose Rates from UNC Fuel

<u>Array</u>		<u>Dose Rate, mrad/hr</u>		
<u>No. of PuO_2 Pins</u>	<u>No. of UO_2 Pins</u>	<u>3/4 in.</u>	<u>6 in.</u>	<u>24 in.</u>
9	27	4.2	1.03	background
36	108	4.4		0.11
21	15	8.4	1.3	0.34
84	60	12.6		0.7
112	3+	16.6		2.04

In each case, except the last, the remaining pins required to complete each 36 pin assembly or sets of assemblies were enriched UO_2 . In the last case listed, only available $\text{UO}_2\text{-PuO}_2$ pins were assembled.

Neutron yields were measured using a Long Counter on the same geometric arrays listed in Table 3.1. In no case was the neutron dose rate more than about 1 mrem/hr at any distance beyond 6 in. from the fuel surface. This neutron dose rate was considerably less than expected.

Gamma dose rates were measured at General Electric-Vallecitos on a BWR fuel assembly containing 9/16 in. diameter fuel pins in a 9 × 9 pin array. The mixed-oxide rods contained different PuO₂ enrichments. The measured gamma dose rate from the fuel assembly was 12 mrad/hr at 3/4 in., 3.2 mrad/hr at 6 in., and 1.9 mrad/hr at 24 in.

The maximum neutron dose rate from the GE fuel assembly was about 2.2 mrem/hr at 24 in. A "Snoopy" neutron dose rate measurement made with the instrument in contact with the fuel cladding (estimated dose at five inches) was 7 mrem/hr for the fuel assembly. Again, the measured neutron dose rates were encouragingly low.

Measurements were also made at PNL on 9/16 in. diameter fuel pins, 36 in. long with 0.030 in. thick Zircaloy-2 cladding. The isotopic composition of the plutonium used in the UO₂-4 wt% PuO₂ fuel was 0.265 wt% ²³⁸Pu, 75.2 wt% ²³⁹Pu, 18.2 wt% ²⁴⁰Pu, 5.15 wt% ²⁴¹Pu, and 1.2 wt% ²⁴²Pu. The plutonium was over two years old which resulted in an appreciable buildup of ²⁴¹Am. Measurements were made on 6 × 6, 7 × 7, and 9 × 15 arrays. Measurements were not made on a 15 × 15 array because of criticality restrictions. The results of the measurements are summarized in Table 3.2.

Neutron dose rates at a distance of one foot from the assemblies measured 3.5 mrem/hr for the 7 × 7 array and 11 mrem/hr for the 9 × 15 array. While these radiation levels are not considered to be high, anticipated increased neutron dose rates from full-length commercial fuels would require shielding for the adequate radiation protection of personnel.

TABLE 3.2. Gamma Dose Rates from Simulated BWR Fuel Arrays

<u>Array</u>	<u>Plutonium Weight, g</u>	<u>Distance, in.</u>	<u>TLD Dose Rate, mrad/hr</u>	<u>CP Dose Rate, mrad/hr</u>
6 × 6	1369	surface	24.5	---
		2-1/2	11.6	10
		6	5.4	5
		12	2.4	2.5
7 × 7	1863	surface	25.4	---
		2-1/2	13.9	12.5
		6	5.3	6
		12	3.0	3
9 × 15	5132	surface	25.4	---
		2-1/2	19	22
		6	10.6	13
		12	5.9	6

PLUTONIUM FUEL DISSOLUTION STUDIES

R. E. Lerch

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

Dissolubility studies have commenced on unirradiated UO_2 - PuO_2 fuel samples. The initial study is being performed on 0.5 in. OD dished and undished cold-pressed-sintered pellets. Fabrication parameters being evaluated in the initial study include four different pressing pressures, sintering time, and sintering temperature. All pellets were made

from the same sources of PuO_2 (oxalate process, calcined at 900 °C) and UO_2 (United Nuclear, ammonium diuranate process). The pellets contain 4 wt% PuO_2 and all were sintered in Ar-8% H_2 . Theoretical densities range from 91.8 to 93.6% while oxygen-to-metal ratios are, in all cases, about 1.984. The effect of other fabrication parameters on dissolution properties of UO_2 - PuO_2 fuels will be evaluated as samples become available.

A second study is being performed to evaluate the effect of PuO_2 content on dissolution properties. Cold-pressed-sintered pellets containing from 1 to 4 wt% PuO_2 have been obtained for the study.

Fuel samples from irradiated PRTR fuel rods will also be used for dissolution studies. Initial dissolution studies on the irradiated fuel will emphasize cold-pressed-sintered specimens with secondary emphasis on vibrationally compacted fuels. The effects of power generation and burnup on dissolution will be studied.

INSTRUMENTED FUEL TESTS

T. B. Burley and M. D. Freshley

Instrumented experiments to monitor internal gas pressure buildup and fuel temperatures in mixed-oxide fuel rods during irradiation will be conducted in the ATR Water Loop and ETR M-3 Loop, respectively.

