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$(\text{Pu}, \text{U})\text{O}_2$ -SODIUM REACTIONS: EFFECT OF FUEL PARAMETERS

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U. S. ATOMIC ENERGY COMMISSION
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(Pu,U)O₂-SODIUM REACTIONS:
EFFECT OF FUEL PARAMETERS

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TABLE OF CONTENTS

ABSTRACT	1-1
1. INTRODUCTION	1-1
2. SUMMARY AND CONCLUSIONS	2-1
3. EXPERIMENT DESCRIPTION	3-1
3.1 Capsule B8A	3-1
3.2 Capsule B8B	3-1
3.3 Capsule B8C	3-1
3.4 Capsule B8D	3-1
4. IRRADIATION HISTORY	4-1
4.1 Capsule B8A (Heat Transfer of Sodium-Bonded versus Gas-Bonded Pins)	4-1
4.2 Capsule B8B (Effect of Stoichiometry)	4-1
4.3 Capsule B8C (Sodium-Fuel Reactivity versus Burnup)	4-1
4.4 Capsule B8D (Sodium-Fuel Reactivity versus O/M, Density and Percent of Pu)	4-2
5. INTERIM AND POST-IRRADIATION EXAMINATION	5-1
5.1 Neutron Radiography	5-1
5.2 Gamma Scans	5-1
5.3 Visual Examination	5-5
5.4 Dimensional Examination	5-5
5.5 Fission Gas Sampling	5-13
5.6 Metallographic Examination	5-14
5.7 Leach Sampling	5-33
5.8 Burnup Analyses	5-33
6. DISCUSSION AND RESULTS	6-1
6.1 Fuel Swelling	6-1
6.2 Thermal Performance of Sodium-Bonded Fuel	6-4
6.3 Fission Gas Release	6-7
6.4 Clad Corrosion	6-7
REFERENCES	R-1
APPENDICES	
A CAPSULE DESIGN DETAILS	A-1
B FUEL PIN DESIGN DETAILS	B-1
C SODIUM BONDING OF FUEL PINS	C-1
D FUEL VOLUME CALCULATIONS	D-1
E THERMAL PERFORMANCE CALCULATIONS	E-1
F FISSION GAS RELEASE CALCULATIONS	F-1
DISTRIBUTION	1

LIST OF ILLUSTRATIONS

Figure	Title	Page
4-1	Capsule B8A Irradiation History	4-2
4-2	Measured Center Temperatures versus Power for Fuel Pin B8B-1	4-3
4-3	Capsule B8B Irradiation History	4-4
4-4	Capsule B8C Irradiation History	4-5
4-5	Capsule B8D Irradiation History	4-6
5-1	Expansion of Fuel Pin B8C-1 Fuel Column with Increasing Burnup	5-2
5-2	Observed Movement of Plug in Center Void of Fuel Pin B8C-2	5-3
5-3	Pin Hole Photograph and Gross Gamma Scan of Capsule B8A Fuel Pins	5-4
5-4	Gross and Single Channel Gamma Scans of Capsule B8B Fuel Pins	5-6
5-5	Beta, Gamma Autoradiograph of Thermocouple Area of Fuel Pin B8B-1	5-7
5-6	Gross Gamma Scan of Capsule B8C Fuel Pins	5-8
5-7	Gross Gamma Scan of Capsule B8D Fuel Pins	5-8
5-8	Visual Appearance of Fuel Pin B8A-2 and Primary Containment	5-9
5-9	Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-1	5-10
5-10	Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-2	5-11
5-11	Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-3	5-12
5-12	"As-Cut" Sections of Fuel Pin B8D-1	5-15
5-13	"As-Cut" Sections of Fuel Pin B8D-3	5-16
5-14	Variation in Width of Reaction Phase in Fuel Pins B8B-1 and B8B-2	5-18
5-15	Observed Variation in Width of Reaction Phase Due to Compositional Differences in the Capsule B8D Fuel Pins	5-19
5-16	Reaction Phase-Fuel Interface in the Low Density Fuel Column of Fuel Pin B8D-2	5-20
5-17	General Appearance of Fuel in Capsule B8A Fuel Pins	5-21
5-18	General Appearance of Fuel in Capsule B8B Fuel Pins	5-22
5-19	General Appearance of Fuel in Capsule B8C Fuel Pins	5-23
5-20	General Appearance of Fuel in Capsule B8D-1 Fuel Pin	5-24

LIST OF ILLUSTRATIONS (CONTINUED)

5-21	General Appearance of Fuel in Capsule B8D-2 Fuel Pin	5-25
5-22	General Appearance of Fuel in Capsule B8D-3 Fuel Pin	5-26
5-23	Porous Layer Surrounding Center Void of Fuel Pin B8C-2	5-27
5-24	Void Migrating from Prior Columnar Grain Boundary in Fuel Pin B8D-1	5-28
5-25	Longitudinal Section Through the Top of Fuel Pin B8C-1 Fuel Column	5-29
5-26	Appearance of Fuel-Plenum Interface in Fuel Pin B8C-1	5-30
5-27	Attack of Fuel Pin B8A-1 Cladding	5-32
5-28	Attack of Fuel Pin B8A-2 Cladding	5-33
5-29	Attack of Screen Wire Sleeve Placed in Gap of Fuel Pin B8C-2	5-34
5-30	Attack of Tungsten-Rhenium Sheath of Central Thermocouple of Fuel Pin B8B-1	5-35
6-1	Center Temperature versus Operating Power Measured for Fuel Pin B8B-1	6-5
6-2	Interface Temperature of Reaction Phase as a Function of Fuel Pin Power	6-5
6-3	Heat Transfer Coefficients versus Fuel-Clad Gap Width	6-6
6-4	Comparison of Fission Gas Release Values for Sodium-Bonded Pins B8C-1, B8C-2, and B8D-1 with Representative Gas-Bonded Fuel Pin Values	6-8
A-1	Radial Cross-Sections Through Capsule B8C	A-2
A-2	Capsule Details	A-3
B-1	Fuel Pin Details	B-2
C-1	Sodium Fill System	C-2
C-2	Sodium Fill System Furnace and Manometer Assembly	C-3
C-3	Appearance of Sodium-Bond in Dummy Fuel Pin	C-4
C-4	Eddy Current Maps of Capsule B8B Fuel Pins	C-5
D-1	Pre- and Post-Irradiation Dimensions for Fuel Pin B8C-1	D-2
D-2	Calculated O/M Profile Through Unreacted Fuel of Fuel Pin B8C-1	D-4
E-1	Thermal Conductivity Data for Mixed Oxide Fuel	E-3
F-1	Bellows Calibration Curve for Fuel Pin B8C-1	F-2

LIST OF TABLES

Table	Title	Page
5-1	Average Clad Diametral Increase Measured for the B8C and B8D Fuel Pins	5-13
5-2	Fission Gas Analysis Results	5-13
5-3	Pre- and Post-Irradiation Gap Size for the B8 Series Fuel Pins (Na Bonded)	5-14
5-4	Plenum Leach Analyses of B8B Fuel Pins	5-36
5-5	Calculated Percent of Total Inventory of each Isotope Found in the Plenum of the B8B Fuel Pins	5-36
5-6	Comparison of Fuel Burnup	5-36
6-1	Measured and Calculated Fuel Volume Changes	6-1
6-2	Average Valence of Fission Product Atoms Generated by Fissioning of Uranium	6-2
6-3	Average Valence of Fission Product Atoms Generated By Fissioning of Plutonium	6-3
6-4	Calculated Values for h_{eff}	6-6
6-5	Calculated Fission Gas Release for Fuel Pins B8C-1, B8C-2, and B8D-1	6-7
6-6	Measured and Calculated Isotopic Fission Gas Yields for Fuel Pin B8C-1	6-8
B-1	Fuel Pin Parameters	B-3

ABSTRACT

A series of eight sodium-bonded pins and one gas bonded pin (mixed-oxide fuel) were irradiated in a thermal flux to investigate the thermal performance of sodium-bonded versus gas-bonded fuel and the effect of fuel stoichiometry, density, plutonium content, and burnup upon sodium-fuel-cladding interactions.

Experimental results indicated that the thermal performance of sodium-bonded fuel is similar to that of gas-bonded fuel. Enhanced fuel volume changes were measured compared to gas-bonded fuel, the rate of change of fuel volume being $\leq 3.7\% \Delta V/V/10,000$ MWd/Te. Sodium fuel reactivity was observed to decrease with decreasing stoichiometry and plutonium content. Reduced clad attack was observed in sodium-bonded pins compared to gas-bonded pins for similar clad temperatures.

1. INTRODUCTION

Basic to the question of failed fuel behavior is the effect of sodium upon the performance characteristics of mixed-oxide fuel. In a previous series of experiments (B2 series)^{1,2}, sodium-bonded mixed-oxide fuel specimens indicated good short-term performance (several hours) up to linear power generation rates of 30 kW/ft. However, failed mixed-oxide fuel elements in contact with sodium or NaK for extended periods of time have shown obvious differences; notably in the area of diametral increases. Failed mixed-oxide fuel elements in contact with sodium or NaK have shown maximum diametral increases in the failed areas of 21³ to 34%.⁴ This compares with diametral increases of a few percent measured for mixed-oxide fuel in the absence of sodium or NaK.

As a result of such obvious differences and the general lack of data concerning the in-pile performance of mixed-oxide fuel in contact with sodium, a series of four capsule experiments containing a total of eight sodium-bonded pins and one gas-bonded pin were irradiated in the General Electric Test Reactor (GETR) to provide information on this subject.* Specific objectives established for the series of tests were: (1) to compare the thermal performance characteristics of sodium-bonded versus gas-bonded fuel and (2) to investigate the effect of the fuel parameters (stoichiometry, density, plutonium content, and burnup) upon sodium-fuel-cladding interactions. This report describes the results of this series of experiments.

* This work was conducted as part of Fast Ceramic Reactor Development Program sponsored by the USAEC under contract AT(04-3)-189, Project Agreement 10 with the General Electric Company.

2. SUMMARY AND CONCLUSIONS

A series of four capsule irradiations were conducted to investigate the thermal performance characteristics of sodium-bonded versus gas-bonded mixed-oxide fuel and to investigate the effect of the fuel parameters (stoichiometry, density, plutonium content, and burnup) upon sodium-fuel-cladding interactions. The four capsules contained a total of eight sodium-bonded pins and one gas-bonded pin. Major fuel and operating parameters for the pins were:

Fuel Pin	Pu/Pu+U	Stoichiometry	Fuel Density	Bond	Cladding	Linear Power (kW/ft)	Burnup (MWd/Te)
B8A-1	0.25	2.00	90	Na	347SS	23	14,670
B8A-2	0.25	2.00	90	He	347SS	23	14,090
B8B-1	0.25	2.00	90	Na	347SS	12.5	13,600
B8B-2	0.25	1.93	90	Na	347SS	11.8	12,860
B8C-1	0.25	1.97	94	Na	316SS	21	129,800
B8C-2	0.25	1.97	94	Na	316SS	22	69,490
B8D-1	0.40	1.97	85 & 94	Na	316SS	18.1	52,740
B8D-2	0.25	1.97	85 & 94	Na	316SS	18.9	53,690
B8D-3	0.25	1.935	85 & 94	Na	316SS	18.1	51,880

In addition, fuel pin B8B-1 contained a central tungsten-rhenium thermocouple in the fuel to measure center temperatures. To study fuel-sodium-cladding reactions, a 200 mesh screen wire sleeve was placed in the fuel-clad gap of fuel pin B8C-2.

Experimental results from the tests indicated:

- (1) No improvement in gap conductance is obtained through sodium-bonding. Heat transfer characteristics of sodium-bonded pins are similar to gas-bonded pins.
- (2) Fission gas release values for sodium-bonded pins are similar to those of thermally irradiated gas-bonded pins.
- (3) Clad attack is reduced in sodium-bonded pins compared to gas-bonded pins for similar clad temperatures.
- (4) Sodium-bonded fuel shows enhanced fuel volume increase compared to gas-bonded fuel due to formation of a low density sodium-fuel reaction product.
- (5) Sodium-fuel reactivity is decreased with decreasing stoichiometry and plutonium content of the fuel.
- (6) The fuel volume increases with increasing burnup.

The relatively large fuel volume changes measured in the present test are due to the high oxygen content in the closed fuel-sodium system inside of each test specimen. The presence of a large oxygen sink, as in a regular reactor system, is expected to result in lower volume changes in fuel operated in the failed condition for longer periods of time in an LMFBR.

3. EXPERIMENT DESCRIPTION

A series of four capsule irradiations were conducted. The capsules were designated B8A, B8B, B8C, and B8D, respectively. A general description of each capsule test is described below. Specific capsule and fuel pin design and fabrication details are discussed in Appendices A and B. Sodium-bonding of the fuel pins is discussed in Appendix C.

3.1 CAPSULE B8A

Capsule B8A was specifically designed to study the heat transfer properties of sodium-bonded fuel versus gas-bonded fuel. The capsule contained two fuel pins fabricated to the same specifications. One pin was sodium-bonded while the other was back filled with helium. Both pins were fabricated with Type 347 stainless steel cladding, 0.250 inch in diameter. Each pin contained 2 inches of mixed-oxide fuel ($\text{UO}_2 - 25 \text{ w/o PuO}_2$) of 2.00 stoichiometry. The sodium-bonded pin contained a central thermocouple to permit direct measurement of the fuel central temperature. However, this thermocouple failed during fabrication so that no direct temperature measurements could be made during irradiation.

