MEASURED LOSS OF SELECTED FISSION PRODUCTS
FROM HIGH-BURNUP MIXED-OXIDE FUEL IN
A MINIATURE PUMPED SODIUM LOOP

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ABSTRACT

A defected section of mixed-oxide fuel preirradiated to $\sim 41,000$ MWd/Te was exposed to $\sim 1150^{\circ}\text{F}$ flowing sodium for 18 hours in a miniature pumped loop in a hot cell. Fuel pin diametral increase and the transfer of fuel and fission product species from the fuel specimen to different portions of the loop were measured, to provide an improved design basis for the accommodation of failed fuel in LMFBR systems. Post-test examination of the fuel specimen showed: (1) an $\sim 3\%$ increase in the average diameter of the fuel section, (2) less than 1% loss of fuel from the specimen with uranium preferentially leached compared with plutonium, and (3) substantial loss of fuel specimen fission products ranging from 29 to 85% for the isotopes investigated. Post-test examination of the loop indicated plate-out of fission products in the colder regions, with the exception of Cs which remained in solution in the sodium.

1. INTRODUCTION

The successful development of liquid-metal-cooled fast breeder reactors (LMFBR's) requires that the reactor operate for extended periods of time with failed fuel pins in the core. Knowledge of the amounts and the species of fission products released from failed fuel pins and the subsequent deposition of these fission products throughout the primary system must be obtained to provide a basis for designing reactors for safe operation, habitation by operating personnel, and economic maintenance.

Loss of fuel and fission products from failed elements to the primary coolant can occur both during irradiation with the elements at power, and during reactor shutdown.

To provide some information on possible fuel and fission product release from failed elements during reactor shutdown, a small pumped sodium loop was fabricated, and a section of preirradiated mixed-oxide fuel was tested to determine the extent of fuel and fission product loss and the increase in diameter resulting from fuel-sodium reaction. This report describes the results of the test.

2. SUMMARY OF RESULTS AND CONCLUSIONS

Results of this 18-hour test in a miniature sodium loop indicate the following facts

- a Significant fission product losses can occur in a relatively short time period. The measured loss of selected fission products from the fuel of the test sample to the loop coolant was

Isotope	Percent Loss
Cs 137	66
Ce-144	32
Ru-106	85
Sr-90	29
Pm-147	32

- b Less than 1% erosion or loss of fuel from the test sample occurred for local sodium velocities of up to 7 ft/sec
- c An average diametral increase of 3% was measured for the test sample, indicating sodium-fuel reaction occurred during the test.
- d Essentially no plate-out of cesium occurred in the loop. Other fission products monitored showed a deposition pattern of increased plate-out in colder regions in the loop.
- e The ratio of uranium to plutonium atoms deposited in the different regions of the loop indicate that uranium was preferentially removed from the fuel
- f The experimental methods appear promising for investigating the transport of fuel and fission product species in pumped sodium loops

3. EXPERIMENT DESCRIPTION

The experiment consisted of exposing a defected mixed-oxide fuel pin section, which had been preirradiated to $\sim 41,000$ MWd/Te as a part of fuel pin F2U¹, to flowing sodium in a miniature pumped sodium loop. A schematic diagram of the loop is shown in Figure 3-1(a). A photo of the loop is shown in Figure 3-1(b).

The loop was filled with sodium and operated for 20 hours prior to inserting the fuel section in the specimen chamber. During this period, oxygen was gettered from the sodium by flowing the sodium through a cartridge of zirconium chips placed in the specimen chamber. At the end of the 20-hour period, the zirconium getter cartridge was removed and the fuel section was placed in the specimen chamber. The loop was operated for 18 hours with the fuel specimen in place. Sodium flow within the loop was then observed to cease, and the test was terminated.

Tables 3-1 and 3-2 list the fuel and loop operating parameters. A detailed description of the loop, test sample, and loop operation is given in the Appendix.