Internal Pressure Monitoring Experiment

An experiment in which internal pressure will be monitored in mixed-oxide fuel rods during irradiation at typical PWR operating conditions to peak burnups of approximately 45,000 MWd/MTM is being fabricated. The instrumented assembly will consist of five rods, each of which will be instrumented with a null-balance pressure transducer. Different pellet fuel

designs are being utilized in four of the rods and packed-particle fuel will be tested in the fifth rod. A composite hafnium-stainless steel neutron absorber basket assembly will enshroud the five-rod element. The calculated power generation in the nominally UO_2 -2 wt% PuO_2 fuel rods is in the range of 15 to 18 kW/ft.

Fuel Temperature Monitoring Experiment

Design of an ETR experiment in which fuel centerline temperatures will be measured in rods containing different mixed-oxide fuel types and pellet designs is being developed. The experiment includes twelve fueled rods with six rods instrumented with thermocouples in the fuel. Because of the flux gradients in the M-3 Loop facility, the thermocouples will be positioned at different axial locations within the rods to permit a more direct comparison of maximum fuel temperatures at comparable power generations. The calculated maximum fuel temperatures at the planned thermocouple locations for the different fuel types as a function of fractional reactor power is shown in Figure 3.1. Fuel temperatures will be monitored during a programmed power increase, during subsequent steady-state operation, and during a programmed power decrease to study and compare the thermal behavior and in-reactor grain growth kinetics of the different fuels.

FIGURE 3.1. Calculated Maximum Fuel Temperatures at the Planned Thermocouple Positions in the Different Fuel Types

DEFECT TESTING

M. D. Freshley and T. B. Burley

Irradiation tests of defected and nondefected mixed-oxide fuels will be performed under high performance conditions in the ETR M-3 Loop facility. The defect testing program includes

an investigation of three important parameters consisting of (1) fuel form (small and large gap solid pellet, dished pellet, and cored pellet), (2) power generation (18.5 to 23 kW/ft), and (3) burnup. In addition to evaluating fuel performance, comparative activity release data will be obtained for the different fuel designs while operating under different conditions.

The defect test assembly containing twelve fueled Zircaloy-clad UO_2 - PuO_2 rods is being irradiated in the ETR M-3 Loop for the planned first cycle of irradiation without an intentionally defected rod. The element also contains six Zircaloy dummy rods which were subjected to different surface treatments to study the effect of these treatments on the in-reactor corrosion behavior of Zircaloy-2 in a coolant system contaminated with fissionable material. The actual measured power generation of the assembly is 390 kW versus a predicted power of 410 kW. A summary of the operating conditions for the tests is presented in Table 3.3.

TABLE 3.3. Operating Conditions for the Defect Test Assembly in the ETR M-3 Loop

Peak Linear Rod Power Generation	~23 kW/ft
Average Linear Rod Power Generation	~14 kW/ft
Peak Surface Heat Flux	546,000 Btu/hr-ft ²
Average Surface Heat Flux	329,000 Btu/hr-ft ²
Calculated Total Power Generation	410 kW
Heat Transfer Film Coefficient	5169 Btu/hr-ft ² -°F
Maximum Cladding Surface Temperature	603 °F
Maximum Cladding Temperature	770 °F
Maximum Fuel Surface Temperature	1165 °F
Maximum Fuel Temperature	4700 °F
Loop Flow Rate	150 gal/min
Loop Pressure	2000 psi
Coolant pH	9.8 to 10.2
Coolant Oxygen Concentration	<0.05 ppm
Saturation Temperature of Loop	635 °F
Coolant Velocity over the Fuel	11.85 ft/sec
Inlet Water Temperature	480 °F
Outlet Water Temperature	502 °F

After the first cycle of irradiation, one of the rods in the three replaceable rod positions will be defected with an 0.015 in. diameter hole. It is planned to operate the defected rod for one cycle at a peak rod power of 18.5 kW/ft and increase the power generation to 23 kW/ft near the end of the cycle. The planned disposition of the three replaceable rods within the defect test assembly for the complete defect test series is shown in Figure 3.2. Procedures and equipment necessary for defecting rods in the ETR basin are being developed.

FUEL TYPE

- ① SOLID (0.006" DIAMETRICAL GAP)
- ② SOLID (0.010" DIAMETRICAL GAP)
- ③ DISHED 3% VOID VOLUME (0.006" DIAMETRICAL GAP)
- ④ DISHED 3% VOID VOLUME (0.010" DIAMETRICAL GAP)
- ⑤ CORED 10% VOID VOLUME (0.006" DIAMETRICAL GAP)

* CHANGE POWER AT MID-CYCLE OR BETWEEN CYCLES BY ROTATING ELEMENT 180°

** EACH TEST REPRESENTS APPROXIMATELY ONE ETR CYCLE. A DEFECTED ROD WILL NOT BE IRRADIATED DURING TEST NO. 1. A VERTICAL ARROW INDICATES REMOVAL OF THE ROD FROM THE TEST FOR POST-IRRADIATION EXAMINATION.