3.2 CAPSULE B8B

Capsule B8B was designed to compare directly the performance of two sodium-bonded fuel elements operating at similar powers but differing in stoichiometry. In addition, because of the failure of the central thermocouple during fabrication of capsule B8A, a central thermocouple (tungsten-rhenium) was incorporated in the top fuel pin of the capsule to permit direct measurement of the fuel central temperature. Each fuel pin contained 2 inches of mixed-oxide fuel ($\text{UO}_2 - 25 \text{ w/o PuO}_2$). The top pin contained fuel of 2.00 stoichiometry while the bottom pin contained fuel of 1.93 stoichiometry. Both pins were fabricated with Type 347 stainless steel cladding, 0.250 inches in diameter.

3.3 CAPSULE B8C

The primary objective of the B8C test was to study sodium-fuel interaction with burnup. The capsule contained two sodium-bonded fuel pins each containing approximately 2-1/4 inches of mixed-oxide fuel ($\text{UO}_2 - 25 \text{ w/o PuO}_2$) of 1.97 stoichiometry. The diameter of the two fuel pins was ratioed such that one pin would accumulate approximately double the burnup of the other pin for the same fluence during irradiation. The smaller diameter pin (B8C-1) also incorporated a bellows as part of the plenum area to permit direct measurement of the fission gas release during irradiation. To study sodium-fuel-cladding interactions at higher temperatures, the larger diameter pin (B8C-2) incorporated a sleeve fabricated of 200 mesh screen wire (Type 304, 316, and 321 stainless steel) in the gap between the fuel and the cladding.

Both pins were fabricated with Type 316 stainless steel cladding, 0.170 and 0.250 inch in diameter. Target burnups on the 0.170 and 0.250 inch diameter pins were approximately 100,000 and 50,000 MWd/Te, respectively.

3.4 CAPSULE B8D

Capsule B8D was designed to investigate sodium-fuel reaction as a function of fuel stoichiometry, density, and Pu content of the fuel. The capsule contained three fuel pins. Each pin was fabricated with fuel of two different densities within the same fuel pin separated by an insulator pellet. The top portion of each fuel column (1-1/4 inches long) was fabricated with 85% theoretical dense fuel while the bottom portion (3/4 inches long) was fabricated with 94% theoretical dense fuel. The fuel composition of the three pins was varied in the following manner:

Fuel Pin	O/M	W/O PuO_2
B8D-1	1.97	40
B8D-2	1.97	25
B8D-3	1.93	25

This variation permitted a direct comparison between pins for the effect of stoichiometry and Pu content of the fuel on sodium-fuel reaction. All pins were fabricated with Type 316 stainless steel cladding, 0.250 inch in diameter.

4. IRRADIATION HISTORY

All capsules were irradiated in the General Electric Test Reactor (GETR) pool using a Mark IVA V-RAFT. The V-RAFT assembly permitted both radial and vertical positioning of the capsules with respect to the reactor core during irradiation. Because multipin capsules were being irradiated, vertical adjustment capability was required to permit power matching between pins within the same capsule. In addition, all sodium-bonded pins showed initially some bubbles within the sodium-bond (see Figure C-4). Radial adjustment permitted low power operation to enhance the sodium-bond prior to full power operation.

The irradiation history of each capsule is discussed below.

4.1 CAPSULE B8A (Heat Transfer of Sodium-Bonded versus Gas-Bonded Pins)

The capsule was operated with the fuel pins straddling the vertical neutron flux peak. This permitted close matching of powers between pins. Figure 4-1 shows a daily plot of fuel pin powers for the length of the capsule irradiation. The capsule was operated at a low power (~ 5 -6 kW/ft) for the first 5 days of irradiation to enhance sodium bonding prior to full power operation (20-23 kW/ft). Following the first 5 days of operation during the second cycle of irradiation, a high temperature alarm occurred on the bottom pin (B8A-2) of the capsule. A 60°F temperature increase of the NaK in the inner annulus about 1/2 inch down from the top of the bottom pin was detected. A simultaneous temperature increase of 5 - 10°F was measured in the outer NaK annulus. No significant temperature perturbations were recorded in the thermocouples surrounding the top fuel pin. After the alarm, the capsule V-RAFT was backed full out to reduce the power. After evaluation of this event, the decision was made to continue irradiation of the capsule to the goal burnup ($> 10,000$ MWd/Te). The capsule was inserted slowly with close monitoring to 50% of full power for 4 days. Subsequently, the capsule was operated at approximately 16-17 kW/ft for the remaining portion of the cycle.

4.2 CAPSULE B8B (Effect of Stoichiometry)

Capsule B8B was operated similar to capsule B8A with the fuel pins straddling the vertical neutron flux peak to permit power matching between pins. Prior to irradiation of capsule B8B, fuel pin B8A-1, which contained the nonfunctioning central thermocouple, was examined. The examination showed that severe attack of the thermocouple sheath had occurred. Therefore, capsule B8B fuel pins were operated at medium powers (10.5 to 12.5 kW/ft) to try to extend the central thermocouple lifetime.

The fuel pins were operated at a low power (4 to 5 kW/ft) for 5 days to enhance sodium-bonding prior to full power operation. This was to assure central temperatures truly representative of full sodium-bonding. Figure 4-2 shows the measured center fuel temperature versus average power of the top fuel pin during the life of the central thermocouple. The top fuel pin was power cycled at intervals to observe changes in gap conductivity with time. Prior to each programmed shutdown of the reactor, the fuel pins were slowly reduced in power to reduce thermal shocking of the central thermocouple.

The central thermocouple showed relatively stable operation for the first 390 hours of irradiation. However, in a number of instances during this period, the central temperature measured did not appear to reflect fuel pin power changes (see Figure 4-2). After 390 hours of operation, the thermocouple continued to function but showed somewhat erratic behavior. After 488 hours of operation, the thermocouple stopped functioning. The irradiation of the fuel pins was continued to goal burnup ($> 10,000$ MWd/Te). Figure 4-3 shows a daily plot of fuel pin powers for the length of the capsule irradiation. Fuel pin powers were maintained in the same range as when the central thermocouple functioned, 10.5 to 12.5 kW/ft.

4.3 CAPSULE B8C (Sodium-Fuel Reactivity versus Burnup)

Capsule B8C was positioned on the vertical neutron flux peak similar to the previous capsules. Because of the dissimilar diameters of the two fuel pins, the fuel was preferentially enriched in the smaller diameter pin (B8C-1) so that similar powers would be obtained in both pins for the same flux level. The fuel pins were operated at a low power (~ 8 kW/ft) for 5 days to enhance sodium bonding prior to full power operation. Both fuel pins were operated in the range of 21-22 kW/ft. Figure 4-4 shows a daily plot of fuel pin power for the length of the capsule irradiation.

At selected intervals during irradiation, the capsule was removed from the reactor and neutron radiographs made of the capsule fuel pins. The fuel in both pins was observed to expand axially into the plenum region of the pins.

The bellows assembly incorporated in the plenum of fuel pin B8C-1, to measure interim fission gas release values, functioned only during the early portion of the capsule irradiation. At the first interim examination of the capsule, the bellows was observed to have compressed slightly as expected indicating increased pressure in the plenum due to fission gas release. However, at the second interim examination of the capsule, the bellows was no longer compressed. Failure of the bellows had occurred sometime between the two examinations.

4.4 CAPSULE B8D (Sodium-Fuel Reactivity versus O/M, Density and Percent of Pu)

Capsule B8D contained three fuel pins. Power matching between fuel pins was achieved by preferentially enriching the fuel in the end pins of the capsule compared to the center pin. By placing the capsule symmetrically on the flux peak, similar powers could be achieved at higher rod banks (i.e., when the flux peak is flattened) during irradiation. During capsule irradiation, powers between the end pins and the middle pin were normally matched within 2 kW/ft or less.

At the start of the capsule irradiation, the fuel pins were operated at a low power (~ 7 -8 kW/ft) for 5 days to enhance sodium bonding. At full power operation the fuel pins operated at 17-18 kW/ft.

Similar to capsule B8C, interim neutron radiographs were taken of the capsule fuel pins. The low density portion of the fuel column of each pin showed expansion into the plenum region. Only minor changes in length were noted for the high density fuel. During the interim examination of the fuel pins, no center void formation could be detected in the end pins due to the high enrichment of the fuel. To insure formation of a center void and columnar grain growth in the pins (to facilitate comparison between pins), the fuel pin powers were increased to 19 kW/ft during the last cycle of the irradiation. Figure 4-5 shows a daily plot of fuel pin powers for the length of the capsule irradiation.

Figure 4-1. Capsule B8A Irradiation History

Figure 4-2. Measured Center Temperatures versus Power for Fuel Pin B8B-1

Figure 4-3. Capsule B8B Irradiation History

Figure 4-4. Capsule B8C Irradiation History

Figure 4-5. Capsule B8D Irradiation History

5. INTERIM AND POST-IRRADIATION EXAMINATION

Following irradiation, the four capsules were disassembled and the fuel pins destructively examined at the General Electric Radioactive Materials Laboratory. The examination of the individual pins generally consisted of the following:

- (1) Neutron radiography and gamma scan of fuel pin length.
- (2) Macroscopic examination of the cladding for corrosive or structural damage.
- (3) Profilometry measurements of fuel pin cladding.
- (4) Puncturing of fuel pin plenum to determine amount and analysis of gases present.
- (5) Detailed metallographic examination of fuel and cladding, including autoradiography.
- (6) Radiochemical analysis to determine fuel burnup.

In addition, interim radiography was performed on capsule B8C and B8D fuel pins. Results of these interim examinations and the post-irradiation examination of the fuel pins are described below:

5.1 NEUTRON RADIOGRAPHY

Neutron radiographs were obtained for all fuel pins except B8A-1 and B8A-2. (The B8A pins were not neutron radiographed because the technique was not developed at the time of the B8A examination.)

Interim and post-irradiation neutron radiographs showed the remaining fuel pins to be sound and intact. A significant feature emphasized by the interim neutrography of the B8C and B8D fuel pins was the mobile nature of the fuel within the pins during irradiation.

Both of the sodium-bonded B8C fuel pins showed substantial movement of the fuel into the plenum region of the pins. The fuel columns were not axially restrained. Figure 5-1 illustrates the axial fuel expansion observed in fuel pin B8C-1. At 129,800 MWd/Te burnup, fuel can be detected up to 0.377 inch above the original top pellet surface. This corresponds to a fuel column length change of 16.7%.

Similar to the B8C fuel pins, the B8D pins also showed significant axial expansion of the fuel into the plenum region. However, the axial expansion was confined to the low density portion of the fuel column in each pin (fuel axially unrestrained). Only minor changes in length were measured for the high density fuel columns. The axial expansion of the low density fuel column varied with composition and stoichiometry. Fuel pin B8D-1, containing mixed-oxide fuel of 40 w/o PuO_2 and of 1.97 stoichiometry, showed the greatest expansion. For the remaining two pins, fuel pin B8D-3 with mixed-oxide fuel of 25 w/o PuO_2 and of 1.935 stoichiometry showed the least expansion while pin B8D-2 with mixed-oxide fuel of 25 w/o PuO_2 and 1.97 stoichiometry showed expansion slightly less than that of fuel pin B8D-1. Figures 5-9, 5-10, and 5-11 show the post-irradiation appearance of the fuel in the B8D pins.

The B8B fuel pins also showed only minor axial expansion of the fuel. These pins contained a top insulator pellet restrained from axial movement by a holddown tube in the plenum. The axial expansion of the fuel in these pins was confined to filling the thermal expansion gap left in the pins to accommodate the fuel swelling on initial rise to power during irradiation. The total fuel swelling of the B8C and B8D fuel pins is discussed in Section 6.1.

Neutron radiography revealed additional movement within the fuel of several pins besides that of axial expansion. In fuel pins B8C-2 and B8D-2 (high density fuel column), material movement into the center void of the pins could be observed subsequent to the formation of the central void. In fuel pin B8C-2, this material was observed as a plug which varied in position at each examination. The plug exhibited a convex, spherical upper surface in each case. This, in conjunction with the extreme mobility of the plug, possibly indicate the plug material was molten during irradiation. The central void was reduced in diameter below the plug in each case. Figure 5-2 shows the appearance of the plug and growth of the central void in fuel pin B8B-2. The center void of fuel pin B8C-1 also showed a number of plugs and the central void diameter varied considerably along the fuel column length.

Neutrography showed no center void was formed in fuel pin B8B-2 (O/M 1.93) while a center void was formed in fuel pin B8B-1 (O/M 2.00) for maximum powers of 11.8 and 12.5 kW/ft, respectively, during irradiation.

Toward the top of the low density fuel column in fuel pin B8D-1, an annular appearing space was formed in the fuel. This space appeared to be stable as it was observed in both the interim and final neutron radiographs of the pin.

5.2 GAMMA SCANS

The gross gamma scan of capsule B8A indicated that the bottom (gas-bonded) fuel pin failed during irradiation. Failure is presumed to have occurred at the time of the high temperature alarm detected on the bottom pin during its second cycle of irradiation. Figure 5-3 shows a gross gamma scan of the capsule and pin hole photographs showing gamma activity along the length of the capsule. In the figure, it can be seen that displaced fuel and fission products are

Figure 5-1. Expansion of Fuel Pin B8C-1 Fuel Column with Increasing Burnup

BURNUP
MWd/Te

PLUG

69,500

PLUG

59,600

PLUG

37,600

18,400

Figure 5-2. Observed Movement of Plug in Center Void of Fuel Pin B8C-2

COMPOSITE PHOTOGRAPH
OF γ ACTIVITY

DISPLACED FUEL
AND FISSION PRODUCTS

B8A-1

B8A-2

Figure 5-3. Pin Hole Photograph and Gross Gamma Scan of Capsule B8A Fuel Pins

present on various horizontal surfaces below the bottom fuel pin. Breach of the secondary containment around the fuel pin is indicated by the presence of fuel and fission products in the very bottom of the capsule. The gamma scan shows the top sodium-bonded pin to be sound and intact since no extraneous activity was detected.