Table 3-1
TEST SAMPLE PARAMETERS

Fuel (Pre-irradiation)

Pu/(Pu + U)	0.20
U-235/(U-235 + U-238)	0.93
Stoichiometry	1.98
Density (% T.D.)	89.6
Diametral Gap (in.)	0.0034

Cladding

Material	347 SS
Nominal Inside Diameter (in.)	0.220
Cladding Wall Thickness (in.)	0.015

Defect

Average Width (mils)	18
Average Length (in.)	2.75

Table 3-2
LOOP OPERATING PARAMETERS

Maximum Sodium Temperature	$\sim 1260^{\circ}\text{F}$
Sodium Temperature at Test Sample	$\sim 1145^{\circ}\text{F}$
Loop ΔT	700°F
Sodium Flow	0.1 gpm
Weight of Sodium in Loop	860 gm
Oxygen Content of Sodium (Estimated)	Initial 20 ppm Final 220 ppm
Velocity of Sodium Past Defect Slot	1.5 to 7 ft/sec
Elapsed Test Time	18 hours

Figure 3-1A. Schematic Diagram of Sodium Loop

Figure 3-1B. Photograph of Sodium Loop in Hot Cell

4. POST-TEST LOOP EXAMINATION

Upon termination of the test, the test sample was removed and the loop was disassembled for examination. Examination of the test sample consisted of: (1) neutron radiography, (2) post-test dimensional measurements, (3) sectioning and macroscopic examination for evidence of fuel loss, and (4) chemical analysis of test sample and control sample for selected fission products.

Figure 4-1 shows a neutron radiograph of the test sample. The view is 90 deg to the defect slot machined in the cladding. No loss of fuel could be distinguished. Some fuel or fuel-sodium reaction product was observed in the slot in the cladding.

After removing the fuel section from the specimen chamber and draining the sodium from the loop (but prior to disassembly of the loop), the gross radioactivity in the loop was measured using a shielded dose rate meter probe. The measured activity in various regions of the loop after the test is shown in Figure 4-2.

The highest activity occurred in the heater of the sample exposure section. During disassembly, however, it was found that this portion of the loop had not drained. Examination subsequent to removal of sodium from the heater showed that it did not contain a significantly greater amount of activity than other sections of the loop, indicating that the fission products in the sodium were the cause of the high activity originally detected.

Examination of the loop components consisted of leaching the inner surfaces with nitric and hydrochloric acids. The sections of the loop selected for analysis are shown in Figure 4-3. Gamma pulse height analysis was used to determine the character of the fission products present in the solutions. Separate analyses for beta emitters such as Pm-147 and Sr-90 were made, using solvent extraction methods followed by beta counting by liquid scintillation. Uranium and plutonium were separated from the sodium and fission products present in the samples by ion exchange methods. Plutonium was determined by alpha counting, and uranium by mass spectrometry, using a known U-233 spike.

Of the fission products analyzed, Cs-137 comprised more than 90% of the total atoms detected. Of the isotopes dissolved in the sodium, Cs-137 accounted for more than 99.9% of the total.

Figure 4-1. Post-Test Neutron Radiograph of Test Sample

Figure 4-2. Relative Activity in Loop

Figure 4-3. Key to Loop Sectioning and Sampling

5. DISCUSSION

5.1 TEST SAMPLE

5.1.1 Fuel Loss

Figure 5.1 shows the pre and post test cladding diameter measurements of the test sample. The greatest diametral increases occurred at 90 deg to the defect slot in the cladding. The average cladding diameter increase was 3%. These data indicate that reaction occurred between the fuel and the sodium during the test.

The measured diametral increase closely agrees with that predicted from previous experimental data^{2,3} where the oxygen content of the sodium was greater than that required to cause reduction of the fuel sodium reaction product, 70 to 130 ppm⁴. Accounting for the oxygen initially in the sodium, the oxygen adsorbed on the surfaces of the loop, and the oxygen in the fuel section, it is estimated that the maximum amount of oxygen that could have been in the sodium in the loop was ~ 220 ppm. In an operating reactor where the oxygen content of the sodium will be maintained at a low value (< 10 ppm), the extent of the fuel sodium reaction and the resulting increase in fuel pin diameter would be expected to be less than that which occurred in this test.

