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FAST FLUX
TEST FACILITY
PROJECT REPORT

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**EXPERIMENT DESCRIPTION
AND HAZARDS EVALUATION FOR THE
PACIFIC NORTHWEST LABORATORY
MIXED OXIDE (UO_2 - PuO_2) CAPSULES FOR
INSERTION IN SUBASSEMBLY X011**

L. A. PEMBER

AUGUST 1968

**AEC RESEARCH &
DEVELOPMENT REPORT**

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PACIFIC NORTHWEST LABORATORY

RICHLAND, WASHINGTON

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MIXED OXIDE ($\text{UO}_2\text{-PuO}_2$) CAPSULES FOR
INSERTION IN SUBASSEMBLY X011

By

L. A. Pember

FAST FLUX TEST FACILITY PROJECT

Approved By

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August 1968

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EXPERIMENT DESCRIPTION AND HAZARDS EVALUATION
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L. A. Pember

INTRODUCTION

PURPOSES AND OBJECTIVES OF THE EXPERIMENT

Pacific Northwest Laboratories (PNL) has reached agreement with the AEC Office of Research and Development Technology on the experiment plan (PNL X-1 through 7) to take advantage of spaces made available by the recent removal of the five pins from the EBR-II Subassembly XO11. Approval-in-principle has been requested from the Argonne National Laboratory Director's Office to irradiate five encapsulated mixed oxide pins in Subassembly XO11 when it is reinserted into the EBR-II. These tests will provide the desired data during the additional exposure of approximately 2200 MWD planned for Subassembly XO11.

The proposed tests comprise five encapsulated pins which will operate at relatively low linear power generation rates to provide information on the following:

1. Kinetics of fuel-cladding gap closure
2. Effect of initial gap size on the fuel radial temperature profile
3. Fission gas release as a function of temperature
4. Extent of fuel restructuring which will take place at these power levels.

These pins will provide support information for the three planned 37 unencapsulated pin subassembly irradiations, for which PNL has received approval-in-principle from the LDO (Reference 1), and preliminary information on the performance of prototypical, mechanically blended mixed oxide FTR fuel. The data from the five test pins will strengthen the technical basis for the fuel-cladding gap specification for FFTF driver fuel pins.

The group of five pins will nominally comprise three pins containing naturally enriched UO_2 -25 wt% PuO_2 and two pins containing 45% enriched UO_2 -25% PuO_2 . Fabricated fuel is available for the natural UO_2 - PuO_2 pins and the pellets containing enriched uranium have been fabricated. For these tests, the smeared density of the fuel will be held constant while the fuel-cladding gap and pellet density are varied.

A Row 3 position is planned for the reinsertion of X011 in EBR-II. A reactor power of 45 MW will produce a maximum linear heat generation rate of approximately 6 kW/ft in the NUO_2 - PuO_2 pins, which is comparable with the minimum linear power generation rate of the FTR driver fuel pins. The 45% enriched UO_2 - PuO_2 pins will generate approximately 10.6 kW/ft (maximum) which is near the average linear pin power generation rate anticipated in the FTR. Most of the mixed oxide pin experiments conducted in EBR-II tests have operated with higher heat ratings. Therefore, these proposed tests will add to the few existing data on the performance of mixed oxide fuel at the minimum and average power ratings expected in the FTR.

Threshold flux monitors will be incorporated into each capsule. The different monitor materials to be used will

include iron, nickel, copper, scandium and titanium. Lengths of monitor wire will be sealed in the stainless steel hypodermic tubing used for wire wrapping the fuel pins to provide spacing between the fuel pins and the inner wall of each capsule. Results from these monitors may help in resolving differences in flux values which are obtained from the fission monitors used by ANL and from the activation monitors used by many other experimentors.

DESCRIPTION OF THE FUEL AND CLADDING AND SUMMARY OF PRIOR IRRADIATION EXPERIENCE

The fuel to be used in these experiments is hypostoichiometric ($O/M = 1.97 - 1.99$) mechanically mixed 25% PuO_2 - 75% UO_2 which has been sintered in a hydrogen atmosphere at 1550 °C for 4 hr to yield pellet densities of 90 to 96% TD. The sintering process results in the formation of a solid solution of PuO_2 and UO_2 which is virtually indistinguishable from solid solutions of PuO_2 and UO_2 prepared from the coprecipitation process.

The cladding material for these fuel specimens is Type 304 SS. Material for the capsule components was obtained from Argonne National Laboratory.