A gross gamma activity scan of the B8B fuel pins showed the activity in the top pin (B8B-1) to be perturbed due to the presence of the thermocouple, Figure 5-4. For the top pin, the dip in the scan at the middle of the fuel column appears to be associated with a slight separation of the fuel column between the third and fourth pellets from the top. This separation was detected in the neutron radiographs of the pin. The dip and peak at the top of the fuel column appear to be due to the migration of fission products from the fuel through a breach in the thermocouple sheath to the level of the fuel-insulator interface. This fission product activity can be observed in the beta-gamma autoradiograph of the thermocouple area of the pin shown in Figure 5-5. Single channel scans of the fuel pins were also performed for selected fission products to determine if the fission product behavior was similar to that observed for gas-bonded fuel. The scans showed I-131 to be frequently concentrated at the fuel pellet interfaces. Conversely, BaLa-140 frequently showed a reduced concentration at the pellet interfaces (see Figure 5-4). Concentration of iodine at pellet interfaces has been monitored in gas-bonded irradiations,⁵ however, depletion of the BaLa-140 at the pellet interfaces appears to be unique to sodium-bonded fuel.

Figures 5-6 and 5-7 show gross gamma scans of capsules B8C and B8D. Pronounced activity peaks can be observed at the top and/or bottom of the fuel columns in the different pins, as well as numerous smaller peaks along the fuel column length. Multichannel scanning identified ruthenium as the source of the major activity in these areas. Subsequent metallographic examination of fuel pin B8C-1 showed a large concentration of metallic fission products in the region of the gamma peak at the top of the fuel column, as well as the presence of metallic ingots in the center void. Since ruthenium is a prime gamma emitter found in such ingots,⁶ one may infer that similar concentration of metallic fission products has occurred in the other pins. For fuel pin B8C-2, the central plug region previously described is clearly delineated on the gross gamma scan by the increased activity associated with it.

5.3 VISUAL EXAMINATION

Figure 5-8 shows the appearance of fuel pin B8A-2 upon removal from the capsule. The fuel pin showed complete melting of the cladding around the circumference of the pin in the area of the top fuel pellet. During failure, molten fuel contacted the primary containment as evidenced by the burned through area adjacent to the failed region of the pin. Failure of the pin is attributed to the small radial gap, ~ 0.0028 in., and axial gap, 0.010 in., built into the pin coupled with gross attack of the cladding (see Section 6.4) which resulted in sufficient strain and degradation of cladding properties to cause failure. This permitted escape of some fission gas from the pin which was trapped in the upper portion of the secondary containment and resulted in blanketing the top of the pin. Thermal analysis calculations indicate blanketing of 0.05 – 0.10 in. of the upper edge of the top-fuel pellet would be sufficient to cause failure of the pin. The calculations assumed complete blanketing of the circumference of the cladding and that failure occurred when the cladding melted. The cladding of all other pins showed no change during irradiation.

5.4 DIMENSIONAL EXAMINATION

Post-irradiation micrometer and/or profilometer traces of the fuel pins showed that clad diametral changes occurred for the higher burnup pins. The low burnup pins B8A-1, B8B-1, and B8B-2 showed no diametral change. The unfailed portion of fuel pin B8A-2 showed an average diametral increase of 4%. The B8A-2 fuel pin exhibited the greatest diametral increase at the failed end of the pin (6.5%). These dimensional changes are presumed to have occurred after failure due to sodium-fuel reaction.

Both B8C fuel pins showed a relatively uniform clad diametral increase along the length of the fuel. However, the high and low density fuel columns in the B8D pins showed significantly different diametral increases. Figures 5-9, 5-10, and 5-11 show the pre- and post-irradiation profilometry traces for the B8D fuel pins. The difference in performance of the two columns would appear to reflect the degree of axial restraint to swelling imposed on each fuel column. The top low density column was subject to the least restraint so that the fuel was more free to expand axially. The swelling that occurred in the low density column, however, tended to restrain the high density column so that the major dimensional changes in the high density column were radial rather than axial. Table 5-1 summarizes the average diametral increases measured for the B8C and B8D fuel pins.

The extreme ridging of the cladding observed in the low density fuel column of fuel pin B8D-1 occurred adjacent to the annular-appearing space observed to have formed in the fuel. This ridging suggests that increased sodium-fuel reaction occurred in this area of the pin.

Figure 5-4. Gross and Single Channel Gamma Scans of Capsule B8B Fuel Pins

Figure 5-5. Beta, Gamma Autoradiograph of Thermocouple Area of Fuel Pin B8B-1

<u>POSITION</u>	<u>MAJOR ACTIVITY</u>	<u>ISOTOPES OBSERVED</u>
1	Ru-106	NbZr-95, Ru-103, Ru-106 Cs-137, BaLa-140, CePr-144
2	Cs-137	Cs-134, Cs-137
3	—	NbZr-95, Ru-106, Cs-137 BaLa-140, CePr-144

Figure 5-6. Gross Gamma Scan of Capsule B8C Fuel Pins

<u>POSITION</u>	<u>MAJOR ACTIVITY</u>	<u>ISOTOPES OBSERVED</u>
1	Ru-106	NbZr-95, Ru-103, Ru-106 Cs-137, BaLa-140, CePr-144
2	—	NbZr-95, Ru-103, Ru-106 Cs-137, BaLa-140, CePr-144

Figure 5-7. Gross Gamma Scan of Capsule B8D Fuel Pins

FUEL PIN B8A-2

1-1/2X

PRIMARY CONTAINMENT

1-1/2X

PRIOR TO CLEANING

1-1/2X

AFTER CLEANING

Figure 5-8. Visual Appearance of Fuel Pin B8A-2 and Primary Containment

Figure 5-9. Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-1

POST-IRRADIATION NEUTRON RADIOGRAPH

94

85

INITIAL FUEL DENSITY (%T.D.)

Figure 5-10. Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-2

Figure 5-11. Pre- and Post-Irradiation Profilometry Traces of Fuel Pin B8D-3

Table 5-1
AVERAGE CLAD DIAMETRAL INCREASE MEASURED
FOR THE B8C AND B8D FUEL PINS

Fuel Pin	Average Diameter Increase, (inches)	
	85% T.D.	94% T.D.
B8C-1	-----	0.0014
B8C-2	-----	0.0007
B8D-1	0.0008	0.0034
B8D-2	0.0008	0.0018
B8D-3	0.0007	0.0013

5.5 FISSION GAS SAMPLING

Using standard procedures, the plenums of all pins except B8A-2, which failed, were sampled for their fission gas content. Table 5-2 lists the volume of fission gas obtained from each fuel pin and summarizes the results of the analysis of each sample. Helium was used as the fill gas in all pins so that its presence in the fission gas samples is to be expected. The high nitrogen and oxygen content of the samples obtained from fuel pins B8A-1, B8B-1, and B8B-2 indicate that leakage occurred during sampling. In addition, the smaller volumes of fission gas recovered from fuel pins B8D-2 and B8D-3 compared to fuel pin B8D-1 indicate that leakage probably occurred in these pins also during sampling. The B8D pins operated at similar powers with roughly comparable center temperatures, therefore, similar fission gas release values would be expected. For the above reasons, only the fission gas samples obtained from fuel pins B8C-1, B8C-2, and B8D-1 are considered representative of the true fission gas release values. Fission gas release values calculated for the above pins are discussed in Section 6.3.

Table 5-2
FISSION GAS ANALYSIS RESULTS
Volume %

Species	B8A-1	B8B-1	B8B-2	B8C-1	B8C-2	B8D-1	B8D-2	B8D-3
Helium	5.0	ND	1.8	1.72	3.07	2.45	1.83	2.97
Xenon	29.0	12.9	15.8	92.0	90.1	90.3	78.0	90.7
Krypton	2.5	0.78	1.57	8.70	7.28	8.53	3.62	9.46
Argon	ND	ND	ND	T	ND	T	ND	ND
Nitrogen	50.0	76.4	35.0	1.45	0.25	0.42	5.55	0.65
Oxygen	3.0	5.69	27.2	0.4	ND	T	1.81	T
Hydrogen	11.0	2.47	17.7	0.74	ND	T	7.97	ND
Carbon Monoxide	2.0	ND	ND	T	ND	ND	ND	ND
Carbon Dioxide	NA	ND	NA	0.05	0.04	0.08	0.35	0.06
Methane	NA	ND	T	ND	ND	ND	ND	ND
Total Volume of Gas Sample (CC STP)	1.58	1.53	0.36	18.8	18.7	11.2	0.37	6.57

ND = NDT Detected

NA = Not Analyzed

T = Trace

5.6 METALLOGRAPHIC EXAMINATION

5.6.1 Fuel

Each pin was transversely sectioned at selected locations along its length. During sectioning, observation of the fuel was difficult because of reactions occurring within the pins with oxygen and moisture from the air. This reaction product tended to coat the fuel surfaces and obscure details. Figure 5-12 shows the appearance of the fuel during sectioning for fuel pin B8D-1. The weeping observed for this pin is typical of all the sodium-bonded pins.

In agreement with neutron radiography, material was observed to fill or partially fill the center voids in a number of pins (B8C-1, B8C-2, B8D-2, and B8D-3). Macroscopically this material appeared to be similar in appearance to the fuel. Figure 5-13 shows the appearance of the material filling the center voids in the high and low density fuel columns of fuel pin B8D-3. Free sodium could be observed in the plenum of all sodium-bonded pins.

During preparation of samples for microscopic examination, kerosene was used as the lubricant during grinding operations rather than water to try to retain any sodium reaction products present. The samples were stored in kerosene when not being examined. For examination, the kerosene was cleaned from the surfaces by wiping with a small amount of ethyl alcohol. Repeated cleaning of the surfaces tended to remove any reaction phase present because of its reactivity with the alcohol. During the examination of the B8C and B8D pins, a modified procedure was used. Xylene was used to clean the surfaces of the specimens instead of alcohol. This method appeared to retain the reaction phase better.

All sodium-bonded pins showed a grey phase at the surface of the fuel. This phase in all pins was found to be extremely reactive. Table 5-3 summarizes the pre- and post-irradiation diametral gaps for each pin.

Table 5-3
PRE- AND POST-IRRADIATION GAP
SIZE FOR THE B8 SERIES FUEL PINS (Na BONDED)

Fuel Pin	Fuel		Gap Size (Diametral)	
	W/O Pu	O/M	Pre-Irradiation (in.)	Average Post-Irradiation (in.)
B8A-1	25	2.00	0.0032	0.0068
B8B-1	25	2.00	0.0106	0.0157
B8B-2	25	1.93	0.0106	0.0072
B8C-1	25	1.97	0.0059	0.0114
B8C-2	25	1.97	0.0166	0.0115
B8D-1 LD ^(a)	40	1.97	0.0054	0.0099
B8D-1 HD ^(b)	40	1.97	0.0053	0.0089
B8D-2 LD	25	1.97	0.0060	0.0181
B8D-2 HD	25	1.97	0.0064	0.0079
B8D-2 LD	25	1.935	0.0067	0.0067
B8D-2 HD	25	1.935	0.0055	0.0046

(a) LD = Low Density (85% T.D.) Fuel Column

(b) HD = High Density (94% T.D.) Fuel Column

Figure 5-12. "As-Cut" Sections of Fuel Pin B8D-1

Figure 5-13. "As-Cut" Sections of Fuel Pin B8D-3

No general correlation was found between post-irradiation gap size and the fuel parameters and operating conditions. However, within individual capsules some correlation with stoichiometry and plutonium content of the fuel appeared to exist.

Figure 5-14 shows the appearance of the gap area in fuel pins B8B-1 and B8B-2. The gap in fuel pin B8B-1 was approximately twice the width of that in fuel pin B8B-2. The original stoichiometry of the fuel in pin B8B-1 was 2.00 while that in pin B8B-2 was 1.93. Similarly, in capsule B8D, the width of the grey phase present in the pins varied with the original stoichiometry or plutonium content of the fuel (high density portion). The average grey phase width was reduced as the plutonium content and stoichiometry decreased. Figure 5-15 shows the appearance of the grey phase in the high density portion of the fuel column in each pin. The fuel pin with the lowest stoichiometry again showed a substantially smaller gap size. In fuel pins B8D-1 and B8D-2 (high density portion), the observed increase in gap size corresponded very closely to the observed average clad diametral increase, 0.00365 versus 0.0034 inch and 0.0015 versus 0.0018 inch, respectively. In the B8C and the low density portion of the fuel column in the B8D pins, no such correlation appeared to exist. As previously discussed, substantial axial movement of the fuel was observed in the B8C fuel pins and the low density portion of the fuel column in the B8D pins. It would appear that the axial movement of the fuel causes other parameters to be controlling.

Growth in width of the grey phase appeared to proceed initially along the grain boundaries with subsequent reaction of the remaining grain. In the low density portion of the fuel column in fuel pin B8D-2 (greatest gap width increase of series), this process can be observed, Figure 5-16.