FIGURE 3.2. Planned Disposition of the Three Replaceable Rods in the Test Assembly for the Defect Test Series

PRTR FUEL EVALUATIONS

D. G. Carter, T. B. Burley, and M. D. Freshley

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

- High Power Density Fuel Evaluation
- Extended Burnup Fuel Evaluation
- Internal Pressure Monitoring Experiment Evaluation
- FERTF Fuel Rod Evaluation.

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

High Power Density Fuel Evaluation

Postirradiation examination of selected PRTR High Power Density Fuel rods which were irradiated as part of the Batch Core Experiment (BCE) is continuing and the current status of the examination is summarized in Table 3.4. These rods were irradiated to peak burnups of approximately 11,000 MWd/MTM at power generations greater than those currently employed in commercial power reactors, i.e., actual peak linear power generations of 20 kW/ft with maximum fuel temperatures near melting. Although the majority of the 66 High Power Density 19-rod cluster fuel elements included in the BCE were vibrationally compacted powder fuel, pellet fuel fabricated by both the hot-pressed and the cold-pressed-sintered processes was included.

TABLE 3.4. Summary of PRTR High Power Density Fuel Rod Postirradiation Examination Status

Element Number	Rod No.	Fuel Type	Estimated Peak Burnup, MWd/MTM	Gamma Scan	Profilometer Scan	Fission Gas Analysis	Burnup Analysis	Ceramography
6065	FE-74	UO ₂ -2 wt% PuO ₂ (VP)	5,000	X	Z	X	Y	Y
	FD-17	(VP)	6,500	X	Z			
	FS-07	(VP)	10,000	X	Y			
6519	FR-72	(VP)	5,700	X	Y			
	FN-86	(VP)	3,700	X	Z			
	FE-69	(VP)	2,800	X	Z			
6520	FR-78	(VP)	11,000	X	Y			
6700	A-14	(HP)	9,600	X	Y			
	A-103	(CPS)	9,600	Z	Y			
	A-102	(CPS)	9,600	X	Y			
6701	A-23	(HP)	3,800	X	Y			
	A-12	(HP)	3,800	Z	Y			

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

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

(HP) = *Hot-pressed pellet fuel rod.*

X = *Examination completed.*

Y = *Examination in progress.*

Z = *Examination not scheduled.*

Postirradiation length measurements and diameter profiles were obtained on three vibrationally compacted fuel rods, three rods containing hot-pressed pellet fuel, and two rods which contained cold-pressed-sintered pellet fuel. The Zircaloy cladding on all rods is nominally 0.565 in. OD with an 0.030 in. wall thickness. The packed-particle rods contained crushed, pneumatically impacted UO_2 - PuO_2 fuel vibrationally compacted to a smear density of nominally 86% TD. The hot-pressed and the cold-pressed-sintered pellet density was approximately 93% TD and the rods were assembled with a nominal 0.012 in. diametral gap.

Results of the fuel rod length measurements are summarized in Table 3.5. The three vibrationally compacted fuel rods exhibited length changes ranging from less than +0.01% for the rod of lowest burnup to +0.05% for one of the rods of highest burnup. The three hot-pressed pellet fuel rods with peak burnups ranging from 3800 to 9600 MWd/MTM exhibited length changes of +0.04%, which is similar to the changes which occurred in the vibrationally compacted rods. The two cold-pressed-sintered pellet fuel rods, both of which had the same estimated peak burnup, exhibited length increase of 0.18 and 0.25%.

Figures 3.3 through 3.6 are typical examples of actual diameter profile scans obtained from the different fuel types examined. Two stationary axial scans 90° apart and a helical scan were made on each rod. Preirradiation diameter profile scans were obtained on only the two cold-pressed-sintered rods. The diameter profile scan data was averaged in a manner illustrated in Figures 3.7 through 3.16. A comparison of the 0° and 90° scans provides an indication of the degree of ovality for any given rod. This is most evident on rods A-102 and A-103. The dashed line $(0^\circ+90^\circ/2)$ corresponds

TABLE 3.5. Profilometer Measurement Data on PRTR High Power Density Fuel Rods

Rod No.	Fuel Type	Estimated Peak Burnup, MWd/MTM	Pre-OD, ^(a) in.	Post-OD, ^(a) in.	% Δ OD	Maximum Ovality, mils	Pre-L, ^(b) in.	Post-L, ^(b) in.	% Δ L
FR-72	Vipac	5,700	No data	0.5611	-	1.0	67.331 ^(c)	67.337	<+0.01
FR-78	Vipac	11,000	No data	0.5625	-	1.5	67.331 ^(c)	67.350	+0.03
FS-07	Vipac	10,000	No data	0.5621	-	4.0	67.331 ^(c)	67.368	+0.05
A-12	Hot-pressed	3,800	No data	0.5639	-	1.0	67.326 ^(c)	67.354	+0.04
A-14	Hot-pressed	9,600	No data	0.5652	-	1.0	67.326 ^(c)	67.354	+0.04
A-23	Hot-pressed	3,800	No data	0.5635	-	1.0	67.326 ^(c)	67.355	+0.04
A-102	Cold-pressed-sintered	9,600	0.5632	0.5621	-0.20	5.0	67.470	67.590	+0.18
A-103	Cold-pressed-sintered	9,600	0.5641	0.5634	-0.12	3.0	67.467	67.636	+0.25

a. Average diameter value of two scans along axial position ($\frac{0^\circ+90^\circ}{2}$).

b. Rod length.

c. The preirradiated mean length, which has a calculated standard deviation of +0.002 in., was determined from a group of similar rods.