The prior irradiation experience of mixed oxide (PuO_2-UO_2) fuel has been recently reviewed in a number of papers.⁽²⁻¹⁰⁾ The latest paper⁽¹⁰⁾ presents preliminary postirradiation examination data from encapsulated mixed-oxide fuel specimens irradiated in EBR-II to peak burnups of about 5 at.%. In general, it can be concluded that the performance of mixed oxide fuel has been reasonably well established up to burnups of 5 at.%. Tests currently in progress in EBR-II and Dounreay will shortly provide data on fuel performance at burnups greater than 5 at.% in a fast flux. Thermal flux tests of mixed oxide fuel specimens to burnups beyond 12 at.% have also been successfully completed.⁽¹¹⁾

MIXED OXIDE CAPSULES

FUEL ELEMENT AND CAPSULE DATA

The design data are presented in Table I, except for the isotopic analyses of the uranium and plutonium feed materials given in Table II. As-fabricated data will be provided in an addendum to this Experiment Description and Hazards Evaluation.

Figures 1 through 3 delineate the fuel specimen and capsule assembly details for these tests. The purpose of the extensometer (Figure 3) in each fuel specimen is to record the maximum elongation of the fuel column and to prevent separation of the pellets during handling. Figure 4 shows the enclosed sodium bonding loop. Appendix A contains the drawings of the shipping cask and shipping tube.

Data not provided in Table I which are common to all specimens and capsules are:

1. The fuel specimens are all helium gas-bonded to the cladding.
2. All the fuel specimens will initially contain helium at 1 atm pressure. The volume of helium at STP is calculated to be 7.1 cm^3 .
3. The design value of the capsule sodium level as shown in Figures 1 and 2 is $5.172 \text{ in.} \pm 1 \text{ in.}$ above the top of the fuel. Figure 5 shows the level as a function of temperature. The capsule is backfilled with helium at 1 atm pressure after being sodium filled. The calculated helium volume is 12.6 cm^3 at STP. Volume and pressure as a function of temperature are given in Figures 6 and 7 respectively.
4. Capsule identification will be by numbers and letters stamped in the locations shown in Figures 1 and 2.

TABLE I. Fuel Specimen Data (Design Values)

Capsule Design- nation	Fuel Column Length	Fuel Pellet OD, in.	Fuel Pellet Density % TD	Fuel Smeared Density % TD	Fuel Weight, g	Fuel Clad Diametral Gap, in.	Grams $^{235}\text{U}/\text{U}$	Activation Flux Monitor
PNL-X-1	13.5	0.2135	91.3	88	79.3	0.004	0.0072	Sc
PNL-X-3	13.5	0.2115	93	88	79.3	0.006	0.0072	Ti
PNL-X-4	13.5	0.2095	94.8	88	79.3	0.008	0.0072	Cu
PNL-X-6	13.5	0.2115	93	88	79.3	0.006	0.45	Ni
PNL-X-7	13.5	0.2095	94.8	88	79.3	0.008	0.45	Fe

TABLE II. Isotopic Analyses of Uranium and Plutonium Feed Materials for Fuel Specimens

Isotope	Percent	
	Natural	45% Enriched
^{235}U	$^{235}\text{U}/\text{U} = 0.72$	45.0
^{238}U	$^{238}\text{U}/\text{U} = 99.2$	55.0
^{239}Pu	87.3	86.8
^{240}Pu	11.2	11.6
^{241}Pu	1.4	1.5
^{242}Pu	0.1	0.1

QTY	DESCRIPTION
1	ASSEMBLY
2	FUEL ROD WITH THERMOCOUPLE .020 DIA X 29 1/2\"/>
1	ROD .012\"/>
1	END CAP
54	FUEL, SOLID PELLETS, 25% PUO ₂ , 75% UO ₂ , 1/2\"/>
2	INSULATOR PELLET, .014 DIA X 1/2\"/>
1	PEDESTAL
1	TUBE .050 DIA X .016 WALL X 3 1/2\"/>
1	SPACER TUBE .188 OD X .148 ID X 10 7/32 LG, 304 SST
AR 10	TUBE .040 OD X .034 ID X 32 3/4 LG, 304 SST
1	EXTENSOMETER, H-B-5214

GENERAL NOTES

UNLESS OTHERWISE SPECIFIED

1. TOL FRAC $\frac{1}{64}$ DEC 3.005 ANG $\pm 0^{\circ}30'$
2. ALL MACHINED SURFACES 63 (ASA B46.1 1962)
3. REMOVE ALL BURRS & BREAK ALL SHARP EDGES EXCEPT ON WELD LIPS AND TUBE ENDS
4. ALL MATL TO BE AS SPECIFIED
5. WELDING TO BE TIG WITH NO FILLER ROD USED
6. REFERENCE DIMENSIONS TO BE CONFIRMED BY RADIOGRAPHY