During the preparation of the metallographic samples, it was found that the material observed during sectioning (and neutron radiographs) to fill portions of the center void in a number of pins was very fragile and could not be retained. Because of this, no positive identification of the material could be made. However, oxide plugs of fission products such as Sr, Ba, Zr, and Si have been observed to form in the center void of gas-bonded pins.⁷ Gamma scanning showed increased activity to be associated with the plug material in fuel pin B8C-2 (see Figure 5-6). Possibly a similar process occurred in the present pins.

Figures 5-17 to 5-22* show the general appearance of the fuel in the different pins. Gross attack of the cladding occurred in fuel pin B8A-2. Fuel pin B8A-2 also showed a significantly larger center void than fuel pin B8A-1. Some slumping of the fuel at the surface of the center void did occur. This slumping is thought to be due to movement of molten fuel into the center void during fuel pin failure, causing localized hot spots. Molten fuel could be observed in the center void of the pin adjacent to the failed area.

In agreement with neutron radiography, metallographic examination showed formation of a center void in the fuel of fuel pin B8B-1. In fuel pin B8B-2 no center void formation was observed.

Both B8C fuel pins showed a spongy, porous fuel layer surrounding the center void. The spongy layers showed a generally dendritic structure. Figure 5-23 shows the appearance of the spongy layer in fuel pin B8C-2. Both pins exhibited very pronounced columnar grain growth. The screen wire sleeve placed in the fuel-clad gap of fuel pin B8C-2 can be clearly observed in Figure 5-19.

The high and low density fuel columns of the B8D pins exhibited microstructural differences. The low density columns showed many narrow columnar grains at the center of the fuel. The high density fuel columns showed a quite different structure. In fuel pin B8D-1, no grain boundaries could be distinguished in the columnar grain growth region; although elongated voids appeared to outline what were previously grain boundaries. Figure 5-24 shows the appearance of one of these elongated bubbles. The larger elongated void appeared to be breaking up into smaller voids and migrating to the center of the fuel. Fuel pin B8D-3 showed the same structure. In addition, a high degree of porosity could be observed throughout the columnar grain growth region. The low density columns in the fuel pins B8D-1 and B8D-3 also showed significant increases in porosity near the center void of the fuel. The high density fuel column in fuel pin B8D-2 showed very wide columnar grains indicating considerable grain boundary migration occurred.

Figure 5-25 shows a longitudinal section through the top of the B8C-1 fuel column. The β - γ radiographs show that significant fission product migration to the top of the fuel column occurred. The α -autoradiograph shows the discontinuous nature of the fuel pin center void. Figure 5-26 shows a higher magnification photograph of the top of the fuel column. It can be seen that the fuel was sufficiently plastic to squeeze up through the holddown spring situated in

* The flat surface on the outside of the cladding of the section from pin B8C-1, Figure 5-19, is an orientation mark applied during post-irradiation examination.

FUEL PIN B8B-1 (INITIAL O/M = 2.00)

50X

FUEL PIN B8B-2 (INITIAL O/M = 1.93)

50X

Figure 5-14. Variation in Width of Reaction Phase in Fuel Pins B8B-1 and B8B-2

Figure 5-15. Observed Variation in Width of Reaction Phase Due to Compositional Differences in the Capsule B8D Fuel Pins

Figure 5-16. Reaction Phase-Fuel Interface in the Low Density Fuel Column of Fuel Pin B8D-2

Figure 5-17. General Appearance of Fuel in Capsule B8A Fuel Pins

B8A-1

SODIUM BONDED PIN 2.00 O/M

30X

B8A-2

GAS BONDED PIN 2.00 O/M

30X

Figure 5-17. General Appearance of Fuel in Capsule B8A Fuel Pins

B8B-1

2.00 O/M

30X

B8B-2

1.93 O/M

30X

Figure 5-18. General Appearance of Fuel in Capsule B8B Fuel Pins

10X

B8C-1

10X

B8C-2

Figure 5-19. General Appearance of Fuel in Capsule B8C Fuel Pins

85% DENSITY

10X

94% DENSITY

10X

Figure 5-20. General Appearance of Fuel in Capsule B8D-1 Fuel Pin

85% DENSITY

10X

94% DENSITY

10X

Figure 5-21. General Appearance of Fuel in Capsule B8D-2 Fuel Pin

85% DENSITY

10X

94% DENSITY

10X

Figure 5-22. General Appearance of Fuel in Capsule B8D-3 Fuel Pin

Figure 5-23. Porous Layer Surrounding Center Void of Fuel Pin B8C-2

Figure 5-24. Void Migrating from Prior Columnar Grain Boundary in Fuel Pin B8D-1

Figure 5-25. Longitudinal Section Through the Top of Fuel Pin B8C-1 Fuel Column

Figure 5-26. Appearance of Fuel-Penum Interface in Fuel Pin B8C-1

the plenum. A number of metallic ingots can be observed in the plenum region. Some ingots appear to have reacted with the holddown spring. In Figure 5-25 several small metallic ingots can also be observed to be located in the center void area.

No unusual activity within the fuel of the different pins could be detected in the α and β - γ autoradiographs. Typically the β - γ autoradiographs showed the major fission product activity to be present at the periphery of the fuel reflecting the severe flux depression within the pins due to self shielding in a thermal flux.

5.6.2 Cladding

Only fuel pins B8A-1 and B8A-2 showed reaction of the fuel with the cladding. The clad temperatures for the B8A pins were the highest of the test series. Maximum inside surface clad temperatures for the B8A pins were in the range 1250°F. The remaining pins were operated with maximum inside surface clad temperatures 50 to 100 degrees below this temperature. Figures 5-27 and 5-28 show the appearance of the clad attack in both B8A pins. The attack in fuel pin B8A-1 varied in thickness from 0.0002 to 0.0007 in. Fuel pin B8A-2 showed more cladding attack. Attack of up to 0.027 in. in depth was measured.

The appearance of the attack varied between the two pins. In fuel pin B8A-1, an oxide appearing layer with some metallic inclusions lies at the surface of the cladding. Overlaying the oxide layer is a continuous metallic layer. In fuel pin B8A-2, general attack of the clad surface occurs around the periphery. However, pockets of gross attack occur at intervals. In many cases finger like projections of metal extend from the pockets of attack into adjacent cracks in the fuel. Intergranular attack of the cladding was observed at the bottom of the pockets of attack. Similar type of attack has been observed in fuel pins in which the clad inside surface temperature was in the same range or above that of the B8A fuel pins.⁴

Examination of the wires composing the sleeve placed in the fuel-clad gap of fuel pin B8C-2 also showed evidence of attack. The cladding of fuel pin B8C-2 was operated with a maximum inside surface clad temperature of 1200°F. Unlike the wires of the sleeve no evidence of attack was observed. The sleeve also contained Type 316 stainless steel, the same as the cladding. The difference in behavior between the 316SS in the two locations indicates the inside surface of the cladding was below the temperature necessary to produce observable attack. Thermal analysis indicates the wires of the sleeve operated up to 480°F above the inside surface temperature of the cladding.

The sleeve was fabricated from three panels of 200 mesh screen wire composed of Types 304, 316 and 321 stainless steel. Marker wires were placed on each panel of the sleeve to permit identification of the panel material. Figure 5-29 shows the representative appearance of the wires in each panel of the sleeve. In many cases the wires have metallurgically bonded to the surface of the cladding. Extensive pitting as well as subsurface attack of the wires can be observed. No contiguous oxide layer was observed at the surface of the wires similar to that of the clad attack in fuel pin B8A-1. No difference in the degree of attack of the three materials could be readily distinguished. The extent of the attack in any area appeared to be a function of its location within the gap and the resulting temperature. The higher the temperature, the greater the attack.

5.6.3 Thermocouples

Metallographic examination of the central thermocouples from fuel pins B8A-1 and B8B-1 showed gross attack of the sheath and thermoelements. Both thermocouple sheaths were breached in essentially the same area. The breach in each case occurred at the top pellet of the fuel column. The central thermocouple from fuel pin B8A-1, which operated at the higher power of the two pins, showed complete separation of the sheath around its circumference. The center annulus between the thermocouple and fuel was completely closed in both pins. The B8A-1 central thermocouple showed gross distortion of the separated portion. A net movement upward of this portion could be detected by the folded over and crumpled configuration of the thermocouple sheath in the breached area.

Figure 5-30 shows the appearance of the sheath attack in the breached area and junction of the B8B-1 central thermocouple. In addition to the general type of attack shown in the two views, attack could be observed to penetrate the sheath wall along grain boundaries.

A number of experiments have been performed using tungsten-rhenium thermocouples to measure fuel center temperatures for gas-bonded pins without manifesting gross deterioration of the sheaths such as the present tests.^{8,9} However, attack such as observed in fuel pins B8A-1 and B8B-1 has been observed when a ready source of oxygen was available.¹⁰ The presence of sodium in the fuel pin may, therefore, contribute to the movement of oxygen to the thermocouple sheath.

Figure 5-27. Attack of Fuel Pin B8A-1 Cladding

5.7 LEACH SAMPLING

To determine the extent of fission product migration to the plenum in the presence of sodium, the plenums of the B8B pins were sectioned just above the top insulator pellet and leached with nitric acid. The solutions were then analyzed for fission products. Table 5-4 shows the results of the analysis.

Using the tables of Blomeke and Todd,¹¹ the total inventory generated in the fuel for each of the fission products identified in the leach samples was calculated. The fission yield of plutonium was considered to be the same as that of uranium. This introduces some error but for the present comparative purposes, the values calculated are sufficiently accurate. Table 5-5 shows the percent of the total inventory of each isotope found in the plenum of each pin. The cesium has been combined for the two pins because some leakage into the sample container occurred during movement of the samples to the laboratory for analysis. Only leakage of cesium could be detected, the other isotopes being below the level of detection.

Except for cesium, little movement of fission products to the plenum region occurred. The large disparity in observed inventory of the two isotopes of cesium may possibly be explained in terms of their precursors. For Cs-134, one precursor is Xe-133 which decays with a half-life of 5.27 days to stable Cs-133. The Cs-133 in turn is neutron activated to Cs-134. For Cs-137, its immediate precursor is Xe-137 which decays with a half-life of 3.8 minutes to Cs-137. The longer half-life would permit a larger amount of the Xe-133 to migrate to the plenum prior to formation of Cs-134. This indicates the cesium was retained locally in the region of the pin in which it was produced and that little axial migration of cesium to the plenum occurred.

5.8 BURNUP ANALYSES

Samples from selected pins were submitted for burnup determinations to provide a check on fuel pin powers during irradiation. Samples were selected from the center region of the fuel columns. Because of the very short length of the fuel columns in the B8D pins, which necessitated use of most of the column for metallographic examination, a sample could not be selected from the mid-plane of the fuel column. To provide some check on measured powers a sample from the bottom end of the high density fuel column of fuel pin B8D-1 was submitted for analysis. Use of an end sample such as this for burnup analysis will include end effects so that the analyzed value can only be compared approximately with the calorimetric value. Table 5-6 shows the results of the burnup analyses.

Good agreement with calorimetric and physics values was obtained for most pins. The analyzed burnup value for fuel pin B8C-2 showed a wide variance with the physics and calorimetric values. The cause for this low value was not determined.

Figure 5-28. Attack of Fuel Pin B8A-2 Cladding

100X

304 STAINLESS STEEL WIRE

316 STAINLESS STEEL WIRE

321 STAINLESS STEEL WIRE

250X

Figure 5-29. Attack of Screen Wire Sleeve Placed in Gap of Fuel Pin B8C-2

Figure 5-30. Attack of Tungsten-Rhenium Sheath of Central Thermocouple of Fuel Pin B8B-1

Table 5-4
PLENUM LEACH ANALYSES OF B8B FUEL PINS

Isotope (Total Atoms)	B8B-1	B8B-2	Leakage	Blank
Cs-137	1.107 X 10 ¹⁷	1.001 X 10 ¹⁸	1.097 X 10 ¹⁸	1.32 X 10 ¹⁷
Cs-134	1.043 X 10 ¹⁵	6.84 X 10 ¹⁵	7.391 X 10 ¹⁵	9.022 X 10 ¹⁴
Ce-144	1.748 X 10 ¹³	2.360 X 10 ¹³		5.868 X 10 ¹²
Ce-141	1.77 X 10 ¹³	1.313 X 10 ¹³		5.88 X 10 ¹²
Sb-125	3.78 X 10 ¹⁴	5.532 X 10 ¹³		5.88 X 10 ¹²
Ru-103	0.925 X 10 ¹³	0.6206 X 10 ¹³		0.265 X 10 ¹³
Ru-106	0.5224 X 10 ¹³	1.000 X 10 ¹³		0.2636 X 10 ¹³
Zr-95	0.53304 X 10 ¹³	0.9013 X 10 ¹³		0.545 X 10 ¹³

Table 5-5
CALCULATED PERCENT OF TOTAL INVENTORY
OF EACH ISOTOPE FOUND IN THE PLENUM
OF THE B8B FUEL PINS

Isotope	Percent of Total Inventory	
	B8B-1	B8B-2
Cs-134		90.0
Cs-137		5.0
Ce-141	1.22 X 10 ⁻⁴	7.78 X 10 ⁻⁵
Ce-144	6.17 X 10 ⁻⁵	9.71 X 10 ⁻⁵
Sb-125	5.43 X 10 ⁻²	6.53 X 10 ⁻²
Ru-103	1.20 X 10 ⁻⁴	6.75 X 10 ⁻⁵
Ru-106	1.98 X 10 ⁻⁴	5.89 X 10 ⁻⁴
Zr-95	—	2.44 X 10 ⁻⁵

Table 5-6
COMPARISON OF FUEL BURNUP

Fuel Pin	Burnup (MWd/Te)			Average Burnup
	Calorimetry	Physics	Analysis	
B8A-1	14,490	14,420	15,100	14,670
B8A-2	14,520	13,670	—	14,090
B8B-1	13,690	13,310	13,800	13,600
B8B-2	12,730	12,750	13,100	12,860
B8C-1	128,700	129,970	130,280	129,800
B8C-2	69,820	69,160	53,300	69,490
B8D-1	51,980	53,910	52,330 ^(a)	52,740
B8D-2	52,900	54,470	—	53,690
B8D-3	51,500	52,250	—	51,880

All burnup values normalized to 205 MeV/fission.