FIGURE 3.3. Preirradiation Diameter Profile Scan of Cold-Pressed-Sintered Pellet Fuel Rod A-103

FIGURE 3.4. Postirradiation Diameter Profile Scan of Cold-Pressed-Sintered Pellet Fuel Rod A-103

FIGURE 3.5. Postirradiation Diameter Profile Scan of Vibrationally Compacted Fuel Rod FS-07

FIGURE 3.6. Typical Postirradiation Diameter Profile Scan of Hot-Pressed Pellet Fuel Rod A-12

FIGURE 3.7. Summary of Preirradiation Diameter Profile Data for Cold-Pressed-Sintered Fuel Rod A-102

FIGURE 3.8. Summary of Postirradiation Diameter Profile Data for Cold-Pressed-Sintered Fuel Rod A-102

FIGURE 3.9. Summary of Preirradiation of Diameter Profile Data for Cold-Pressed-Sintered Pellet Fuel Rod A-103

3.20

FIGURE 3.10. Summary of Postirradiation Diameter Profile Data for Cold-Pressed-Sintered Fuel Rod A-103

FIGURE 3.11. Summary of Postirradiation Diameter Profile Data for Vibrationally Compacted Fuel Rod FR-78

3.22

FIGURE 3.12. Summary of Postirradiation Diameter Profile Data for Vibrationally Compacted Fuel Rod FS-07

FIGURE 3.13. Summary of Postirradiation Diameter Profile Data for Vibrationally Compacted Fuel Rod FR-72

3.24

FIGURE 3.14. Summary of Postirradiation Diameter Profile Data for Hot-Pressed Pellet Fuel Rod A-12

BNWL-1358

FIGURE 3.16. Summary of Postirradiation Diameter Profile Data for Hot-Pressed Pellet Fuel Rod A-14

well with the average value of the helical scan. The width of the helical scan also represents the degree of ovality of the rod. Preliminary observations related to the rod dimensional data are summarized as follows:

- Cladding ovality and length changes, which seem to be related, were more pronounced in the cold-pressed-sintered pellet rods than in the hot-pressed pellet and vibrationally compacted rods. Although pre- and post-irradiation mean diameter data are available on only the cold-pressed-sintered pellet rods, it is suspected that the change in mean diameter is also related to the degree of ovality.

It is reasonable that the fuel in the vibrationally compacted rods provides support for the cladding and thereby reduces the amount of ovality. Considerably less ovality occurred in the hot-pressed pellet rods than in the rods which contained cold-pressed-sintered pellets although both were assembled with a nominal 0.012 in. diametral gap. It is suspected that the hot-pressed pellet fuel, because of its high stoichiometry (~ 2.15) and reduced thermal conductivity, operated at higher temperatures than the cold-pressed-sintered fuel which resulted in closure of the fuel-clad gap and prevention of cladding collapse and distortion. The hot-pressed pellet rod data indicate that the cladding support is provided early during the irradiation because the dimensional behavior of rods with estimated peak burnups of 3800 MWd/MTM and 9600 MWd/MTM is similar. The validity of the hypothesis will be evaluated further during the postirradiation evaluation.

- Although the Zircaloy cladding is calculated to be free-standing under PRTR operating conditions, ovality is the result of creep. The creep rate may be enhanced during neutron irradiation.

- There was no evidence of general or local clad swelling in any of the rods.
- There was no evidence of ridging at pellet interfaces in either the hot-pressed or cold-pressed-sintered pellet fuel rods.

Analyses of the gas collected from the high power density fuel rods are summarized in Table 3.6. The total amount of gas collected from the rods of any one fuel type is related to the operating conditions to which the rods were subjected. More gas was collected from the vibrationally compacted and hot-pressed pellet fuel rods than from the rods containing cold-pressed-sintered pellet fuel because of the higher fuel temperatures and gas contents associated with these fuel types. Also, the hot-pressed pellet fuel rods released more gas than the vibrationally compacted rods, and the gas contained considerable amounts of CO_2 and CO which probably originated with the fabrication process. The gas release fractions for these rods have not been calculated.

Extended Burnup Fuel Evaluation

Postirradiation evaluation of selected fuel rods irradiated to extended burnups in PRTR fringe positions continued. The status of the examination of these rods is summarized in Table 3.7.