The post test profile of the fuel section diameter shown in Figure 5.1 shows that very little increase in diameter occurred in the region where the sodium flow was directed at the fuel from the nozzle (see Figure 5.2). It is hypothesized that the diameter change was decreased in this region as a result of the sodium fuel reaction product's being removed by the sodium stream as it was formed. Examination after the test showed that there was very little fuel sodium reaction product present in this region of the fuel section.

Although a quantitative value cannot be placed upon the amount of fuel lost, an upper limit estimate based on metallographic examination of the fuel indicated that no more than 1% of the fuel was lost. Leach analyses of the loop sections confirmed this observation as only minor amounts of fuel were found deposited throughout the loop.

5.1.2 Fission Product Loss from the Fuel

Table 5.1 shows the post test analysis of the test sample for selected fission products. These fission products are assumed to be representative of the remaining fission products in the sample. Also shown in Table 5.1 are values determined from a control sample taken immediately adjacent to the test sample. Comparison of the two analyses shows that a significant portion of the total inventory of each fission product isotope was lost during the 18 hours of loop operation. Column 4 in Table 5.1 shows the percent loss measured for each isotope.

Comparison of the present data was made with a failed fuel pin which operated for 80 to 85% of its irradiation in the failed condition to a burnup of 53,000 MWd/Te⁵. That fuel pin was contained in a capsule which permitted NaK to circulate past the fuel pin at velocities of 0.5 to 0.75 ft/sec. Subsequent examination of the fuel pin showed multiple cracks along the length of the pin so that intimate contact probably occurred between the NaK and the fuel. During reactor shutdown, no flow of NaK past the pin occurred.

About 15% of the cesium produced in the fuel of the pin was found to be dissolved in the NaK. This value is significantly smaller than the 66% measured in the present test. These data indicate that high release rates such as those measured in the present test do not occur unless sufficient coolant flow and liquid metal accessibility to the fuel is present.

The foregoing data show that fission product losses can be very significant when sodium coolant is free to penetrate throughout the fuel, as can occur in failed elements during reactor shutdown. Thus, to obtain true fission product release values, the exposure of the failed element to the flowing sodium both during the reactor shutdowns and during power operation must be considered.

5.2 FISSION PRODUCT DEPOSITION IN THE LOOP

The fission products lost from the fuel were found both in the bulk sodium and in deposits on the loop walls. Shown in Table 5.2 are the concentrations of fission products found in the bulk sodium, and these values are compared with the fission products lost from the fuel. Note that only cesium dissolved in the sodium in any appreciable quantity. This is in agreement with observations by the Russians in the BR-5 reactor.⁶

Table 5.3 shows the total amount of fuel and fission products contained in the residual sodium and plated out on the loop walls in the various sample locations shown in Figure 4.3. To separate the actual amount of fission products deposited on the loop walls from the gross amount reported in Table 5.3, the fission product activity in the residual sodium must be subtracted. Two methods for estimating this activity were employed, and the results are given in Table 5.4. For samples 1A, 1B, 1C, 1D, and 4, actual sodium masses in the samples were measured and then multiplied

by the fission product concentration values in Table 5-2 to yield the total fission product atoms in the sodium. Within the accuracy of the techniques ($\sim 2X$), the results provided further evidence that the predominant Cs activity remained in the sodium (see Tables 5-3 and 5-4). Because of this, the measured Cs-137 activities on the remaining samples (2A, 2B, 2C, 3, 5, 6A, 6B, 7) were used to indicate the residual sodium mass present in the samples instead of by direct measurement. Again applying the data of Table 5-2, fission product activity in the bulk sodium was computed.