(a) Corrected for end effect by reducing measured value by the ratio:

$$\frac{\text{Area of End of Sample}}{\text{Circumferential Area of Sample} + \text{Area of End of Sample}}$$

6. DISCUSSION AND RESULTS

6.1 FUEL SWELLING

Column one of Table 6-1 shows the total fuel volume* increase measured for each of the B8C and B8D fuel pins. Volume increase values shown include that due to solid and gaseous fission product swelling of the fuel in addition to that due to sodium-fuel reaction. Using the data of Nelson, et al.,¹² the contribution to the total measured fuel volume increase, due to retention of solid and gaseous fission products within the fuel, may be estimated.

The total % ΔV minus the % ΔV calculated to be due to the retention of fission products is attributed to sodium fuel reactions. It is listed in column 3 of Table 6-1.

Table 6-1
MEASURED AND CALCULATED FUEL VOLUME CHANGES

Fuel Pin	Total Fuel Volume Change (%)	Volume Change Due to Solid and Gaseous Fission Products (%)	Volume Change Attributed to Sodium-Fuel Reaction (%)	O/M Shift of Fuel		$(\Delta V/V)$ 0.01 O/M Shift
				Radial ($\times 10^2$)	Burnup ($\times 10^2$)	
B8C-1	22.3	4.3	18.0	2.5	6.3	2.05
B8C-2	25.6	1.3	24.3	2.1	4.3	3.80
B8D-1 LD ^(a)	25.4	1.6	23.8	6.0	3.1	2.62
B8D-1 HD ^(b)	12.9	1.7	11.2	5.3	3.1	1.33
B8D-2 LD	23.0	1.6	21.4	3.3	4.0	2.93
B8D-2 HD	12.0	1.8	10.2	2.5	4.2	1.52
B8D-3 LD	16.3	1.5	14.8	-0.8 ^(c)	3.0	6.73
B8D-3 HD	11.3	1.7	9.6	-1.2	2.6	6.86

(a) LD = Low density (85% T.D.) fuel column

(b) HD = High density (94% T.D.) fuel column

(c) Negative value indicates initial fuel stoichiometry was below equilibrium value.

The correlation of these values with the major fuel test parameters appears to be primarily a major effect due to density and a minor effect due to stoichiometry. Since stoichiometry is known to be a major parameter in fuel sodium reaction, the data indicate that factors other than the major fuel parameters were controlling for fuel volume increase during irradiation. Based on the following analysis, it is concluded that sodium availability was controlling.

Sufficient work has been performed by Aitken¹⁴ to permit a rough estimation of the total oxygen available to participate in the sodium-fuel reaction in each pin due to the thermally induced radial oxygen shift in the fuel. Using the approach of Anselin,¹³ the oxygen available to participate in the sodium-fuel reaction in each pin due to burnup was estimated. Summation of these two values gives the total oxygen available for reaction in each pin.

Using the fission yield tables of Meek and Rider,¹⁵ the average valence of the fission product atoms generated by fissioning of uranium or plutonium was calculated to be 1.695 and 1.221, respectively. (See Tables 6-2 and 6-3.) Therefore, 0.31 and 0.78 excess atoms of oxygen are released per fissioning of an uranium or plutonium atom. Using these values, the equivalent O/M shift of the fuel calculated for the B8C and B8D pins due to the radial redistribution of the oxygen and burnup are shown in columns four and five of Table 6-1. Division of the fuel volume change due to sodium-fuel reaction by the total O/M shift in the fuel of each pin gives the fuel volume increase per 0.01 O/M shift of the fuel (column six of Table 6-1). A sample calculation is shown in Appendix D.

As can be observed in Table 6-1, the measured fuel volume increase per 0.01 O/M shift of the fuel generally varies. These data may be interpreted in terms of limited sodium availability. The fuel in each pin would react as long as sodium was available locally for reaction. However, upon depletion of the sodium and the attendant fuel swelling, the region was sealed from further access to the sodium in the reservoir at the top of the pin (except by long term diffusion processes and fuel pellet cracking on thermal cycling) and no further reaction would occur even though greater amounts of oxygen would become available for reaction.

* The total volume change was defined as follows:

$$\Delta V_{\text{Total}} = (V_F - V_{CV}) - V_O$$

Where V_F = Volume enclosed by post-irradiation unreacted fuel surface plus the volume of the reaction phase.
 V_{CV} = Excess volume, if any, in center void of fuel above that due to fuel densification.
 V_O = Volume enclosed by pre-irradiation fuel surface.

Table 6-2
**AVERAGE VALENCE OF FISSION PRODUCT
 ATOMS GENERATED BY FISSIONING OF URANIUM**

Fission	Yield	Normalized	Valence	Normalized Fission Yield X Valence
Ge	0.000826	—	—	—
As	0.0012	—	—	—
Se	0.418	0.406	2-	-0.812
Br	0.18	0.175	1-	-0.175
Kr	3.651	3.541	0	0
Rb	3.83	3.715	1+	+3.715
Sr	9.410	9.127	2+	+18.254
Y	4.76	4.617	3.5+	+16.160
Zr	31.00	30.068	4+	+120.272
Mo	24.921	24.171	0	0
Tc	6.160	5.748	0	0
Ru	11.002	10.671	0	0
Rh	3.0	2.910	0	0
Pd	1.577	1.530	0	0
Ag	0.03	0.029	0	0
Cd	0.077	0.075	0	0
In	0.0099	0.010	0	0
Sn	0.099	0.096	0	0
Sb	0.051	0.050	0	0
Te	2.560	2.483	2-	-4.966
I	1.137	1.103	1-	-1.103
Xe	27.342	26.520	0	0
Cs	19.32	18.739	0	0
Ba	6.716	6.514	2+	+13.028
La	6.48	6.285	3.5+ ^(a)	+21.998
Ce	14.94	14.491	4+	+57.964
Pr	6.10	5.917	3.5+	+20.710
Nd	17.76	17.226	3.5+	+60.291
Pm	2.19	2.124	3.5+	7.434
Sm	1.283	1.245	3.5+	+4.385
Eu	0.619	0.600	3.5+	+2.100
Gd	0.0241	0.023	3.5+	+0.081
Tb	0.00104	0.001	3.5+	+0.004
Dy	0.000085	—	—	—
	206.200	200.210		339.313

$$\text{Average Valence} = \frac{339.313}{200.21} = 1.695 \quad (2.00 - 1.695 = 0.31 \text{ Excess Oxygen Atoms})$$

(a) Rare earth metals assumed to be at equivalent valence of the plutonium.

Table 6-3
**AVERAGE VALENCE OF FISSION PRODUCT
 ATOMS GENERATED BY FISSIONING OF PLUTONIUM**

Fission	Yield	Normalized	Valence	Fission Yield X Valence
Ge	0.00408	0.004	0	0
As	0.0014	0.001	0	0
Se	0.266	0.259	2-	-0.518
Br	0.178	0.173	1-	-0.173
Kr	1.634	1.587	0	0
Rb	1.47	1.428	1+	+1.428
Sr	3.610	3.507	2+	+7.014
Y	1.73	1.681	3.5+	+5.884
Zr	19.03	18.489	4+	+73.956
Mo	23.704	23.030	0	0
Tc	5.9	5.732	0	0
Ru	17.685	17.182	0	0
Rh	5.6	5.441	0	0
Pd	19.220	18.674	0	0
Ag	1.2	1.166	0	0
Cd	0.514	0.499	0	0
In	0.039	0.038	0	0
Sn	0.341	0.331	0	0
Sb	0.169	0.164	0	0
Tc	3.638	3.535	2-	-7.070
I	2.16	2.099	1-	-2.099
Xe	27.805	27.015	0	0
Cs	20.37	19.791	0	0
Ba	6.006	5.835	2+	+11.670
La	5.67	5.501	3.5+ ^(a)	+19.282
Ce	12.28	11.931	4+	+47.724
Rr	5.75	5.586	3.5+	+19.551
Nd	14.50	14.088	3.5+	+49.308
Pm	1.87	1.817	3.5+	+6.360
Sm	1.895	1.841	3.5+	+6.444
Eu	1.36	1.321	3.5+	+4.624
Cd	0.219	0.213	3.5+	+0.746
Tb	0.021	0.020	3.5+	+0.070
Dy	0.00808	0.008	3.5+	+0.028
Ho	0.00014	-	-	-
	205.948	199.995		244.229

$$\text{Average Valence} = \frac{244.229}{199.995} = 1.221 \quad (2.00 - 1.221 = 0.78 \text{ Excess Oxygen Atoms})$$

(a) Rare earth metals assumed to be at equivalent valence of the plutonium.

Since the fuel pins were deficient in sodium *in the fueled region* during irradiation, the original test objective of directly measuring the effect of stoichiometry, plutonium content of the fuel, density, and burnup upon sodium-fuel reaction was confounded. However, sufficient differences between pins do occur to permit estimation of the effect of these parameters upon the reactivity of the fuel with sodium. Profilometry of the B8D pins most clearly shows the effect of stoichiometry and plutonium content upon the degree of reactivity of the fuel with sodium (see Table 5-1).

Although similar fuel-volume changes were measured for the high density fuel columns in the B8D-1 and B8D-2 pins, the pin with the higher plutonium content exhibited a significantly larger diametral increase. From these data one may infer that the reactivity of mixed-oxide fuel decreases with decreasing plutonium content of the fuel and stoichiometry.

6.2 THERMAL PERFORMANCE OF SODIUM-BONDED FUEL

The work of Ippolitova, Simanov, Kovba, Murav'eva, and Krasnoyarskaya,¹⁷ indicates that sodium-uranium compounds are relatively stable to about 1200°C. It has also been observed in a number of failed pins⁴ that no fuel-sodium reaction phase was present in the highest temperature regions indicating that the sodium-fuel reaction product is unstable at high temperatures. Therefore, for the present analysis, it was assumed that the upper temperature limit of the interface between the fuel and the reaction phase is in the range of 1200°C.

Since no data are presently available on the thermal properties of the fuel-sodium reaction phase, comparison between the thermal performance of sodium-bonded and gas-bonded fuel was made in terms of an effective heat transfer coefficient (h_{eff}). This effective heat transfer coefficient is defined as follows:

$$\Delta T_{eff} = \Delta T \text{ between fuel surface and clad inside surface}$$

$$h_{eff} = \frac{Q}{A \Delta T_{eff}}$$

Where A = Area of heat transfer surface, i.e., inside surface of cladding.

Q = Total heat being transferred through area A.

Comparison of h_{eff} with h , the heat transfer coefficient for gas-bonded fuel, then provides a direct measure of the thermal performance characteristics in the gap between sodium-bonded and gas-bonded fuel (for similar powers and geometry) since each coefficient is related to a temperature drop between the inside surface of the cladding and the edge of the fuel.

To apply this method ΔT_{eff} must be determined independently as a function of the fuel pin power. Figure 6-1 shows center temperatures versus powers measured for fuel pin B8B-1 (central thermocouple). The curve shows a pronounced bend at about 630°C. The initial portion of the curve represents center temperatures measured during the initial rise to power during reactor startup. The fuel pin was then irradiated at a low power for 5 days. The upper portion of the curve represents center temperatures measured during subsequent higher powered operation. Plotted on the same curve are center temperatures measured during a similar experiment performed for a gas-bonded pin.⁹ The two curves agree quite closely (prior to the bend in the B8B-1 curve). This similarity between the two curves indicates relatively similar thermal properties between the two pins. A radical improvement in gap conductance as evidenced by the bend in the B8B-1 curve is most unlikely. It is, therefore, concluded that the B8B-1 central thermocouple failed sometime during the initial 5 days of low power operation.

Assuming the correct functioning of the thermocouple during reactor startup, h_{eff} and the fuel and clad interface temperatures were calculated. Values calculated for the two parameters are shown in Table 6-4. A sample calculation is shown in Part E.1 of Appendix E.

Extrapolation of the fuel surface temperatures calculated for fuel pin B8B-1 indicated that a fuel surface temperature of 1200°C is reached at approximately 13 kW/ft (Figure 6-2). The B8B fuel pins operated at a maximum power of 12.5 kW/ft. All other pins operated at higher maximum powers. Therefore, the fuel surface temperatures in the higher powered pins were established at 1200°C. Table 6-4 shows the value of h_{eff} calculated for these pins.