Fuel rod ZZ-12 from FE-5186, which contained cold-pressed-sintered UO_2 - PuO_2 pellet fuel, developed a defect in the cladding while the rod was being removed from the element in the PRTR basin. The defect in the Zircaloy cladding appeared to be a small crack located about 2 in. down from the top end plug. Metallographic examination of the defect area reveals a localized massively hydrided region with severe corrosive attack on the inner surface which is penetrated by a small crack through the cladding (Figure 3.17). This type of attack and the shape

TABLE 3.6. PRTR High Power Density Fuel Rod Fission Gas Analysis

Rod No.	Gas Volume, cm ³ (STP)	Molar Percent								
		Xe	Kr	CO ₂	Ar	O ₂	N ₂	CO	He	$\frac{\text{H}_2}{\text{CH}_4}$
Fe-74	43.3	29.55	1.85	0.02	0.03	0.10	3.46	<0.10	65.0	<0.01
FO-17	53.5	30.55	2.01	0.05	0.01	0.08	20.8	<0.10	46.5	<0.01
FS-07	190.0	76.44	5.27	0.03	<0.01	<0.01	2.34	<0.10	15.8	<0.01
FR-72	52.7	43.32	2.66	0.06	0.02	0.05	0.28	<0.10	53.6	<0.01
FN-86	35.6	12.12	0.78	0.04	0.03	0.28	1.13	<0.10	85.6	<0.01
FE-69	44.6	4.32	0.30	12.7	0.07	0.14	1.63	18.1	62.8	<0.01
FR-78	148.0	75.15	5.16	0.03	<0.01	<0.01	0.14	<0.10	19.5	<0.01
A-103	35.9	19.75	1.40	0.02	<0.01	<0.01	0.11	<0.10	78.7	<0.01
A-102	35.9	20.78	1.48	0.02	0.01	<0.01	0.11	<0.10	77.6	<0.01
A-23	191.0	22.68	1.65	57.8	<0.01	<0.01	3.08	<0.10	14.8	<0.01
A-12	225.0	20.62	1.54	57.3	0.01	0.06	7.37	<0.10	13.0	<0.01
A-14	249.0	66.51	4.81	5.17	<0.01	<0.01	7.26	3.82	12.4	<0.01

TABLE 3.7. Summary of the Postirradiation Examination Status for Fuel Rods Irradiated to Extended Burnup in PRTR Fringe Positions

Element Number	Rod No.	Fuel Type	Estimated Peak Burnup, MWd/MTM	Gamma Scan	Profile meter Scan	Fission Gas Analysis	Burnup Analysis	Ceramo-graphy
1037	A-1330	UO ₂	14,500	X	Z	X	O	Y
	A-1377		---	X	Z	X	O	Y
	A-1398		---	X	Z	X	O	Y
	A-1325		---	Z	Z	X	Z	Y
5118	DB-63	UO ₂ -0.5 wt% PuO ₂	18,500	X	Z	X	Z	Y
	DA-48		18,500	Z	Z	X	Z	Y
	ZZ-17		---	X	Z	X	Y	Y
	ZZ-7		---	X	Z	X	Y	Y
5186	ZZ-12		12,500	X	Z	X	Y	Y
	CN-5		12,500	X	Z	Lost	Z	Y
	CN-7		12,500	Z	Z	X	Z	Z
	CM-85		12,500	Z	Z	X	Z	Z
5224	CS-27	UO ₂ -1 wt% PuO ₂	13,500	X	Z	X	Y	Y
5226	DF-80		11,500	X	Z	X	Y	Y
6003	5C-27	UO ₂ -2 wt% PuO ₂	13,000	X	Z	X	Y	Y
6005	5F-36		8,000	X	Z	X	X	Y
6521	No. 6	ThO ₂ -5 wt% PuO ₂	Low	X	-	Lost	Z	Y

(VP) = Vibrationally compacted fuel rod

(SW) = Swage compacted fuel rod

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

(MM) = Heterogeneously enriched or incrementally loaded fuel material

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

X = Examination completed

Y = Examination in progress

Z = Examination not scheduled

O = Examination not started

FIGURE 3.17. Severely Corroded and Hydrided Zircaloy Cladding in the Defected Region of a PRTR Rod (ZZ-12) Which Contained Cold-Pressed-Sintered $\text{UO}_2\text{-PuO}_2$ Pellet Fuel (~65X)

of the hydrided region suggest Zircaloy cladding failure originating from within the rod which was possibly caused by contaminants in the fuel--such as moisture. There was no evidence of localized hydriding in the top end plug crevice region which has been shown to be particularly sensitive to this type of attack. Since the rod withstood irradiation to an estimated peak burnup of 12,500 MWd/MTM without an apparent defect, the level of internal contamination must have been low.

Analyses of the gas collected from the remaining four rods from this group are summarized in Table 3.8. The analyses are typical of the gas collected from packed particle UO_2 - PuO_2 fuels.