H-3-28930

FIGURE 2. X011 Replacement Capsule Fuel Pin Assembly and Details

QTY	PT NO	DESCRIPTION	MATL OR CAT NO
1	1	ASSEMBLY	
1	2	PIN	
1	3	SPRING	
1	4	SLEEVE	
1	5	PIN, $\frac{1}{8}$ DIA $\times \frac{3}{16}$ LG "SPIROL" LIGHT DUTY 302 SST

H-3-15274

FIGURE 3. Extensometer Assembly and Details

Neg. 0670169

FIGURE 4. Enclosed Sodium Bonding Loop

FIGURE 5. Sodium Level Versus Sodium Temperature

FIGURE 6. Initial Void Volume as a Function of Capsule Temperature

FIGURE 7. Internal Capsule Pressure as a Function of Capsule Temperature

These symbols will correspond to those given in Table I for each fuel specimen.

5. Depleted UO_2 pellets are used as insulators at each end of the fuel column. The DUO_2 pellet diameter is 0.214 in. and a 0.5 in. length is used at each end of the fuel column. DUO_2 weight is 5.8 g/pin.
6. Fuel pin cladding is 0.250 in. OD, 0.2175 in. ID Type 304 SS.

As illustrated in Figures 1 and 2 the spiral wire wrapping has been replaced by 304 SS tubing which contains copper, nickel, scandium, titanium or iron flux monitor wire. The tubing is sealed so the flux monitor wire does not come in contact with the bonding sodium. The incorporation of the flux monitor wire in this manner is in cooperation with the PNL FFTF-oriented dosimetry studies program.

FABRICATION AND TESTING OF CAPSULES PRIOR TO IRRADIATION

Fuel Specimens

Quality assurance for the fuel specimens will be attained through:

1. Ultrasonic and eddy-current inspection of cladding material. Standards for rejection was any defect greater than 1 mil deep, 4 mils wide or 30 mils long.
2. Air gage and profilometer measurements for the cladding ID and OD, respectively.
3. Selective fitting of fuel pellets to cladding to obtain specified test parameters (gap and smeared density).
4. Mass spectrometer helium leak tests of closure welds. Limit of leak detector sensitivity is $1.0 \times 10^{-10} \text{ cm}^3/\text{sec}$.

5. X-radiography of completed specimens to ascertain proper fuel and component positions, dimensions, and weld quality.

Capsules

1. Capsule tubes were subjected to ultrasonic and eddy current inspection to verify soundness. Any defect greater than 10% of the wall thickness was cause for rejection.
2. Closure welds were x-radiographed and helium leak tested. Leak detector sensitivity was $1.0 \times 10^{-10} \text{ cm}^3/\text{sec.}$
3. Sodium filling was accomplished in accordance with the procedure given in Appendix B.
4. Sodium bonding was eddy current tested to detect voids larger than 0.031 in. diam.
5. X-radiographs of the capsules were taken to verify fuel location and component positions and to evaluate further the sodium bond integrity.

SUBASSEMBLY DATA

Capsule Materials and Weights

<u>Fuel (from Table I)</u>	<u>Total Weight, g</u>
5 specimens at 79.3 g each	396.5
<u>Stainless Steel</u>	
<u>Fuel Pin</u>	
<u>Cladding</u>	
5 specimens 0.250 in. OD x 0.016-in. wall at 47.5 g each	237.5
<u>Pedestal</u>	
5 specimens at 5.12 g each	25.60

Spacer Tube

5 specimens at 4.35 g each	21.75
----------------------------	-------

Spiral Wrap Tube

5 specimens at 4.45 g each	22.25
----------------------------	-------

Extensometer

5 specimens at 6.33 g each	31.65
----------------------------	-------

End Cap

5 specimens at 2.98 g each	<u>14.90</u>
	355.65

CapsuleTop Cap

5 specimens at 24.68 g each	123.40
-----------------------------	--------

End Plug

5 specimens, 2 per specimen at 3.37 g	33.70
--	-------

Bottom Cap

5 specimens at 10.50 g each	52.50
-----------------------------	-------

Tube

5 specimens at 106.20 g each	531.00
------------------------------	--------

Fill Tube

5 specimens at 1.0 g each	<u>5.0</u>
	646.87

Total stainless steel	1000.00
-----------------------	---------

Nickel

5 specimens at 11.5 g	57.5
-----------------------	------

Sodium

5 specimens at 12.75 g	63.75
------------------------	-------

Copper

1 specimen at 1.35 g	1.35
----------------------	------

Iron

1 specimen at 1.2 g	1.2 g
---------------------	-------

Nickel

1 specimen at 1.2 g	1.2 g
---------------------	-------

Scandium

1 specimen at 1.8 g	1.8 g
---------------------	-------

Titanium

1 specimen at 0.7 g	0.7 g
---------------------	-------

Desired Reactor Location

Subassembly XO11 is scheduled for reinsertion in a Row 3 position. At the present time the exact position is unknown. Calculations are made on the basis of a 3E2 position.