Values for fuel-cladding gap coefficients are not readily available. Craig, et al.,¹⁸ have investigated this parameter for gas-bonded mixed-oxide fuels. Figure 6-3 shows the range measured for the heat transfer coefficient as a function of the post-irradiation diametral gap width for a series of thermal flux irradiations. Plotted in Figure 6-3 are the values of h_{eff} calculated for the B8D fuel pins. Since gap conductance is a function of the operating power, only the h_{eff} values from the B8D pins, which operated in a similar power range, can be compared directly with the gas-bonded fuel pin data. In the range in which the data overlap, the B8D fuel pins appear to show slightly better heat transfer properties. The heat transfer through the reaction phase of the B8D pins appears to be equivalent to that of a gas-bonded pin with a diametral gap of approximately 0.003 in. operating in the same power range.

Figure 6-1. Center Temperature versus Operating Power Measured for Fuel Pin B8B-1

Figure 6-2. Interface Temperature of Reaction Phase as a Function of Fuel Pin Power

Table 6-4
CALCULATED VALUES FOR h_{eff}

Fuel Pin	Power	Inside Cladding Temperature ($^{\circ}$ F)	Temperature ($^{\circ}$ F)	h_{eff} Btu/h-ft 2 - $^{\circ}$ F
B8A-1	23	1247	1200	1438
B8B-1	2.3	244	275 ^(b)	460
	3.7	316	389 ^(b)	493
	4.6	365	473 ^(b)	505
	12.5	747	1160	515
B8C-1	20.8	1114	1200	1650
B8C-2	22.1	1208	1200	1300
B8D-1 LD ^(a)	18.1	1024	1200	905
B8D-1 HD	18.1	1024	1200	895
B8D-2 LD	18.9	1062	1200	970
B8D-2 HD	18.9	1062	1200	970
B8D-3 LD	18.1	1024	1200	900
B8D-3 HD	18.1	1024	1200	900

(a) LD = Low Density Fuel Column—85% T.D.
HD = High Density Fuel Column—94% T.T.

(b) Calculated from central temperature data

Figure 6-3. Heat Transfer Coefficients versus Fuel-Clad Gap Width

Although the B8D fuel pins show slightly better heat transfer properties compared to gas-bonded pins with a gap size similar in width to the reaction phase, no major improvement was measured.

6.3 FISSION GAS RELEASE

As previously discussed in Section 5.5, only the fission gas samples obtained from fuel pins B8C-1, B8C-2, and B8D-1 are considered representatives of the true fission gas release values. Table 6-5 shows the fission gas release values calculated for the three pins. Two values are shown for fuel pin B8C-1. The lower value represents the fission gas release measured at the first interim examination of the pin prior to failure of the bellows assembly. The isotropic fission gas yields compiled by Meek and Rider¹⁵ were used for the calculations. A sample calculation is shown in Appendix F. Good agreement was obtained between measured and calculated isotropic fission gas yields (see Table 6-6).

Figure 6-4 shows fission gas release values compiled by Bailly, et al.,²⁰ for a number of mixed-oxide fuel, thermal flux irradiations (gas-bonded). The data plotted cover a wide range of powers and fuel parameters. The fission gas release values measured for fuel pins B8C-1, B8C-2, and B8D-1 are also plotted in Figure 6-4. The B8C and B8D values fall well within the scatter of the data. From these data, it is concluded that fission gas release in sodium-bonded fuel is in the same range as that measured for gas-bonded fuel. This lends support to the similar thermal performance of sodium-bonded and gas-bonded fuel. If fuel temperatures were grossly dissimilar for the two cases, one would expect this to be reflected in dissimilar fission gas release values.

6.4 CLAD CORROSION

Of primary interest is the difference in the degree of clad attack noted between fuel pins B8A-1 and B8A-2. Fuel pin B8A-1, which was sodium-bonded, showed only minor attack while gross attack of the cladding of gas-bonded fuel pin B8A-2 occurred. The clad inside surface temperatures in both pins were in the range of 1250°F. Although fuel pin B8A-2 failed, analysis of the failure indicated that no sustained high clad temperatures occurred below the region of the failure. The more severe attack of the cladding in the lower portion of fuel in B8A-2 cannot, therefore, be attributed to the failure of the pin. The difference in attack between the two pins then reflects a true difference between sodium-bonded and gas-bonded cladding.

The process by which clad corrosion such as the above occurs has been investigated by Aitken.²¹ Based on thermodynamic considerations and the out-of-pile work of Lauritzen,²² the corrosion process appears to occur by the following process. Initially, when oxygen from the fuel is readily available, the iron, nickel, and chromium are oxidized to Fe_3O_4 , NiO , and Cr_2O_3 . However, as oxygen becomes less readily available, only the chromium continues to oxidize since chromium has the greatest affinity for oxygen. The Fe_3O_4 and NiO present are reduced to iron and nickel and the oxygen forms more Cr_2O_3 . The result of this corrosion is a metallic layer composed chiefly of nickel and iron overlaying an oxide layer of Cr_2O_3 on the clad surface (in agreement with metallographic observation). With sodium present, compounds such as Na_2FeO_2 may also form at the low temperature regions of the cladding. Formation of the fuel-sodium reaction phase, however, could substantially reduce the total oxygen available for reaction with the cladding, hence, the reduced clad attack of fuel pin B8A-1 compared to pin B8A-2. These data suggest that small additions of sodium to a fuel pin, which are insufficient to cause significant fuel-sodium reaction, may act as a getter and permit operation of the pins at higher clad temperatures than presently contemplated.

Table 6-5
CALCULATED FISSION GAS RELEASE FOR
FUEL PINS B8C-1, B8C-2, AND B8D-1

Fuel Pin	Burnup (MWd/Te)	Percent Fission Gas Release
B8C-1	34,200	45
	129,800	84
B8C-2	69,490	77
B8D-1	52,740	78

Table 6-6
MEASURED AND CALCULATED ISOTOPIC
FISSION GAS YIELDS FOR
FUEL PIN B8C-1

Isotope	Volume Percent	
	Measured	Calculated
Kr-83	14.81	14.61
Kr-84	27.89	26.78
Kr-85	7.05	7.57
Kr-86	<u>50.25</u>	<u>51.06</u>
	100.00	100.02
Xe-131	12.29	11.43
Xe-132	17.82	16.57
Xe-134	28.63	28.55
Xe-136	<u>41.26</u>	<u>43.45</u>
	100.00	100.00
Xe/Xe + Kr	0.887	0.894

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APPENDIX A

CAPSULE DESIGN DETAILS

To meet test requirements, each capsule was designed to permit two or three axially aligned fuel pins to be irradiated simultaneously. Essentially the same capsule design was used for all four tests with minor modifications to meet the requirements of the individual test.

The capsule design consisted of a NaK filled stainless steel inner containment enclosing each fuel pin. Each inner containment was self-contained to reduce the chance that failure of one pin would propagate to another pin. The inner containment was also equipped with a stainless steel bellows. The bellows eliminated the necessity of leaving a void at the top to provide for expansion of the NaK thermal bond during irradiation. This design permitted the fuel pins to be handled in any attitude during capsule loading and reduced the possibility of gas entrapment in the NaK filled heat transfer annuli. Such entrapment of a bubble could result in failure of the pin during irradiation. The inner containment was surrounded by an aluminum sleeve which served as a thermal barrier. The thermal barrier was used to adjust clad temperatures to realistic LMFBR values. A NaK annulus on each side of the aluminum thermal barrier thermally bonded it to the inner and outer containments. The outer containment consisted of a Zircaloy-2 (B8A) or stainless steel tube.

Each fuel pin was monitored by means of two pairs of chromel-alumel thermocouples (1/16-inch o.d.). One thermocouple from each pair was positioned in the NaK annulus between the inner containment and the inner edge of the thermal barrier. The remaining thermocouple of each pair was positioned in the NaK annulus between the outer edge of the thermal barrier and the outer containment directly opposite the first thermocouples. The pairs of thermocouples were positioned 180 degrees apart within the capsule. These thermocouples were used to position the capsule and to measure fuel pin power during irradiation. Figure A-1 shows two radial cross sections through capsule B8C. The cross-sectional dimensions of the 1/4 inch pin are representative of the remaining capsules.

Each capsule was also instrumented with six chromel-alumel thermocouples (1/16-inch o.d.) located on the outer surface. During irradiation, these thermocouples (two inlet and four outlet) were located in the water annulus surrounding the capsule through which cooling water was pumped. The thermocouples were used to perform calorimetry measurements on the capsules and provided a second independent means of establishing fuel pin power. Figure A-2 shows a schematic representation of the essential details of each capsule.

Figure A-1. Radial Cross-Sections Through Capsule B8C

Figure A-2. Capsule Details

APPENDIX B

FUEL PIN DESIGN DETAILS

Fuel pin designs varied greatly between capsules depending upon specific test objectives. Figure B-1 shows a schematic representation of the fuel pin designs used in the different capsules. The detailed fabrication parameters for each fuel pin are listed in Table B-1. The fuel columns in the B8A and B8B pins were axially restrained by a hold-down tube placed in the plenum region of the pins. The remaining pins contained a spring in the plenum region which compressed the fuel column during fabrication. The springs were pre-loaded to exert ~ 1.6 pounds force on the fuel (0.8 pound force in pin B8C-1) or approximately 45 psi stress. This design change between the B8A and B8B pins and the B8C and B8D pins was made to reduce the axial heat flow from the plenum region.

An insulator pellet was placed at the top and bottom of the fuel columns in the B8A and B8B fuel pins. However, subsequent thermal analysis indicated only a limited portion of the sodium reservoir in the plenum region of the pin was molten during irradiation. Therefore, in all the other fuel pins, no insulator pellet was placed at the top of the fuel pin to increase the volume of molten sodium available for reaction with the fuel. In these pins, a washer containing transverse grooves was placed below the bottom insulator pellet to help distribute the sodium around the periphery of the fuel and prevent streaming of the sodium during sodium filling.

The central thermocouple used in fuel pin B8B-1 (also B8A-1) was of the tungsten-5 wt. % rhenium/tungsten-26 wt. % rhenium type with a 0.062-inch o.d. tungsten-26 wt. % rhenium sheath. The thermocouple was insulated with vitrified BeO. The top 4 pellets of the fuel column in the pin were annular to accommodate the thermocouple. The annular pellets contained a 0.070 inch diameter central hole. The junction of the thermocouple projected 7/8 of an inch (3-1/2 pellet lengths) into the fuel column.

The bellows assembly which made up part of the plenum region of fuel pin B8C-1 was enclosed in a separate helium filled chamber to retard heat flow. Condensed sodium in the bellows chamber could prevent operation of the bellows. Thermal calculations indicated that during irradiation, the bellows chamber was maintained at a temperature of approximately 300 to 400°F by γ -heating of the assembly.

The screen wire sleeve placed in the gap of fuel pin B8C-2 was fabricated over a mandrel which duplicated the diameter of the fuel column. The sleeve was fabricated from three panels of 200 mesh screen wire composed of Types 304, 316, and 321 stainless steel. The sleeve was formed by tweezer welding the edges together over 1-mil thick by 50-mil wide stainless steel strips. The length of the sleeve was the same as the fuel column. The sleeve was inserted around the fuel column prior to loading into the cladding.

Each fuel pin (except B8A-2) was sodium filled to approximately 1 inch above the fuel column. This provided additional sodium for reaction with the fuel besides the sodium contained in the fuel-clad gap area. Details of the sodium bonding of the fuel pins are discussed in Appendix C.