TABLE 3.8. Extended Burnup Fuel Rod Fission Gas Analysis

Rod No.	Gas Volume, cm^3 (STP)	Molar Percent				
		Xe	Kr	CO_2	Ar	O_2
5F-36	211.0	27.06	2.12	0.02	0.01	<0.01
5C-27	142.0	74.23	5.00	0.05	<0.01	0.02
CN-7	144.0	31.55	2.20	<0.01	<0.01	<0.01
CM-85	133.0	26.06	2.01	0.03	<0.01	0.03
		N_2	CO	He	H_2	CH_4
5F-36	211.0	4.28	<0.10	66.5	<0.01	<0.01
5C-27	142.0	6.27	<0.10	14.3	<0.01	<0.01
CN-7	144.0	0.04	<0.10	66.2	<0.01	<0.01
CM-85	133.0	0.10	<0.10	72.8	<0.01	<0.01

Internal Pressure Monitoring Experiment Evaluation

Vibrationally compacted $\text{UO}_2\text{-PuO}_2$ fuel rods instrumented to monitor internal gas pressure during irradiation were successfully irradiated in PRTR to determine the effects of fuel temperature and burnup on both sorbed and fission gas release. Peak rod power generations of approximately 20 kW/ft and estimated peak burnups of 11,000 MWd/MTM were achieved. The detailed postirradiation examination of the instrumented rods is nearly complete.

Fuel structures, β - γ autoradiographs, and α autoradiographs taken at axial positions among the four $\text{UO}_2\text{-PuO}_2$ fuel rods which operated at comparable power generations were compared (Figures 3.18 through 3.21). There is excellent structural consistency among sections that operated at comparable conditions to these significant burnups (Table 3.9). The structures formed in vibrationally compacted rods irradiated under the same conditions to low burnups are less consistent. These data also show that the extent of the sintering boundary is a more reliable structural indicator of operating conditions than the columnar grain growth boundary.

There was no evidence of sintering or homogenization of the $\text{UO}_2\text{-PuO}_2$ fuel material at rod power generations of 8 kW/ft with estimated maximum fuel temperatures of 1150 °C (Figure 3.18). However, sintering and homogenization occurred at power generations above 11.5 kW/ft. The extent of in-reactor restructuring and fission product migration which occurred at the peak power of 20 kW/ft (Figure 3.21) are representative of an operating history during which the peak power occurred for a short period of time early in the irradiation followed by extended operation at a lower power, which was the case with these rods.

FIGURE 3.18. Transverse Sections from the 8.0 kW/ft Positions in the Two Low Performance Instrumented Fuel Rods

FIGURE 3.19. Transverse Sections from the Peak Power (11.5 kW/ft) Positions in Two Low Performance Instrumented Fuel Rods

FIGURE 3.20. Transverse Sections from the 14.5 kW/ft Positions in Two High Performance Instrumented Fuel Rods

FIGURE 3.21. Transverse Sections from the Peak Power (20.6 kW/ft) Positions in the Two High Performance Instrumented Fuel Rods

TABLE 3.9. Comparison of Irradiated Fuel Structural Features in the Instrumented Vibrationally Compacted $\text{UO}_2\text{-PuO}_2$ Fuel Rods

Fuel Rod Number	Specimen Number	Max. Rod Power Generation, kW/ft	Sintering Boundary, % r	Column Grain Boundary, % r	Central Void, % r	Estimated Max. Fuel Temp., °C
2	1	8.0	0	0	0	1150
4	1	8.0	0	0	0	1150
2	2	11.5	48	0	0	1900
4	2	11.5	46	0	0	1900
7	1	14.5	61	52	6	2250
8	2	14.5	55	39	3	2250
7	1	20.6	75	65	12	<2800
8	2	20.6	78	69	15	<2800

The inner surface of the Zircaloy cladding was examined to determine the thickness of the ZrO_2 layer which formed during irradiation (Figure 3.22). These instrumented rods were not autoclaved prior to irradiation so consequently, the oxide layer formed during irradiation with trapped gases and moisture providing the limited source of oxygen. The maximum ZrO_2 layer thickness of about 0.0002 in. occurred at the peak power position. A maximum thickness at this location can be attributed to higher cladding temperatures and/or neutron flux enhanced corrosion rates. It is also possible and perhaps more likely that the ZrO_2 thickness variation is related to the average fuel temperature in the immediate proximity because the pressure monitoring data indicated that the released gases reacted rapidly with the cladding. The thicker ZrO_2 layers were more uniform with the most nonuniform layer occurring in the plenum region where there was no fuel present. Intermittent reaction between the $\text{UO}_2\text{-PuO}_2$ fuel and the ZrO_2 layer was also observed (Figures 3.22c and 3.22d).

565X

FIGURE 3.22. Typical ZrO₂ Layer on the Inner Cladding Surface of Instrumented Vibrationally Compacted UO₂-PuO₂ Fuel Rods

Two rods in a vibrationally compacted element containing ThO_2 -5 wt% PuO_2 fuel were also instrumented to monitor internal gas pressure during irradiation. This element was irradiated in PRTR at a peak rod power generation of 7.5 kW/ft with maximum fuel temperatures of approximately 1200 °C. As expected, no structural changes occurred in the fuel under these operating conditions (Figure 3.23).