Capsule Positions

These capsules may be loaded in any available space in the subassembly. It is desirable to place capsules of equal UO_2 enrichment in positions which will give comparable power generations.

EBR-II TEST RESULTS

NUCLEAR EFFECTS OF THE SUBASSEMBLY IN EBR-II

This can be calculated best by EBR-II personnel.

PERFORMANCE OF THE SUBASSEMBLY DURING IRRADIATION

The performance predictions for each fuel specimen were made with the aid of the EBR-II Hazards Analysis code prepared by the late C. R. Reinke, Argonne National Laboratory, to predict fuel specimen performance for a number of EBR-II irradiations including mixed oxide specimens SOV-5 and SOV-6.⁽¹²⁾ The equations used in the code are those given in the guide.⁽¹³⁾ Fissioning rates are based on the 91 subassembly core.⁽¹⁴⁾ Gamma heating values are based on the revised values for uranium, sodium and iron given in Figure 17 (Revision 2, corrected by MAB) of the guide.

Specimen Heat Generation Rates and Temperatures

Table III presents the calculated heat generation rates, temperatures, capsule surface heat fluxes, and coolant ΔT 's for each specimen at the peak position in the subassembly. Curves of these parameters as a function of flow rate will be found in Appendix C.

The total heat generation, coolant flow rate and orifice size of the subassembly can best be calculated by EBR-II personnel.

Thermal conductivities for the materials in the capsules are presented in Table IV.

Figure 8 presents experimental data for gap conductivity in 0.25 in. diam mixed oxide pins.

Figures 9 and 10 show axial temperature with and without uncertainty factors for the hottest fuel specimen

TABLE III. Specimen Heat Generation Rates and Temperatures*

Specimen Number	Maximum Heat Generation Rate, kW/ft**	Maximum Mean Capsule Wall Temperature, °C†	Maximum Capsule Surface Heat Flux, W/cm ²	Coolant ΔT , °C†
PNL-X-1	5.8	440	63.9	56.1
PNL-X-3	5.8	440	63.9	56.1
PNL-X-4	5.8	440	63.9	56.1
PNL-X-6	10.5	496	115.9	102.0
PNL-X-7	10.5	496	115.9	102.0

TABLE IV. Thermal Conductivity Values for Capsule Materials

Material	Thermal Conductivity
	Btu/hr-ft ² - F°/ft
304 SS	9.9
Sodium	36.2
Fuel	Variable thermal conductivity (see Figure 11) for $\int k dt$ vs T curve)
Fuel-Cladding Gap	1000 Btu/hr-ft ² - F°
Conductance	(see Figure 10)

* Based on core position 3E2.

** Based on the hottest position in the subassembly.

† Based on a coolant flow rate of 1.5 gal/min/specimen.

FIGURE 8. Conductivity Versus Gap Size, Argon Atmosphere

FIGURE 9. *Calculated Axial Temperatures For PNL-X-7 in Position 3E2 at 45 MW Power Without Uncertainty Factors (1.5 gal/min Flow)*

FIGURE 10. *Calculated Axial Temperatures for PNL-X-7 in Position 3E2 at 45 MW with Uncertainty Factors (1.5 gal/min)*

(PNL-X-6). The fuel center temperature was estimated by means of the $\int_0^{T_c} k dt$ versus temperature curve (Figure 11) to account for the variation of fuel thermal conductivity with temperature. A typical calculation is given in Appendix C.

Radial and Axial Expansion and Growth of Fuel

At the low burnup anticipated in these capsules, no fuel swelling is expected. The fuel cladding gap will accommodate the thermal expansion of the fuel without stressing the cladding.

Fission Gas Pressure and Resultant Cladding Stress

The volume of fission gases formed in the hottest specimen after 10,000 MWd/tonne is 14.7 cm^3 . Assuming 100% gas release and a plenum temperature of 460°C , the pressure in the pin will be 186 psig.

Cladding Thermal and Hoop Stress

The calculated thermal and hoop stresses in the cladding of the hottest specimen are:

<u>Specimen</u>	<u>Thermal Stress, psi</u>	<u>Hoop Stress, psi</u>
PNL-X-6	11,800	1240

Irradiation Effect on Cladding and Capsule Material Strength

Based on currently available data, an increase in cladding yield strength and a decrease in ductility can be anticipated as a consequence of fast flux irradiation. However, at the low anticipated exposures there should be no appreciable effect.

Calculated capsule wall thermal stress in the hottest capsule is 9690 psi. Calculated hoop stress is 353 psi under normal conditions and 677 psi with a ruptured fuel pin.