Figure B-1. Fuel Pin Details

**Table B-1
FUEL PIN PARAMETERS**

Fuel	B8A-1	B8A-2	B8B-1	B8B-2	B8C-1	B8C-2		B8D-1	B8D-2	B8D-3
Material	← Coprecipitated (Pu,U)O ₂ →									
Ratio Pu/Pu+U	0.25	0.25	0.25	0.247	0.248	0.246	Top FC ^(a)	0.396	0.252	0.248
Ratio U-235/U-235+238	0.40	0.40	0.40	0.414	0.929	0.299	Bot FC		0.246	
Fuel Density (% T.D.)							Top FC	0.9308	0.089	0.930
Range	89.4-91.2	90.6-91.2	88.5-90.6	90.3-90.8	93.8-95.0	93.9-95.4	Bot FC		0.082	
Average	90.5	90.9	89.2	90.5	94.6	94.9		85.8-86.4	85.4-86.0	84.6-86.3
								86.1	85.6	85.2
								92.8-93.0	94.4-94.6	93.5-94.9
								93.0	94.5	94.2
O/M Ratio										
Range	1.999-2.003	2.001	2.000	1.931	1.966	1.961-1.966		1.965-1.971	1.967-1.973	1.935
Average	2.001	2.001	2.000	1.931	1.966	1.962		1.966	1.971	1.935
Pellet Dimensions (in.)										
O.D.-Range	0.2156-0.2176	0.2168-0.2177	0.2158-0.2171	0.2155-0.2169	0.1433-0.1443	0.2019-0.2043		0.2146-0.2150	0.2136-0.2148	0.2133-0.2150
Average	0.2168	0.2172	0.2164	0.2163	0.1438	0.2037		0.2147	0.2142	0.2141
Annulus	0.070 Nom.	—	0.070 Nom.	—	—	—		—	—	—
Fuel Column Length (in.)	1.997	2.002	2.0055	1.994	2.263	2.291	Top FC	1.005	1.047	1.0245
							Bot FC	0.700	0.714	0.710
Diametral Gap										
Range	0.0024-0.0044	0.0023-0.0032	0.0080-0.0120	0.0099-0.0113	0.0051-0.0069	0.0157-0.0188		0.0050-0.0060	0.0052-0.0070	0.0050-0.0072
Average	0.0032	0.0028	0.0106	0.0106	0.0059	0.0166	Top FC	0.0054	0.0060	0.0067
Fuel Cladding Bond	Sodium	Helium	Sodium	Sodium	Sodium	Sodium	Bot FC	0.0053	0.0064	0.0055
Insulator								Sodium	Sodium	Sodium
Material	← Natural or Depleted UO ₂ →									
Density (% T.D.)										
Top Pellet	90.0 Nom.	90.0 Nom.	90.0 Nom.	90.0 Nom.	—	—		—	—	—
Middle Pellet	—	—	—	—	—	—		92.9	95.1	95.0
Bottom Pellet	90.0 Nom.	90.0 Nom.	90.0 Nom.	90.0 Nom.	94.7	95.7		95.0	95.6	95.7
Pellet Dimensions (in.)										
Top Pellet										
O.D.	0.2172	0.2167	0.2167	0.2170	—	—		—	—	—
Length	0.2586	0.2092	0.2536	0.2538	—	—		—	—	—
Middle Pellet										
O.D.	—	—	—	—	—	—		0.2129	0.2141	0.2148
Length	—	—	—	—	—	—		0.1299	0.1337	0.1341
Bottom Pellet										
O.D.	0.2173	0.2171	0.2168	0.2170	0.1433	0.2130		0.2150	0.2128	0.2137
Length	0.2582	0.2512	0.2493	0.2511	0.1865	0.2547		0.2493	0.2543	0.2558
O/M Ratio										
Top Pellet	2.00 Nom.	2.00 Nom.	2.00 Nom.	2.00 Nom.	—	—		—	—	—
Middle Pellet	—	—	—	—	—	—		1.998	1.996	1.998
Bottom Pellet	2.00 Nom.	2.00 Nom.	2.00 Nom.	2.00 Nom.	1.998	1.998		1.998	1.998	1.998
Cladding										
Material	347 SS	347 SS	347 SS	347 SS	316 SS	316 SS		316 SS	316 SS	316 SS
O.D. (Nom.)	0.250	0.250	0.250	0.250	0.170	0.250		0.250	0.250	0.250
Wall Thickness (in.)	0.015 Nom.	0.015 Nom.	0.015 Nom.	0.015 Nom.	0.010 Nom.	0.015 Nom.		0.015 Nom.	0.015 Nom.	0.015 Nom.
Gap Sleeve										
Material	—	—	—	—	—	200 Mesh Screen Wire (304,316,321 SS)		—	—	—
Length (in.)	—	—	—	—	—	2.5 Nom.		—	—	—

(a) FC = Fuel Column

APPENDIX C

SODIUM BONDING OF FUEL PINS

The fuel pins were sodium-bonded using a pressure-manometer technique. Figures C-1 and C-2 show the apparatus used. During heat-up, filling, and cool-down operations, the whole apparatus was placed in a fume hood and the interior of the plexiglass box inerted with nitrogen.

At the start of the test series and prior to sodium-bonding of the actual fuel pins, a series of dummy pins was filled to check out the system. In one series of tests urania and yttria pellets were used to simulate the fuel in the dummy pins. After sodium-bonding, an "eddy current" map was made of the dummy pins. The pins were then destructively examined for completeness of bonding. Figure C-3 shows the representative appearance of the sodium bond obtained in the pins. Examination of the pins confirmed that a relatively uniform distribution of sodium could be obtained around the fuel-clad gap of a pin. Comparison of the "eddy current" maps with the actual sodium distribution found in the pins indicated that "eddy current" measurements only distinguished areas where the sodium wet the surface of the cladding. The tests further indicated that sufficient clad surface area is wet to permit reliable estimation of the sodium fill level.

Sodium-bonding of the fuel pins was achieved in the following manner. The fuel pin made up one leg of the manometer tube and a clear quartz tube made up the other leg. Both legs of the manometer were inside a small clam shell furnace. The fuel pin and quartz tube were heated and baked at temperature (150-200°C) for several hours while being continuously purged with high purity helium. A small pot of molten sodium, which was connected to the bottom of the manometer assembly, was then pressurized and the sodium was forced into both legs of the manometer. The pot was filled from a sodium source containing approximately 20 ppm oxygen prior to each filling operation. The actual oxygen content of the sodium entering the pins would be somewhat higher due to some oxygen pickup from adsorbed gases and from the oxide film on the surface of the stainless steel tubing connecting the pot to the fill system. By alternately pressurizing the two manometer legs, the sodium was forced up and down inside the fuel pin. This was done several times to evenly distribute the sodium within the fuel pin. Finally, the sodium was forced up into the fuel pin to its design level which was checked by "eddy current" measurements. After the correct level was achieved, the fill tubes on the fuel pin were pinched closed and welded. The fill level was further checked by x-ray radiography.

After closure, all pins were baked for several hours at a temperature of approximately 400°C to enhance bonding between components. Figure C-4 shows an "eddy current" map of capsule B8B fuel pins showing the distribution of the sodium wetting the surface of the cladding after sodium filling.

Figure C-1. Sodium Fill System

Figure C-2. Sodium Fill System Furnace and Manometer Assembly

NOTE SODIUM FITTING CONTOUR OF PELLET

Figure C-3. Appearance of Sodium-Bond in Dummy Fuel Pin

Figure C-4. Eddy Current Maps of Capsule B8B Fuel Pins

APPENDIX D

FUEL VOLUME CALCULATIONS

The fuel volume increase and release of oxygen from the fuel to participate in the sodium-fuel reaction were calculated for the B8C and B8D fuel pins as shown below. The present sample calculation is for fuel pin B8C-1.

FUEL PIN B8C-1

D.1 PERCENT FUEL VOLUME INCREASE

From neutron radiographs and metallography, the post-irradiation fuel dimensions were determined. Figure D-1 shows a schematic diagram of the pre- and post-irradiation fuel dimensions used in the following calculations.

VCV

Post-irradiation center void volume, V_{cv} :

$$V_{c1} = \frac{3.14 \times (0.02)^2 \times 0.32}{4} = 0.00010 \text{ in.}^3$$

$$V_{c2} = \frac{3.14 \times (0.04)^2 \times 0.2}{4} = 0.00025 \text{ in.}^3$$

$$V_{c3} = \frac{3.14 \times (0.05)^2 \times 0.1}{4} = 0.00019 \text{ in.}^3$$

$$V_{c4} = \frac{3.14 \times (0.04)^2 \times 0.85}{4} = 0.00106 \text{ in.}^3$$

$$V_{c5} = \frac{3.14 \times (0.035)^2 \times 0.36}{4} = 0.00034 \text{ in.}^3$$

$$V_c = V_{c1} + V_{c2} + V_{c3} + V_{c4} + V_{c5} = 0.00194 \text{ in.}^3 \times 16.39 \frac{\text{cc}}{\text{in.}^3} = 0.0318 \text{ cc}$$

Volume of initial porosity in fuel, V_p :

Initial pellet density - 94.6% T.D.

$$V_p = \frac{3.14 \times (0.1438)^2 \times 2.263 \times 16.39 \frac{\text{cc}}{\text{in.}^3} \times (1 - 0.946)}{4} = 0.0338 \text{ cc}$$

$$V_{cv} = 0 \quad V_p > V_c \text{ (See Section G.1)}$$

 V_F

Volume of unreacted fuel and reaction phase, V_F :

$$V_{F1} = \frac{3.14 (0.1511)^2 \times 0.21}{4} = 0.0436 \text{ in.}^3$$

$$V_{F2} = \frac{3.14 (0.08)^2 \times 0.21}{4} = 0.00074 \text{ in.}^3$$

$$V_F = V_{F1} + V_{F2} = 0.0443 \text{ in.}^3 \times 16.39 \frac{\text{cc}}{\text{in.}^3} = 0.7261 \text{ cc}$$

 V_o

Initial volume of fuel, V_o :

$$V_o = \frac{3.14 \times (0.1438)^2 \times 2.263 \times 16.39 \frac{\text{cc}}{\text{in.}^3}}{4} = 0.6032 \text{ cc}$$

PRE-IRRADIATION

POST-IRRADIATION

ALL DIMENSIONS ARE IN INCHES

Figure D-1. Pre- and Post-Irradiation Dimensions for Fuel Pin B8C-1

V_T Total fuel volume increase, V_T :

$$V_T = (V_F - V_C) - V_o$$

$$V_T = (0.7261 - 0) - 0.6032 = 0.1343 \text{ cc}$$

% Fuel Volume Increase

$$\% \Delta V/V = V_T/V_o \times 100 = 0.1343/0.6032 \times 100 = \underline{\underline{22.3}}$$

D.2 RADIAL OXYGEN SHIFT OF FUEL

The radial oxygen shift of the fuel was calculated by means of the following formula:

$$\frac{X_2}{X_1} = \exp \left[\frac{Q}{R} \left(\frac{1}{T_2} - \frac{1}{T_1} \right) \right] \quad (23)$$

where

 Q = heat of transport for oxygen (-5.7 Kcal) X = oxygen value for the deviation from stoichiometry ($\text{MO}_2 \pm X$) $T_1 X_1$ = absolute temperature and oxygen value at the fuel surface $T_2 X_2$ = absolute temperature and oxygen value of fuel at radius of interest R = gas constant ($0.0019872 \frac{1}{(\text{kcal}) (\text{gm-mole})}$)

Using the above formula along with the calculated temperature profile and surface equilibrium O/M value** the O/M profile through the fuel was calculated. Figure D-2 shows the O/M profile obtained. A sample calculation for one value is shown below:

Fuel Surface Temperature - 1200°C Temperature of fuel at radial fraction 0.98 - 1287°C

$$X_1 = 2.00 - 1.97 = 0.03$$

$$X_2 = 0.03 \exp \left[\frac{-5.7}{1.9872 \times 10^{-3}} \left(\frac{1}{1297 + 273} - \frac{1}{1200 + 273} \right) \right] = 0.0334$$

$$\text{O/M}_{0.98} = 2.00 - 0.0334 = 1.9666$$

Using the calculated O/M profile, the average volumetric O/M value (O/M_{AV}) of the fuel was then obtained by numerically integrating across the fuel cross section using the O/M profile.

$$\text{O/M}_{AV} = \left(\frac{1.97 + 1.966}{2} \right) \times \frac{1^2 - 0.88^2}{1^2 - 0.364^2} + \dots = 1.9415$$

The radial O/M shift of the fuel due to the presence of sodium was then calculated.

Initial O/M of Fuel - 1.966

$$\Delta \text{O/M} = 1.966 - 1.9415 = 0.0245 \text{ or } \underline{\underline{0.025}}$$

* Center void radial fraction.

$$\begin{array}{l} \text{B8C-1} \\ \text{B8C-2} \\ \text{B8D-2} \\ \text{B8D-3} \end{array} \left\{ \begin{array}{l} 0.75 \text{ UO}_2 + 0.25 \text{ PuO}_{1.88} = (\text{U, Pu})\text{O}_x \\ x = 0.75 \times 2 + 0.25 \times 1.88 = \underline{1.97} \end{array} \right.$$

$$\begin{array}{l} \text{D8D-1} \\ \text{D8D-2} \\ \text{D8D-3} \end{array} \left\{ \begin{array}{l} 0.6 \text{ UO}_2 + 0.4 \text{ PuO}_{1.88} = (\text{U, Pu})\text{O}_x \\ x = 0.6 \times 2 + 0.4 \times 1.88 = \underline{1.952} \end{array} \right.$$

Figure D-2. Calculated O/M Profile Through Unreacted Fuel of Fuel Pin B8C-1

D.3 OXYGEN SHIFT OF FUEL DUE TO BURNUP

Total fissions (F_T)

Wt of fuel - 6.28 gm (88.2% metal atoms by wt.)

Fuel Burnup - 129,800 MWd/Te

$$F_T = \frac{6.28 \text{ gm} \times 0.882 \times 129,800 \text{ MWd/Te} \times 5.3916 \times 10^{23} \text{ MeV/MWd}}{10^6 \frac{\text{gm}}{\text{Te}} \times 205 \frac{\text{MeV}}{\text{fission}}} = 18.9 \times 10^{20} \text{ fissions}$$

Number of oxygen atoms per 0.01 O/M shift, $O_{0.01 \Delta O/M}$:

Mole wt of fuel - 268.2 gm

Wt. of oxygen per 0.01/O/M shift of a gm-mole of fuel - 0.16 gm

$$O_{0.01 \Delta O/M} = \frac{6.28 \text{ gm of fuel}}{268.2} \times \frac{0.16}{16} \times 6.0235 \times 10^{23} \frac{\text{atoms}}{\text{gm-mole}} = 14.1 \times 10^{19} \text{ atoms of oxygen}$$

Excess oxygen generated due to fissioning, of

$$\frac{Pu}{U+Pu} = 0.25$$

$$\frac{U-235}{U-235 + U-238} = 0.93$$

Using GETR fissions cross sections of 595 and 401 for Pu-239 and U-235, an estimate of the partitioning of the total fissions between the plutonium and uranium may be made.*

$$\% \text{ fissions due to uranium} = \frac{1}{1 + \frac{595 + 0.25 \times 0.92}{401 \times 0.75 \times 0.93}} \times 100 = 67.2$$

$$\% \text{ fissions due to plutonium} = (1 - 0.672) \times 100 = 22.8$$

$$F_{U-235} = 18.9 \times 10^{20} \text{ fissions} \times 0.672 = 12.70 \times 10^{20} \text{ fissions}$$

$$F_{Pu-239} = 18.9 \times 10^{20} \text{ fissions} \times 0.228 = 6.20 \times 10^{20} \text{ fissions}$$

$$O_{U-235} = 12.70 \times 10^{20} \times \frac{0.31 \text{ atoms of oxygen}}{U \text{ fission}} = 4.02 \times 10^{20} \text{ atoms of oxygen}$$

$$O_{Pu-239} = 6.20 \times 10^{20} \times 0.78 \frac{\text{atoms of oxygen}}{Pu \text{ fission}} = 4.85 \times 10^{20} \text{ atoms of oxygen}$$

$$O_F = O_{U-235} + O_{Pu-239} = 8.87 \times 10^{20} \text{ atoms of oxygen}$$

Equivalent O/M Shift of Fuel Due to Burnup, $\Delta O/M$:

$$\text{Number of } 0.01 \text{ O/M shifts} = \frac{O_F}{O_{0.01 \Delta O/M}} = \frac{8.87 \times 10^{20}}{14.1 \times 10^{19}} = 6.29$$

$$\Delta O/M = \underline{\underline{0.063}}$$

* To be consistent, this method of partitioning the fissions was used for all pins because more exact values as determined during the burnup analysis of the fuel were not available for all pins.