Subtracting the fission product species in the sodium only (Table 5-4) from the species in the sodium, plus those plated out on the loop walls (Table 5-3), yields the plate-out on the loop walls. This is shown in Table 5-5.

Note from the foregoing deposition data that plate-out occurs on the cooler surfaces of the loop, with the greatest plate-out occurring at the start of the downward temperature gradient, i.e., samples 2A, 2B, and 2C (see Figure A-5 in Appendix for temperature distributions around the loop). This is in general agreement with the tracer level work of Brehm, et al.,⁷ and also follows the pattern of corrosion product deposition observed by Rowland, et al.⁸

Application of these observations to LMFBR plant design suggests that fission products will concentrate in the primary system cold trap and in the cooler portions of the intermediate heat exchanger. Adequate provision for shielding and/or effective contamination procedures should be developed as part of the overall maintenance plan for these components.

The initial uranium-to-plutonium ratio of the fuel was 4 to 1. However, the ratio of uranium to plutonium found in the leach samples throughout the loop varied from 1643 to ~ 42 to 1. These data suggest uranium was preferentially leached from the fuel compared to plutonium. It is important to recall that less than 1% of the fuel was lost from the test specimen. However, these data suggest the possibility that, over long periods of time, the local enrichment of failed fuel pins may vary. Further development work is required in this area.

Figure 5-1. Pre- and Post-Test Cladding Diameter Measurements of Test Sample

Figure 5-2. Post-Test Appearance of Test Sample

Table 5-1
FISSION PRODUCTS IN TEST AND CONTROL SAMPLES
ATOMS/ATOM U + Pu

Isotope	Test Sample After Exposure to Loop	Control Sample	Percent Loss from Test Sample ^(a)
Cs-137	9.5×10^{-4}	2.4×10^{-3}	66
Ce-144	4.0×10^{-5}	5.1×10^{-5}	32
Ru-106	9.8×10^{-6}	5.5×10^{-5}	85
Zr-95	$<3.0 \times 10^{-7}$	$<3.0 \times 10^{-7}$	0
Sr-90	9.2×10^{-4}	1.1×10^{-3}	29
Pm-147	1.3×10^{-4}	1.6×10^{-4}	32

(a) Based on the gamma scan of the fuel pin, the ratio of activity in the test sample to that in the control sample was 1.16 to 1, respectively. Control sample values were then normalized to the test sample values and the percent loss was computed.

Table 5-2
RETENTION OF FISSION PRODUCTS IN SODIUM

Specie	Concentration (atoms/gm Na)	Total Atoms in Sodium (atoms)	Total Atoms Lost from Fuel (atoms)	Retention in Sodium (%)
Cs-137	2.7×10^{16}	2.3×10^{19}	4.6×10^{19}	50
Sb-125	6.6×10^{13}	5.6×10^{16}	(a)	
Cs-134	5.1×10^{13}	4.3×10^{16}	(a)	
Pm-147	2.5×10^{13}	2.2×10^{16}	6.5×10^{18}	0.34
Sr-90	1.4×10^{13}	1.2×10^{16}	5.0×10^{19}	0.024
Ru-106	$<1.3 \times 10^{11}$	$<1.1 \times 10^{14}$	4.7×10^{17}	<0.023
Ce-144	$<5.2 \times 10^{10}$	$<4.5 \times 10^{13}$	1.9×10^{18}	<0.0024
Zr-95	7.3×10^9	6.2×10^{12}	(a)	
Nb-95	3.4×10^9	2.9×10^{12}	(a)	
Total U	6.6×10^{15}	5.6×10^{18}	(a)	
Total Pu	1.4×10^{13}	1.2×10^{16}	(a)	