FIGURE 11. Curve Mixed Oxide $\int_0^{T_c} k dt$ Design

POSTIRRADIATION CHARACTERISTICS OF SUBASSEMBLY AND CAPSULES

The decay heat production of the subassembly must be calculated by EBR-II personnel. Figure 12 shows the decay heat of each type pin as a function of time.

The minimum cooling period required in the EBR-II storage basket will be dependent upon the higher exposure pins contained in the remainder of the subassembly.

The estimated gamma radiation level of subassembly and stainless steel hardware must be calculated by EBR-II personnel.

Table V shows the isotopic curie content for each type of capsule.

Drawings of the containers to be used for shipment of the capsules to and from EBR-II are included in Appendix A.

HAZARDS EVALUATION

A number of hazards evaluations for the irradiation of encapsulated mixed oxide fuel specimens have been prepared by others.^(12, 15, 16, 17, 18) These analyses show that the irradiation of encapsulated mixed oxide fuel specimens does not present any significant hazard to the operating safety of EBR-II. Moreover, as was recently demonstrated, the presence of failed fuel specimen cladding (Specimen F-2-F, Subassembly XG06) can be successfully tolerated by the capsule.

Reactivity Effects

The overall reactivity effect of the subassembly must be calculated by EBR-II personnel. The reactivity effect associated with axial fuel motion has not been investigated. However, it is known that no significant reactivity effects will occur in the absence of fuel melting.⁽⁶⁾ Since the specimens will normally operate at less than 11 kW/ft and

TABLE V. *Isotopic Curie Content of Each Capsule after 10 Days Cooling*

<u>Isotope</u>	<u>Curies/Rod 6.0 kW/ft</u>	<u>Curies/Rod 10.6 kW/ft</u>
Sr	81	143
Y	94	166
Zr	115	204
Nb	61	107
Mo	51	91
Tc	52	93
Ru	121	214
Rh	128	227
Ag	7	12
Sm	2	3
Sb	8	14
Te	94	167
I	143	254
Xe	152	269
Cs	1	2
Ba	188	332
La	213	375
Ce	204	260
Pr	236	417
Nd	77	135
Pm	12	21
Sm	1	2
Eu	4	6
Tb	1	2
Total	<u>2051</u>	<u>3623</u>

since melting of mixed oxide fuel has not been observed at less than 22 kW/ft⁽²⁾ one must postulate a gross overpower condition or severe transient in order to assume fuel melting. Furthermore, the performance of the fuel specimens at a 25% overpower condition (equivalent to a reactor power level of 56.25 MW) has been analyzed. These analyses (Appendix C) show that the calculated fuel specimen and capsule temperatures are still well within the limits of the EBR-II irradiation test criteria.

FIGURE 12. Decay Heat From Single Pin as a Function of Cooling Time

Flux Perturbation

The flux perturbation in a driver fuel subassembly adjacent to the experimental subassembly cannot be calculated without knowledge of the remainder of the experiments in the subassembly.

Misloading Subassemblies

The assumed position of the subassembly (3E2) coincides with the highest flux position for Row 3.

Sodium Bond Defects

Sodium bond discontinuities are not suspected in these test capsules. Previous calculations,⁽¹⁶⁾ however, show that a 0.375 in. x 0.625 in. void in the annulus of a specimen operating at more than 20 kW/ft would not result in capsule failure.

Immersion in Water

Flooding these test subassemblies with water will not result in a nuclearly critical condition.

Consequence of Fuel Specimen Cladding Failure

Failure of fuel specimen cladding with attendant release of fission gases to the capsule results in a hoop stress of 677 psi in the hottest high-burnup specimen. The thermal stress of 9690 psi combined with this hoop stress is still well below the failure point of the capsule wall material.

Fuel Contact with Capsule Wall

Rupture of the fuel specimen cladding might result in the ejection of fuel toward the capsule wall. The sodium that is present would provide a better heat transfer bond

between the fuel and the capsule wall, resulting in a decrease in fuel temperature. The change in heat flux at the capsule-coolant interface would be small unless there were axial fuel redistribution.

Capsule Leak

If a leak were to develop at the bottom of the capsule without fuel specimen failure, the sodium level in the capsule would vary with reactor coolant pressure as shown in Figure 13.

Nonstandard Materials

No materials other than standard subassembly and capsule structural materials are exposed to the reactor coolant.

Consequences of Capsule Failure

In the unlikely event of gross capsule failure:

1. The fission product inventory (see page 25) would be released to the reactor coolant.
2. Leaching soluble fission products from the fuel and entrainment of the fuel particles by the sodium would be the effects of a large rupture in both the cladding and capsule walls.
3. The capsules are sufficiently isolated so that the probability of failure propagation from one capsule to another is nearly nonexistent.
4. The capsules are isolated from the reactor by the subassembly hardware in addition to that separating the capsules from one another. Thus, the probability of failure propagation to an adjacent subassembly appears extremely unlikely.