FERTF Fuel Rod Evaluation

Examination of eight vibrationally compacted UO_2 -2 wt% PuO_2 fuel rods irradiated together in the Fuel Element Rupture Test Facility (FERTF) in PRTR permits an evaluation of fuel structural differences formed among rods irradiated under the same conditions. The eight rods studied were irradiated in the FERTF at a peak rod power generation of 16.4 kW/ft to a peak burnup of 380 MWd/MTM at a calculated maximum fuel temperature of 2550 °C. The rod-to-rod peak power variation was determined to be within $\pm 1\%$.

Examination of the fuel cross sections delineates the extent of variation in the distinguishing structural features, i.e., the equiaxed grain growth, columnar grain growth, and central void boundaries, among the eight rods (Table 3.10). Measurements indicate that the variation in the equiaxed grain growth boundaries is approximately 9% with variation of 22 and 8% in the columnar grain growth and central void boundaries, respectively. The size of the central void is related to the extent of the columnar grain growth boundary. In some rods, however, the formation of elongated columnar grains was difficult to distinguish (Figure 3.24).

FIGURE 3.23. Transverse Section from the Peak Power Position (7.3 kW/ft) in the Instrumented ThO_2 - PuO_2 Fuel Rod (FR-6)

The results provide further substantiation of the observation that the extent of the equiaxed grain growth boundary is the most reliable indicator of fuel operating conditions. Structural variations such as those observed among these rods tend to disappear at higher burnups.

TABLE 3.10. Summary of Distinguishing Structural Features Among Eight Vibrationally Compacted Rods Which Were Irradiated Under the Same Conditions

<u>Number</u>	<u>Equiaxed Grain Growth Boundary, % r</u>	<u>Columnar Grain Growth Boundary, % r</u>	<u>Central Void Boundary, % r</u>
FS-41	70	48	3
FS-42	69	50	6
FS-43	70	41	1
FS-44	74	53	6
FS-45	73	57	9
FS-46	78	35	3
FS-47	70	45	2
FS-48	74	50	3

7X

FIGURE 3.24. Transverse Composite Strips from Eight Vibrationally Compacted UO₂-2 wt% PuO₂ Fuel Rods with Identical Irradiation Histories. The fuel structures represent a linear rod power generation of 16.4 ± 0.2 kW/ft and a burnup of 380 MWd/MTM

4.0 REACTOR SAFETY

TRANSIENT TESTING

M. D. Freshley and T. B. Burley

A knowledge of the behavior of reactor fuels subjected to high energy short duration power excursions is essential to permit a complete appraisal of fuel performance and reactor safety. The transient behavior of oxide fuels is being investigated by irradiations in SPERT and TREAT to determine whether special considerations associated with adding PuO_2 enrichment to different types of UO_2 fuels would influence reactor operation.

A joint program involving Pacific Northwest Laboratory, General Electric Company, and Idaho Nuclear Corporation was initiated to investigate the possible effects on the transient behavior of adding PuO_2 enrichment to UO_2 fuels in different ways.

The first test series involves four Zircaloy-clad fuel pins containing cold-pressed-sintered enriched UO_2 pellets. Single 500 μm diameter PuO_2 spherical particles are strategically positioned within alternate pellets in the 5 in. long fuel column. The pins will be tested at three different energy levels in SPERT to evaluate the possible effects of adding the large PuO_2 particles, which is considered to represent a worst case situation. The irradiations are being preceded by thorough characterization of the pins and they will be followed by a complete evaluation of the irradiation effects. If no significant effects are detected with the initial series of single particle experiments, a series of tests with multi-particle and pre-irradiated UO_2 - PuO_2 pins may be considered.

Confirming experiments to compare the transient behavior of pelleted and vibrationally compacted fuels are being conducted in the transparent vessel in TREAT. Detailed evaluation

of high speed motion pictures taken of the rods while undergoing transient irradiation permit studies of the failure thresholds and failure modes for the different fuel types.

Six pins (four cold-pressed-sintered pellet and two vibrationally compacted) containing enriched UO_2 fuel were fabricated for irradiation in TREAT. The pins, which were fabricated by prototypic processes, are nominally 9/16 in. OD with 0.030 in. thick Zircaloy cladding. They are designed with an active fuel length of 5.5 inches and a scaled-down fission gas plenum. Hafnium neutron absorber cups will be placed over each end of the pins to eliminate power peaking at the ends of the fuel columns (Figure 4.1).

The pins will be irradiated at what is expected to be the failure threshold energy for each type of fuel and slightly above.

PRTR ZIRCALOY-2 PRESSURE TUBING

The draft copy of BNWL-SA-3143, "Fracture Behavior of Flawed Zircaloy-2 Pressure Tubes," has been completed and submitted to NUCLEAR ENGINEERING AND DESIGN for publication. This paper deals with differences observed in testing tubes with simular flaw sizes produced by both fatigue crack generation and machining. The paper also applies an empirical failure criterion to the fatigue test results to determine their relationship to published values of fracture toughness.