(a) Below limits of detection

Table 5-3
TOTAL ATOMS FROM LOOP SAMPLES

Isotope	Sample Number ^(a)					
	1A	1B	1C	1D	2A	2B
Total U	7.1×10^{16}	1.9×10^{17}	1.6×10^{17}	2×10^{16}	9.2×10^{18}	1.2×10^{18}
Total Pu	3.9×10^{15}	1.3×10^{16}	4.6×10^{15}	6.2×10^{15}	5.6×10^{15}	6.0×10^{15}
Sb-125	4.8×10^{14}	3.2×10^{15}	1.4×10^{14}	1.3×10^{14}	-	-
Cs-137	2.2×10^{17}	7.9×10^{17}	3.5×10^{16}	2.5×10^{16}	2.0×10^{15}	4.6×10^{14}
Cs-134	1.6×10^{14}	6.1×10^{14}	3.5×10^{13}	2.7×10^{13}	-	-
Cs-144	1.6×10^{13}	1.6×10^{13}	6.3×10^{13}	4.2×10^{13}	1.7×10^{13}	1.9×10^{13}
Zr-95	1.7×10^{12}	2.7×10^{11}	2.1×10^{12}	2.6×10^{12}	2.3×10^{12}	2.2×10^{12}
Nb-95	1.1×10^{12}	3.2×10^{11}	2.0×10^{12}	1.4×10^{12}	-	-
Ru-106	8.7×10^{12}	1.0×10^{13}	3.6×10^{14}	2.9×10^{13}	3.3×10^{13}	2.9×10^{13}
Pm-147	2.3×10^{14}	3.2×10^{14}	5.3×10^{14}	8.5×10^{14}	5.8×10^{13}	8.5×10^{13}
Sr-90	2.6×10^{14}	5.9×10^{14}	9.1×10^{14}	1.2×10^{15}	3.2×10^{14}	2.7×10^{14}

(a) Sample location shown in Figure 4-3

ND = Not Detected

Table 5-3 (Continued)

2C	3	4	Sample Number ^(a)		6A	6B	7
			5				
2.6×10^{17}	1.7×10^{17}	$<2.1 \times 10^{16}$	1.3×10^{17}		$<2.1 \times 10^{16}$	$<2.0 \times 10^{16}$	$<2.0 \times 10^{16}$
2.0×10^{15}	3.5×10^{14}	2.6×10^{14}	4.4×10^{13}		5.2×10^{12}	4.5×10^{14}	9.2×10^{14}
-	-	-	-		-	-	-
7.1×10^{15}	2.2×10^{16}	1.1×10^{16}	2.3×10^{16}		2.4×10^{16}	2.1×10^{15}	7.3×10^{13}
-	-	-	-		-	-	-
1.9×10^{13}	1.6×10^{12}	6.6×10^{12}	ND		ND	8.4×10^{12}	2.4×10^{12}
2.8×10^{12}	1.3×10^{11}	5.8×10^{11}	ND		ND	6.2×10^{11}	1.2×10^{11}
-	-	-	-		-	-	-
1.6×10^{13}	8.4×10^{11}	ND	ND		ND	5.2×10^{12}	2.2×10^{12}
6.4×10^{13}	9.9×10^{12}	9.5×10^{12}	7.1×10^{12}		4×10^{12}	2.1×10^{13}	4.3×10^{12}
2.2×10^{14}	3.5×10^{13}	2.0×10^{14}	1.3×10^{13}		5.7×10^{12}	7.4×10^{13}	2.5×10^{13}