Consequences of Cleaning Failed Capsule

Consequences of subjecting a defective radioactive capsule with defective fuel cladding to the cleaning process in

FIGURE 13. Sodium Level Versus Reactor Coolant Pressure in Capsule with Leak at the Lower End (Specimen PNL-X-7)

the IBC would depend on the rupture; much or none of the activity could be released depending on the geometry around the rupture. It is possible that a double rupture might escape detection at this point since the defective fuel could release all its fission gas in the reactor and then have the rupture close during the cleaning process. Analysis of the behavior of subassembly X011 should provide further guidance in these respects.

Behavior During Hypothetical Accident

In accordance with discussion with R. Neidner⁽¹⁸⁾ the power transient described as Case 4 in Reference 19 (page 206) was not used as a basis for evaluating the performance of the test specimens under abnormal conditions. Instead, the steady-state temperatures at a 25% overpower condition (56.25 MW reactor power) were determined (Figure 14). These analyses show that with uninterrupted coolant flow the hottest capsule (specimen number PNL-X-7) attains a maximum mean capsule wall temperature of 551 °C (1024 °F) with uncertainty factors included (Figure 15). This is substantially less than the 1105 °F specified in criterion V-A-4 in the guide.⁽¹³⁾

Calculations by others^(12, 15, 16, 17) on the temperature history of mixed oxide fuel specimens (initially operating at substantially higher rod powers than the present specimens) under the prescribed power transient have shown that the hazard to the reactor from mixed oxide specimens is no greater than that from the EBR-II driver fuel itself.

Behavior During Loss of Flow Accident

The ANL EBR-II Hazards Analysis Code was also used to evaluate fuel specimen behavior during the prescribed loss of

FIGURE 14. Calculated Axial Temperature Profile for PNL-X-7 in Position 3E2. (Reactor Power 56.25 MW; 13.2 kW/ft Peak Without Uncertainty Factors)

FIGURE 15. Calculated Axial Temperature Profile for PNL-X-7 in Position 3E2. (Reactor Power 56.25 MW; 13.2 kW/ft Peak With Uncertainty Factors)

flow accident (paragraph 13, page 50 of the guide).⁽¹³⁾

Figure 16 shows the results of these calculations for the hottest specimen.

Additional Loss of Flow Calculations

The temperatures calculated in the preceeding section show that failure of the capsule need not be anticipated, thus, further calculations are not required.

FIGURE 16. Capsule and Specimen Temperatures During Loss of Flow Accident for Hottest Specimen in Subassembly Position 3E2/1.

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APPENDIX A

REFERENCE DRAWINGS

ALL PARTS, UNLESS OTHERWISE SPECIFIED, SHALL
CONFORM TO THE FOLLOWING

1. TOLERANCES FRACTIONAL $\pm .005$
DECIMAL $\pm .005$
ANGULAR $\pm 1^\circ$
2. PARTS TO SHOW GOOD WORKMANSHIP, FREE FROM BURRS, UNMARRED AND CLEAN.
3. BREAK ALL CORNERS.
4. ALL MACHINED SURFACES $\sqrt{}$
5. ALL MATERIAL TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY.
6. CASK TO BE WELDED WATERTIGHT.
7. ALL EXPOSED SURFACES INCLUDING WELDS TO BE POLISHED.
8. LEAD FILL SHALL BE MADE IN ONE CONTINUOUS POUR TO ELIMINATE VOIDS. PRE HEAT CASK BEFORE STARTING LEAD POUR TO ASSURE ALL THE LEAD IN THE CASK IS MOLTEN; AS THE LEAD SOLIDIFIES THE PIPING SHALL BE KEPT FILLED WITH MOLTEN LEAD. WELD IN LAST PLUG (PART 5) AFTER CASK HAS COOLED.
9. SEQUENCE OF ASSEMBLY, MACHINING & WELDING TO BE VENDORS RESPONSIBILITY.
10. ALL MATL TO BE 300 SERIES SST UNLESS OTHERWISE SPECIFIED.