APPENDIX E

THERMAL PERFORMANCE CALCULATIONS

Pre- and post-irradiation power profiles were calculated for all pins using fuel compositions and one dimensional transport theory. Using the post-irradiation microstructural changes observed in the fuel of the different pins, the post-irradiation power curves were also corrected for densification changes in the fuel (assuming 97% T.D. for equiaxed grain growth region and 99% T.D. for columnar grain growth region).

An eight term polynomial expression was then derived for each power curve. Using an eight term polynomial equation for the power curve, the following expression may be derived for evaluating the integral conductivity of the fuel between two radii. The expression derived was

$$\int_{T_a}^{T_b} K dt = \frac{r^2 Q^*}{2.54} [(A' + B \ln r_b) - (C - D \ln r_a)]$$

where

$$Q^* = \frac{\left(\frac{Q/L + 1000}{12} \right)}{2\pi r^2 [E - r_a^2 (F)]}$$

$$A = \frac{C0r_b^2}{2^2} + \frac{C1r_b^3}{3^3} + \frac{C2r_b^4}{4^2} + \frac{C3r_b^5}{5^2} + \frac{C4r_b^6}{6^2} + \frac{C5r_b^7}{7^2} + \frac{C6r_b^8}{8^2} + \frac{C7r_b^9}{9^2} + \frac{C8r_b^{10}}{10^2}$$

$$B = \frac{C0r_b^2}{2} + \frac{C1r_b^3}{3} + \frac{C2r_b^4}{4} + \frac{C3r_b^5}{5} + \frac{C4r_b^6}{6} + \frac{C5r_b^7}{7} + \frac{C6r_b^8}{8} + \frac{C7r_b^9}{9} + \frac{C8r_b^{10}}{10}$$

$$C = \frac{C0r_a^2}{2^2} + \frac{C1r_a^3}{3^3} + \frac{C2r_a^4}{4^2} + \frac{C3r_a^5}{5^2} + \frac{C4r_a^6}{6^2} + \frac{C5r_a^7}{7^2} + \frac{C6r_a^8}{8^2} + \frac{C7r_a^9}{9^2} + \frac{C8r_a^{10}}{10}$$

$$D = \frac{C0r_a^2}{2} + \frac{C1r_a^3}{3} + \frac{C2r_a^4}{4} + \frac{C3r_a^5}{5} + \frac{C4r_a^6}{6} + \frac{C5r_a^7}{7} + \frac{C6r_a^8}{8} + \frac{C7r_a^9}{9} + \frac{C8r_a^{10}}{10}$$

$$E = \frac{C0}{2} + \frac{C1}{3} + \frac{C2}{4} + \frac{C3}{5} + \frac{C4}{6} + \frac{C5}{7} + \frac{C6}{8} + \frac{C7}{9} + \frac{C8}{10}$$

$$F = \frac{C0}{2} + \frac{C1r_a}{3} + \frac{C2r_a^2}{4} + \frac{C3r_a^3}{5} + \frac{C4r_a^4}{6} + \frac{C5r_a^5}{7} + \frac{C6r_a^6}{8} + \frac{C7r_a^7}{9} + \frac{C8r_a^8}{10}$$

r = Outside radius of fuel, in.

r_a = Radial fraction at point a

r_b = Radial fraction at point b

Q^* = Volumetric heat generation at surface of fuel, w/in.³

Q/L = Linear power of fuel, kW/ft.

$C0, C1, C2, \dots$ = Coefficients of power curve (Note: power curve normalized to surface value.) $r_b > r_a$

This expression in conjunction with the integral conductivity curves suggested by Bailey, et al.,¹⁹ was used to calculate fuel temperatures.

E.1 CALCULATION OF h_{eff} AND FUEL-REACTION PHASE INTERFACE TEMPERATURES FOR FUEL PIN B8B-1

The interface temperatures were calculated from the measured center temperatures using the following equation:

$$\int_0^{T_c} k dt - \int_{T_i}^{T_c} k dt = \int_0^{T_i} k dt$$

where

T_c = Fuel center temperature, °C

T_i = Fuel interface temperature, °C

Using the appropriate integral conductivity in Figure E-1 (90% T.D.), the integral $\int_0^{T_c} k dt$ was evaluated.

$T_c = 630^\circ\text{C}$

$$\int_0^{T_c} k dt = 30.47 \text{ w/cm}$$

Using the derived expression given above, the integral $\int_{T_i}^{T_c} k dt$ was then computed.

$$\int_{T_i}^{T_c} k dt = \frac{r^2 Q^*}{2.54} [(A - B \ln r_i) - (C - D \ln r_e)]$$

where

$Q/L = 4.6 \text{ kW/ft}$

$r = 0.10895 \text{ in.}$

$r_e = 0.2868$

$r_i = 1$

$C0 = 0.0898813$

$C1 = 0.3234765$

$C2 = 5.957579$

$C3 = 49.38188$

$C4 = -206.462$

$C5 = 484.6254$

$C6 = -637.1875$

$C7 = 437.488$

$C8 = -121.3016$

Evaluation of the equation gives:

$$\int_{T_i}^{T_c} k dt = 5.27 \text{ w/cm}$$

$$\text{therefore: } \int_0^{T_i} k dt = 30.47 - 5.27 = 25.2 \text{ w/cm}$$

Using the appropriate integral conductivity curve in Figure E-1, T_i was found to be

$T_i = 474^\circ\text{C} (885^\circ\text{F})$

h_{eff} was calculated using the following expression:

$$h_{eff} = \frac{Q/L \times 3413 \times 12}{\pi D \Delta T_{eff}}$$

where:

$\Delta T_{eff} = T_i - T_{i.d. \text{ clad}}$

Q/L = Linear power of fuel, kW/ft

D = Inside diameter of cladding, in.

$T_i = 885^\circ\text{F}$

$T_{i.d. \text{ clad}} = 365^\circ\text{F}$

$D = 0.2285 \text{ in.}$

$$h_{eff} = \frac{4.6 \times 3413 \times 12}{3.14 \times 0.2285 \times (885 - 365)} = 505 \frac{\text{Btu}}{\text{h-ft}^2 \cdot ^\circ\text{F}}$$

Values of h_{eff} for the remaining pins were calculated in a similar manner.

Figure E-1. Thermal Conductivity Data for Mixed Oxide Fuel

APPENDIX F

FISSION GAS RELEASE CALCULATIONS

Fission gas release values were calculated as follows for Fuel Pin B8C-1:

F.1 POST-IRRADIATION

Total fissions (F_T)

Weight of fuel - 6.28 gm (88.2% metal atoms by wt)

Fuel burnup - 129,800 MWd/Te

$$F_T = \frac{6.28 \text{ gm} \times 0.882 \times 129,800 \text{ MWd/Te} \times 5.3916 \times 10^{23} \text{ MeV/MWd}}{10^6 \frac{\text{gm}}{\text{Te}} \times 205 \frac{\text{MeV}}{\text{fission}}} = 18.9 \times 10^{20} \text{ fissions}$$

Partitioning of total fissions between fissionable isotopes

Values determined during burnup analysis (U-238 not included):

	% Total Fissions
U-235	77.817
Pu-239	21.625
Pu-241	0.558
$F_{U-235} = 18.9 \times 10^{20} \times 0.77817$	$= 14.70 \times 10^{20} \text{ fissions}$
$F_{Pu-239} = 18.9 \times 10^{20} \times 0.21625$	$= 4.09 \times 10^{20} \text{ fissions}$
$F_{Pu-241} = 18.9 \times 10^{20} \times 0.00558$	$= 0.11 \times 10^{20} \text{ fissions}$

Atoms of Kr generated

$$Kr_{U-235} = 14.70 \times 10^{20} \text{ fissions} \times \frac{3.65 \text{ atoms Kr}^*}{100 \text{ fissions}} = 53.66 \times 10^{18} \text{ atoms Kr}$$

$$Kr_{Pu-239} = 4.09 \times 10^{20} \times \frac{1.63}{100} = 6.667 \times 10^{18} \text{ atoms Kr}$$

$$Kr_{Pu-241} = 0.11 \times 10^{20} \times \frac{1.233}{100} = \frac{0.14 \times 10^{18} \text{ atoms Kr}}{60.47 \times 10^{18} \text{ atoms Kr}}$$

Volume Kr

$$V_{Kr} = \frac{60.47 \times 10^{18} \text{ atoms Kr} \times 2.24 \times 10^4 \frac{\text{cc}}{\text{gram-mole}}}{6.0235 \times 10^{23} \frac{\text{atoms}}{\text{gram-mole}}} = 2.25 \text{ cc}$$

Atoms Xe generated

$$Xe_{U-235} = 14.70 \times 10^{20} \text{ fissions} \times \frac{26.47 \text{ atoms Xe}}{100 \text{ fissions}} = 3.89 \times 10^{20} \text{ atoms Xe}$$

$$Xe_{Pu-239} = 4.09 \times 10^{20} \times \frac{28.63}{100} = 1.17 \times 10^{20} \text{ atoms Xe}$$

$$Xe_{Pu-241} = 0.11 \times 10^{20} \times \frac{28.46}{100} = 0.03 \times 10^{20} \text{ atoms Xe}$$

$$\text{Total} = 5.09 \times 10^{20} \text{ atoms Xe}$$

* Based of fission yields from Reference 15.

Volume Xe

$$V_{Xe} = \frac{5.09 \times 10^{20} \text{ atoms Xe} \times 2.24 \times 10^4 \frac{\text{cc}}{\text{gram-mole}}}{6.0235 \times 10^{23} \frac{\text{atoms}}{\text{gram-mole}}} = 18.93 \text{cc}$$

Total volume fission gas generated

Volume of Kr and Xe in fission gas sample - 17.9cc

$$\frac{17.9 \text{cc}}{21.2 \text{cc}} \times 100 = \underline{\underline{84\%}}$$

F.2 INTERIM EXAMINATION

The bellows was measured and found to be compressed by 0.0725 inch. Figure F-1 shows the calibration curve for the bellows. From the curve, compression of the bellows by 0.0725 inch corresponds to a plenum pressure of 28 psi.

Figure F-1. Bellows Calibration Curve for Fuel Pin B8C-1

Volume of gas in pin

Initial Plenum Volume - 0.2188 in.³

Volume Increase Due to Compression of Bellows - 0.0725 in. X 0.2469 in.² (effective area of bellows)
= 0.0129 in.³

Fuel Center Void Volume - $\frac{3.1416 \times (0.016)^2 \times 2 \text{ in.}}{4}$ = 0.0004 in.³

Total = 0.2371 in.³

Mass of gases present, M_{TOTAL}

$$PV = MRT \text{ (compressibility factor} = 1)$$

where

 P = Pressure, psi V = Volume, in.³ M = Mass of gas, lb-moles
 R = Gas constant, $18,540 \frac{(\text{in.}^3) \left(\frac{\text{lb}}{\text{in.}^2} \right)}{(\text{lb-mole})(^\circ R)}$
 T = Temperature, $^\circ R$ ($595^\circ R^*$)

$$M_{TOTAL} = \frac{28 \times 0.2371}{18,540 \times 595} = 6.02 \times 10^{-7} \text{ lb-moles}$$

Mass of back fill gas, M_{BF}

$$M_{BF} = \frac{0.2188 \text{ in.}^3}{1728 \frac{\text{in.}^3}{\text{ft}^3} \times 359 \frac{\text{lb-moles}}{\text{ft}^3}} = 3.53 \times 10^{-7} \text{ lb-moles}$$

Mass of fission gas generated, M_{FG}

Burnup - 34,200 MWd/Te

$$V_{FG} = 21.2 \times \frac{34,200 \text{ MWd/Te}}{129,800 \text{ MWd/Te}} = 5.6 \text{ cc}$$

$$M_{FG} = \frac{5.6 \text{ cc}}{16.39 \frac{\text{cc}}{\text{in.}^3} \times 1728 \frac{\text{in.}^3}{\text{ft}^3} \times 359 \frac{\text{lb-moles}}{\text{ft}^3}} = 5.5 \times 10^{-7} \text{ lb-moles}$$

Fission gas release

$$\frac{M_{TOTAL} - M_{BF}}{M_{FG}} = \frac{6.02 \times 10^{-7} - 3.53 \times 10^{-7}}{5.51 \times 10^{-7}} \times 100 = \underline{\underline{45\%}}$$

* Temperature measured at time of neutrography.

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