Table 5-4
TOTAL ATOMS RETAINED IN BULK SODIUM PORTION OF SAMPLES

Isotope	Sample Number ^(a)					
	1A	1B	1C	1D	2A	2B
Total U	8.7×10^{16}	1.5×10^{17}	1.5×10^{16}	2.6×10^{15}	4.9×10^{14}	1.1×10^{14}
Total Pu	1.8×10^{14}	3.1×10^{14}	3.2×10^{13}	5.6×10^{12}	1.0×10^{12}	2.4×10^{11}
Sb-125	8.7×10^{14}	1.4×10^{15}	1.5×10^{14}	2.6×10^{13}	-	-
Cs-137	3.5×10^{17}	5.9×10^{17}	6.3×10^{16}	1.1×10^{16}	$2.0 \times 10^{15} \text{ (b)}$	$4.6 \times 10^{14} \text{ (b)}$
Cs-134	6.7×10^{14}	1.1×10^{15}	1.2×10^{14}	2.0×10^{13}	-	-
Ce-144	$<6.8 \times 10^{11}$	$<1.1 \times 10^{12}$	$<1.2 \times 10^{11}$	$<2.1 \times 10^{10}$	$<3.8 \times 10^9$	$<8.8 \times 10^8$
Zr-95	9.5×10^{10}	1.6×10^{11}	1.7×10^{10}	2.9×10^9	5.4×10^8	1.2×10^8
Nb-95	4.5×10^{10}	7.4×10^{10}	7.8×10^9	1.4×10^8	-	-
Ru-106	$<1.7 \times 10^{12}$	$<2.8 \times 10^{12}$	$<3.0 \times 10^{11}$	$<5.2 \times 10^{10}$	$<9.6 \times 10^9$	$<2.2 \times 10^9$
Pm-147	3.3×10^{14}	5.5×10^{14}	5.7×10^{13}	1.0×10^{13}	1.9×10^{12}	4.3×10^{11}
Sr-90	1.9×10^{14}	3.2×10^{14}	3.2×10^{13}	5.6×10^{12}	1.0×10^{12}	2.4×10^{11}

(a) Sample location in Figure 4-3

(b) Assumed to be same as measured value in Table 5-3

ND = Not Detected

Table 5-4 (Continued)

Sample Number ^(a)						
2C	3	4	5	6A	6B	7
1.7×10^{15}	5.4×10^{15}	2.7×10^{15}	5.6×10^{15}	5.9×10^{15}	5.1×10^{14}	1.8×10^{13}
3.7×10^{12}	1.1×10^{13}	5.7×10^{12}	1.2×10^{13}	1.2×10^{13}	1.1×10^{12}	3.8×10^{10}
-	-	-	-	-	-	-
$7.1 \times 10^{15} (b)$	$2.2 \times 10^{16} (b)$	1.1×10^{16}	$2.3 \times 10^{16} (b)$	$2.4 \times 10^{16} (b)$	$2.1 \times 10^{15} (b)$	$7.3 \times 10^{13} (b)$
-	-	-	-	-	-	-
$<1.4 \times 10^9$	$<4.2 \times 10^{10}$	$<2.1 \times 10^{10}$	ND	ND	$<4.1 \times 10^9$	$<1.4 \times 10^8$
1.9×10^9	5.9×10^9	3.0×10^9	ND	ND	5.7×10^8	2.0×10^7
-	-	-	-	-	-	-
$<3.4 \times 10^{10}$	$<1.1 \times 10^{11}$	ND	ND	ND	$<1.0 \times 10^{10}$	$<3.5 \times 10^8$
6.6×10^{12}	2.0×10^{13}	1.0×10^{13}	2.1×10^{13}	2.2×10^{13}	2.0×10^{12}	6.8×10^{10}
3.7×10^{12}	1.1×10^{13}	5.7×10^{12}	1.2×10^{13}	1.2×10^{13}	1.1×10^{12}	3.8×10^{10}

Table 5-5
CALCULATED DEPOSITION OF SPECIES
PER SQUARE INCH OF LOOP SAMPLE ($\times 10^{-12}$)