FIGURE A.1. Shipping Cask for Irradiated Fuels (Page 1 of 3)

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FIGURE A.2. Shipping Cask for Irradiated Fuels (Page 2 of 3)

FIGURE A.3. Shipping Cask for Irradiated Fuels (Page 3 of 3)

QTY.	NO.	DESCRIPTION	MATL	REMARKS
	1	ASSEMBLY		
	2	BASE		
	3	BRACKET		
	4	YOKE		
	5	YOKE		
10	6	CAP SCR 1/2-6UNC-2A-6" LG	STL	HEX HD
10	7	CAP SCR 1/2-6UNC-2A-6" LG	STL	HEX HD
	8	TURNBUCKLE 1/2 DIA X 6" BODY ONLY	GALVANIZED STL	CROSS-DRIVE LINCHPIN
	9	JAM NUT 1/2-6UNC-2B HEX	STL	
	10	NUT 1/2-6UNC-2B HEX		
	11	LOCK WASHER 1/2		
	12	SPACER 1/2 ID X 2 1/2 OD X 1/4 THK		

GENERAL NOTES

- ALL PARTS, UNLESS OTHERWISE NOTED, SHALL CONFORM TO THE FOLLOWING:
1. TOLERANCES FRACTIONAL $\pm \frac{1}{16}$ DECIMAL ± 0.005 ANGULAR $\pm 2^\circ$
 2. PARTS TO SHOW GOOD WORKMANSHIP, FREE FROM BURRS UNMARRIED AND CLEAN
 3. ALL MACH SURFACES 125
 4. ALL MATL TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY
 5. PAINT PARTS 2 & 3 WITH 1 COAT PRIMER & 2 COATS MACH GRAY PAINT

PART NO.	DIM A	DIM B	MATL
4	13 1/2	6 1/2	MILD STL
5	22	2 1/2	

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FIGURE A.4. Pallet for Irradiated Fuels Shipping Cask

① ASSEMBLY
ALL WELDED CONST

SECT "A-A"

VIEW "B-B"

GENERAL NOTES

1. FRACTIONAL TOLERANCE $\pm \frac{1}{32}$ UNLESS OTHERWISE SPECIFIED
2. PAINT ALL EXTERIOR SURFACES WITH ONE COAT PRIME & TWO COATS DARK HAMMERTONE BLUE
3. ALL MATERIAL TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY

DETAIL "C"

H-3-14721

FIGURE A.5. Model 10 Shipping Tube

APPENDIX B

SODIUM BONDING PROCEDURE

APPENDIX B

EBR-II SODIUM BONDING PROCEDURESCOPE

This procedure covers the sodium bond of cladding to cladding EBR-II irradiation capsules.

PROCEDURES

1. Make up an assembly of three capsules and helium leak check.
2. Purge capsules with helium to exclude air.
3. Mount the capsules in the bonding furnace so the desired level corresponds with the eddy current coil in the level detection line. Connect the capsules to the sodium fill lines.
4. Turn on all heaters; set controllers for the sodium supply vessel, heat tape on piping, and sump, at 350 °F.
5. A continuous vacuum is to be maintained on the sump.
6. Blow down through the level detection line to the sump to clear the line of sodium.
7. Blow down through the capsules to the sump to ensure that the line is clear.
8. Open all valves required to evacuate the system with the exception of the sodium supply vessel.
9. Check and adjust the helium pressure in the sodium supply vessel. Pressure should read ~6 psi at operating temperature.
10. When vacuum reaches 50 μ , switch on bonding furnaces and heat capsules to 1000 °F.
11. Fill capsules with sodium until the level indicator shows the sodium to be above the desired level.
12. The capsules should remain at 1000 °F with the sodium above the desired level and a vacuum above the sodium, for 2 hr.

13. Drain the sodium from the capsule into the sump and refill the capsule to the desired level with clean sodium from the sodium supply vessel.
14. After the filling has been completed, pressurize the space above the sodium in the capsule to 15 psi with helium.
15. Recheck sodium level and adjust if necessary.
16. Soak capsules at 1000 °F with 15 psi helium above the sodium for 2 hr.
17. Recheck sodium level periodically during the soak period and adjust as necessary.
18. Close all valves in the sodium system, turn off both bonding furnaces and line heaters.
19. Open the door of the bottom bonding furnace and allow the capsules to cool to room temperature.
20. Check the sodium level in each capsule with the portable level-detector before removing the capsules from the bonding loop.
21. Release the helium pressure above the sodium in the capsules to atmospheric pressure and remove the capsule from the bonding loop. Reheat to 1000 °F and directionally cool from the bottom with flowing air.
22. Eddy current test capsules for voids in the sodium annulus.
23. Helium leak test completed capsule.

APPENDIX C

TYPICAL CALCULATIONS

APPENDIX C

TYPICAL CALCULATIONSPERFORMANCE PREDICTIONS FOR SPECIMENS IN CORE POSITION 3E2

Performance predictions for each fuel specimen were made with the aid of the EBR-II Hazards Analysis Code prepared by the late C. R. Reinke, Argonne National Laboratory. As a code was not fully documented prior to Mr. Reinke's death, a description of the input required for the various parts of the computer program can be found in Reference 20.