This paper shows that fracture testing of Zircaloy-2 pressure tubes for simulating the effect of a sharp crack should involve the use of fatigue cracks rather than machined notches or slots, since the latter do not represent as severe a stress concentration as an actual crack. It has also been shown that an empirical failure criteria may be used to treat the failure of pressure tubes containing axial through-wall cracks. The criterion may be used in conjunction with published values of

FIGURE 4.1. Zircaloy-Clad TREAT Pins with and Without the Hafnium Neutron Absorber Cups Placed on Each End

4.0 REACTOR SAFETY

TRANSIENT TESTING

M. D. Freshley and T. B. Burley

A knowledge of the behavior of reactor fuels subjected to high energy short duration power excursions is essential to permit a complete appraisal of fuel performance and reactor safety. The transient behavior of oxide fuels is being investigated by irradiations in SPERT and TREAT to determine whether special considerations associated with adding PuO_2 enrichment to different types of UO_2 fuels would influence reactor operation.

A joint program involving Pacific Northwest Laboratory, General Electric Company, and Idaho Nuclear Corporation was initiated to investigate the possible effects on the transient behavior of adding PuO_2 enrichment to UO_2 fuels in different ways.

The first test series involves four Zircaloy-clad fuel pins containing cold-pressed-sintered enriched UO_2 pellets. Single 500 μm diameter PuO_2 spherical particles are strategically positioned within alternate pellets in the 5 in. long fuel column. The pins will be tested at three different energy levels in SPERT to evaluate the possible effects of adding the large PuO_2 particles, which is considered to represent a worst case situation. The irradiations are being preceded by thorough characterization of the pins and they will be followed by a complete evaluation of the irradiation effects. If no significant effects are detected with the initial series of single particle experiments, a series of tests with multi-particle and pre-irradiated UO_2 - PuO_2 pins may be considered.

Confirming experiments to compare the transient behavior of pelleted and vibrationally compacted fuels are being conducted in the transparent vessel in TREAT. Detailed evaluation

fracture toughness for various degrees of cold-work, hydride concentration, irradiation levels, and temperatures, thus avoiding expensive component tests for each separate condition. However, conducting an occasional component test to verify the validity of the criterion would be advisable.

Also, a report reviewing all the pressure tube surveillance work done in support of PRTR is being prepared. Unflawed strength, flawed strength (both fatigue and machined cracks), and stress rupture data for the tubes in the as-received, hydrided and irradiated conditions will be presented along with other property data.

5.0 PLUTONIUM RECYCLE TEST REACTOR

PRTR DEACTIVATION

K. L. Young

The activity at PRTR during the past quarter has been surveillance of the plant and shipment of PRTR fuel to chemical separations facilities. The surveillance activities have disclosed no problem areas during the first three months of deactivation.

The deactivation of PRTR required that a basic core loading of 70 fuel assemblies be retained for possible future operation of the reactor. All fuel material in excess of these 70 assemblies were to be disposed of through chemical separations facilities. This excess fuel material is being moved to facilities operated by Atlantic-Richfield Hanford Company (ARHCO) in 25 shipments. Eighteen of the fuel shipments were completed during January and February.

6.0 PUBLICATIONS AND PRESENTATIONS

PUBLICATIONS

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2. L. W. Brackenbush and L. G. Faust. Neutron Spectra of Plutonium Compounds, Part I: ^3He and ^6Li Spectrometer Measurements, BNWL-1262. February 1970.
3. C. R. Hann, T. B. Burley, and M. D. Freshley. Proposal to Irradiate Intentionally Defected Mixed-Oxide Fuel Rods in the ETR M-3 Loop Facility - The 23 kW/ft Test, Series - (BNW-5-1 through -6), BNWL-CC-2445. February 1970.
4. C. E. Leach. Flow Pressure Drop Characteristics of 19-Rod Wire Wrapped Fuel bundle, Annular Orifice, Annular Flow Passage, and In-Pile Tube Simulating the ETR M-3 Loop, BNWL-CC-2431. January 1970.
5. T. B. Burley and M. D. Freshley. Internal Gas Pressure Behavior in Vibrationally Compacted Mixed-Oxide Fuel Rods During Irradiation, BNWL-SA-2960. November 1969; Submitted for publication in Nuclear Applications and Technology.

PRESENTATIONS

1. M. D. Freshley, D. G. Carter, and T. B. Burley. "Plutonium Recycle Fuel Evaluations at PNL," BNWL-CC-3148 S. February 1970. Submitted for presentation at the 16th Annual Meeting of the American Nuclear Society in Los Angeles, June 28-July 2, 1970.
2. T. B. Burley, M. D. Freshley, and D. G. Carter. "Internal Gas Pressure Behavior in Vibrationally Compacted Mixed-Oxide Fuel Rods - Postirradiation Examination," BNWL-SA-3159 S. Submitted for presentation at the 16th Annual Meeting of the American Nuclear Society in Los Angeles, June 28-July 2, 1970.

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