Isotope	Sample Number ^(a)												
	1A	1B	1C	1D	2A	2B	2C	3	4	5	6A	6B	7
Total U	†	4800	2200	4100	9.2×10^6	1.2×10^6	2.6×10^5	1.7×10^5	$<2.1 \times 10^4$	1.3×10^5	$<2.1 \times 10^4$	$<2 \times 10^4$	$<2 \times 10^4$
Total Pu	880	1500	690	1400	5600	6000	2000	2200	260	440	0	480	920
Sb-125	†	210	†	25	—	—	—	—	—	—	—	—	—
Cs-137	†	†	†	†	(b)	(b)	(b)	(b)	†	(b)	(b)	(b)	(b)
Cs-134	†	†	†	†	(b)	(b)	(b)	(b)	†	(b)	(b)	(b)	(b)
Cs-144	3.8	1.9	9.5	9.5	12	18	24	1.1	5.7	ND	ND	10	3.9
Zr-95	0.41	0.013	0.32	0.59	1.6	2.0	3.6	0.09	0.5	ND	ND	0.74	0.2
Nb-95	0.26	0.030	0.30	0.32	—	—	—	—	—	—	—	—	—
Ru-106	2.1	1.2	54	6.6	23	27	20	0.5	ND	ND	ND	6.2	3.6
Pm-147	†	†	71	190	38	79	73	†	†	†	†	22	7.1
Sr-90	12	33	140	270	220	250	280	†	167	†	†	75	41
Surface Area (in. ²)	4.2	8.4	6.6	4.4	1.44	1.08	0.78	1.44	1.15	1.22	1.22	0.84	0.61
$\frac{U}{Pu}$	—	3.2	3.1	2.9	1643	200	130	77	<81	295	—	<42	<22

(a) Sample location shown in Figure 4-3

(b) Results normalized to cesium

† Essentially no deposition observed within experimental error

ND = Not Detected

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APPENDIX. EXPERIMENT DESCRIPTION

A.1 LOOP

The test loop consisted of the following components: electromagnetic (E.M.) pump, flowmeter, sodium sample tubes, sample exposure section (including heaters and sodium purification equipment), cooler assembly (fin tube cooler, cowling, blower, and air damper), and dump tank. A schematic diagram of the loop is shown in the preceding text (Figure 3-1A).

Because of the high activity level of the fuel sample to be used in the test, remote operation of the loop was required. The loop was mounted inside a steel containment box which provided containment for the alpha-particle-emitting material in the loop in the event of failure during operation and also during subsequent disposal of the loop. The containment box can be seen in Figure A-1. The containment box was placed inside one of the radiation locks to the hot cells at the General Electric Radioactive Materials Laboratory. This permitted use of the interlock manipulators for remote operation of the loop. Pertinent details of the loop are described in the following paragraphs.

The sample section of the loop was provided with a hoist assembly by means of which a canister of zirconium chips was lowered into position (Figure A-2).

Only a limited quantity of heat could be dumped inside the containment box without exceeding the permissible temperature limit of the manipulators. Loop sodium flow was therefore limited to 0.1 gpm. Because of the low flow rate, a nozzle arrangement was used to obtain a more realistic sodium velocity past the cladding defect region of the test sample (for test sample details, see subsection A.2 and Figure 5-2). A system of baffles was also placed inside the finned cooler to facilitate deposition of fuel and fission products released from the fuel during the test.

All loop tubing and component parts were rinsed in acetone prior to assembly to remove any extraneous particles and grease present. Major loop components were connected to the loop tubing by means of Swagelok tube fittings. This facilitated disassembly of the loop upon completion of the test.

Loop components were preheated prior to sodium filling by means of flexible heating tape wound around the outside surfaces. The heating tape was wound in sections to permit individual temperature control of different portions of the loop. During loop operation, the heating tape was used to reduce heat loss through the loop walls, since insufficient space was available to permit the loop tubing to be heavily insulated. The cooler was preheated by means of two strip heaters placed within the cowling. The E.M. pump contained its own integral heater. Two infra-red heat lamps were placed within the containment box to provide additional heating capacity when required. Loop temperatures were monitored by means of 24 thermocouples.

The holder containing the test sample was attached to the hoist assembly by means of a snap lock device. The hoist assembly was then bolted to the top of the sample exposure section by means of a flange (O-ring seal). Leaktightness was checked by evacuating the loop and monitoring to see that no inleakage of air occurred. To further ensure no leakage of air into the loop during operation of the test, a flow of argon (15 cc/min) to the plenum region of the sample exposure section was maintained at all times during the test. Minor traces of oxygen were removed from the argon prior to the loop by passing the argon over copper turnings heated to 1200°F. The argon was dumped from the loop to the containment box by means of a bubble jar containing mercury.