Since the exact location of each capsule was unknown, the calculations were made on the basis of the maximum power position in the subassembly. Part I of the computer program was employed to determine the heat generation of each type of fuel specimen. Uncertainty factor calculations were made only for the hottest specimen, PNL-X-7. Tabulation of the uncertainty factors are included for both the 45 MW and 56.2 MW power cases at a flow rate of 1.5 gal/min. Stress calculations were made for the hottest specimen for an average burnup of 10,000 MWd/tonne. Part II of the program (stress calculations) also includes calculations for the variation of sodium level, capsule pressure, and void volume with sodium temperature and the variation of sodium level with reactor coolant pressure in a capsule with a leak at its lower end. Text Figures 5, 6, 7, and 13, respectively, show these data.

The axial temperature profiles presented in Text Figures 9 and 10 were determined from the Part I calculations, the uncertainty factor calculations, and hand calculations for the fuel center temperature. Hand calculations for fuel center temperatures were necessary since the code does not contain a routine for treating the variation of fuel thermal conductivity with temperature.

The fuel center temperatures were calculated from the expression

$$\int_0^{T_c} k dt = \int_0^{T_s} k dt + \int_{T_s}^{T_c} k dt$$

where

T_s is the fuel surface temperature as determined from the computer calculations, and

$$\int_{T_s}^{T_c} k dt = \frac{Q}{4\pi}$$

Figure 11 in the text presents the $\int_0^T k dt$ curve for mixed oxide fuel used for these calculations.

Fuel center temperature reduction attendant with center void formation in the fuel was not considered in these calculations. Thus, the estimated temperatures are only representative of start-of-irradiation conditions.

The temperature histories of the hottest specimen under loss-of-reactor coolant condition were calculated with Part IV of the program using power and flow decay input specified in the Guide.⁽¹³⁾ Figure 16 in the text shows these data.

In lieu of the reactor power transient accident specified in the Guide and in accordance with instructions from

from R. Neidner,⁽¹⁸⁾ the steady-state performance of the sub-assembly at a 25% overpower conditions (56.25 MW reactor power) was evaluated. The calculations show that no additional hazards to the reactor are presented by the subassembly under these conditions.

PNL-X-7

PNL-X-7	$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
	Out	In	Max		Max			
45 MW, 1.59 gal/min	Coolant $\Delta T = 101.7$ F D		Film $\Delta T = 17.2$ F D		Cap Wall $1/2 \Delta T = 18.7$ F D		Σd	$(\Sigma d)^2$
Neutron + γ Flux	0.10	10.2	0.10	1.7	0.10	1.9	13.8	190
Flow Rate	0.05	5.1					5.1	26
Flow Profile	0.05	5.1					5.1	26
MFG Deviation								
Cladding Dimension					0.05	0.95	0.95	0.9
U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
235,239 Dimension	0.01	1.0	0.01	0.2	0.01	0.2	1.4	2.0
Cap Wall Thermal Cond.					0.10	1.9	1.9	3.6
Film Coefficient			0.02	0.4			0.4	0.2
Power Level	0.02	2.0	0.02	0.4	0.02	0.4	2.8	7.8
Transient Overload	0.05	5.1	0.05	0.9	0.05	0.9	6.9	48

Uncertainty Factor = $\sqrt{305.1}$ = 17.5

Maximum Capsule
Mean Wall Temp. = 496
+ 17.5
= 513.5 °C
= 957 °F

PNL-X-7	$T_1 - T_1$		$T_2 - T_1$ Max		$T_3 - T_2$ Max			
56.2 MW, 1.59 gal/min	Coolant $\Delta T = 127.1$		Film $\Delta T = 21.3$		Cap Wall $1/2 \Delta T = 23.0$			
	F	D	F	D	F	D	Σd	$(\Sigma D)^2$
Neutron + γ Flux	0.10	12.7	0.10	2.1	0.10	2.3	17.1	294
Flow Rate	0.05	6.4					6.4	42
Flow Profile	0.05	6.4					6.4	42
MFG Deviation								
Cladding Dimension					0.05	1.2	1.2	1.4
U, Pu Dimension			0.02	0.4	0.02	0.5	0.9	0.8
235,239 Dimension	0.01	1.3	0.01	0.2	0.01	0.2	1.7	2.9
Cap Wall Thermal Cond.					0.10	2.3	2.3	5.3
Film Coefficient			0.02	0.4			0.4	0.2
Power Level	0.02	2.6	0.02	0.4	0.02	0.5	3.5	12.3
Transient Overload	0.05	6.4	0.05	1.1	0.05	1.2	8.7	74

Uncertainty Factor = $\sqrt{574.9}$ = 24.0

Maximum Capsule
Mean Wall Temp. = 527.2

574.9
551.2 °C
= 1024 °F

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