A.2 TEST SAMPLE

The test sample consisted of a 3-inch section removed from fuel pin F2U¹. See Table 3-1 in preceding text for a listing of the initial fuel pin fabrication parameters.

Fuel pin F2U was irradiated in EBR-II to 40,800 MWd/Te burnup. Figure A-3 shows the post-irradiation gross gamma scan of fuel pin F2U. Marked on the gamma scan is the location of the test sample and control sample used in the test. The fuel had been removed from the reactor for 2 years prior to the testing and contained only the longer-lived fission products.

For the test, a slot 2-3/4 inches long was cut through the cladding of the test sample, using a 0.010-inch-thick cutting wheel. A slight springing-open of the cladding at the slot occurred so that the actual area of the slot through the cladding was 0.046 in.² (average width of opening 0.018 inch). In cutting the slot in the cladding, a groove ~0.015 inch deep was made in the fuel surface.

The sample holder was provided with caps which sealed the ends of the test sample. Any fuel and fission product loss from the sample was essentially limited to the defect slot by the sample holder. Figure A-4 shows the test sample mounted in the holder.

A.3 SODIUM FILL

Prior to sodium fill, the loop components and lines were preheated to 350°F, and the loop was evacuated. This process was continued for 4 hours to remove any acetone residue and adhering gases from the inner surfaces of the loop prior to sodium fill. Upon completion of the bake-out, the loop was back-filled with argon.

Sodium filling of the loop was accomplished by pressuring the sodium dump tank and venting the plenum region at the top of the sample test section. By monitoring the loop thermocouples, the progress of the sodium into the loop could be followed. The correct sodium fill level was obtained using a series of five thermocouples at the top of the sample exposure section.

A.4 LOOP OPERATION

After bake-out and filling of the loop with sodium, purification of the sodium was initiated. Sodium was circulated through the zirconium-chip cartridge for 20 hours at 975°F.

Prior to introducing the test sample into the sodium, a sample of loop sodium was obtained by means of the sample tube provided (see Figure 3-1A in Section 3). This sample was to be analyzed to determine the oxygen content of the sodium at the start of the test. However, subsequent analysis of the sodium indicated that alpha contamination had occurred sometime during sodium fill or sodium purification. Since facilities available for analyzing sodium for oxygen could not handle alpha-contaminated samples at the time the experiment was conducted, no oxygen analysis was obtained.

Because of the failure of the heating tape at the plenum region of the sample test section during sodium purification, isothermal sodium temperatures in the plenum and at the entrance of the cooler could not be maintained. Therefore, fission products were free to deposit ahead of the cooler. Figure A-5 shows the temperatures maintained in various regions of the loop during operation. Refer to Table 3-2 in Section 3 for the loop operating parameters during the test. Although the test was planned to run for 100 hours, after 18 hours of operation, flow was observed to stop in the loop. Attempts to restart the flow were unsuccessful and the test was terminated. During disassembly, it was found that significant quantities of sodium had frozen on the underside of the hoist assembly in the region of the flange. The loss of the heater tape in the region of the plenum allowed the temperature at the bottom of the hoist to drop below the melting point of the sodium and act as a condenser, thus removing sodium from the circulating inventory. The loss of sodium flow in the loop was attributed to the sodium level's dropping below the outlet tube at the top of the sample exposure section.

Figure A-1. Loop Containment Box

Figure A-2. Positioning of Gettering Cartridge and Fuel Sample During Sodium Purification and Test

Figure A-3. Gross Gamma Scan of Fuel Pin F2U Showing Location of Test and Control Samples

Figure A-4. Test Sample Mounted in Sample Holder

Figure A-5. Temperatures Maintained in Various Regions of Loop During Operation

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