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Gulf General Atomic

AEC RESEARCH AND
DEVELOPMENT REPORT

GA-10264

HIGH-TEMPERATURE FAST-FLUX IRRADIATION EXPERIMENT FOR
MIXED-OXIDE FUEL RODS

Prepared under
Contract AT(04-3)-167
Project Agreement 23
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

October 15, 1970

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Gulf General Atomic

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ABSTRACT

The GGA fast-flux experiment in EBR-II will study fuel-rod specimens with cladding temperatures in the range of 650°C to 800°C at higher temperatures not heretofore covered in the LMFBR program and at linear power-generation rates of 14 to 16 kW/ft. The mixed oxide fuel is contained in 20% cold-worked 316 stainless steel and will achieve burnups ranging from 2.7 to 10.7% FIMA. The capsules are generally of the B-7B type with a stainless steel annular thermal barrier sodium bonded to the fuel rod and to the capsule tube. Silicon-carbide temperature monitors are placed in various positions along the thermal barrier to allow postirradiation determination of capsule temperatures utilizing dimensional changes during isochronal annealing of the SiC. Activated charcoal has been placed in all of the fuel rods to determine the effects of fast neutron irradiation on the sorption characteristics of the charcoal for fission products. All eight fuel rods contain sealed charcoal capsules for postirradiation sorption experiments. Three of the fuel rods also contain charcoal traps connected to the fueled region to study the fission-product distribution in the trap under reactor operating conditions.

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1. F-1 EXPERIMENT DESCRIPTION

A. PURPOSE AND OBJECTIVES

The general objective of the Gulf General Atomic (GGA) F-1 experiment in the EBR-II fast-flux environment is to study fuel-rod specimens over a wider range of temperatures and to a higher range of cladding temperatures than heretofore carried out in the Liquid Metal Fast Breeder Reactor (LMFBR) program. Variables that will be kept constant are the cladding material (316 stainless steel, 20% cold-worked), the fuel (mixed (Pu-U)O₂), and the fuel fabrication technique. These fixed test conditions are summarized in Table 1. The nominal range of temperatures and heat-generation rates to be studied and the purpose of each set of experimental conditions are listed in Table 2. The temperature range will overlap that already being studied under the LMFBR program to provide a point of commonality for comparison of results. The reference condition selected is the same as that being used in the thermal irradiation of GGA's Capsule 04-P9 in the Oak Ridge Research Reactor (ORR)⁽¹⁾; this will provide a direct comparison of a fast versus a thermal neutron test environment. Interim nondestructive tests are planned as well as destructive post-test examinations to study the influence of the test variables on integral rod behavior and individual components, including the charcoal-loaded fission-product traps. Passive temperature and flux monitors are to be extensively used to define the actual experimental environment.

Approval in principle for the proposed experiments was granted by the Atomic Energy Commission on February 12, 1970. An amendment to the condition of approval relative to the inclusion of fission-product traps was granted May 7, 1970.

The F-1 experiment has been assigned subassembly designation X094 by the EBR-II Experiment Manager.

Table 1

FIXED TEST CONDITIONS FOR THE INITIAL F-1 EXPERIMENT

Cladding	316 SS, 20% cold-worked
Cladding OD	0.300 in.
Cladding OD/ID ratio	~1.15
Linear power generation	~15 kW/ft
Fuel pellets, sol-gel derived	85 wt-% UO_2 , 15 wt-% PuO_2
Fuel smear density	85% TD
External pressure	Ambient
Internal helium pressure, hot	~28 psia

B. DESIRED IRRADIATION CONDITIONS1. Reactor Location

Capsules G-1 through G-7 are to be inserted into a Mark B-7B subassembly. It is desired that this subassembly be irradiated in Sector E, Row 7, Position 4 (7E4) of the EBR-II. After capsule G-3 has received an exposure equivalent to 138 full power reactor days at 62.5 MW, it will be replaced by capsule G-8. The F-1 experiment is designed for the 7E4 reactor core position in EBR-II. Preliminary calculations also indicate that the F-1 experiment, loaded with capsules G-1 through G-7, will not depress the EBR-II flux in the 7N4 area, which was a point raised in the EBR-II Newsletter, April 17, 1970. If a change of position is imperative, the 7N3 position would be acceptable.

2. Coolant Flow Rate

The coolant flow rate for the Mark B-7B subassembly is 17 gpm for an average specific heat of 0.305. Flow strips (see Fig. 1) have been installed within the subassembly to regulate sodium flow around each capsule for outlet-temperature equalization.

Table 2
FAST-FLUX IRRADIATION CAPSULE LOADINGS

Capsule	Temperature (°C) ^a		Design Heat-generation Rating (kW/ft)	Burnup (MWd/tonne x 1000)				Purpose
	Nominal	Design		25	50	75	100	
Test Condition								
G-3	700	702	15.5	////				Temperature calibration
G-6	700	721	14.4	////	////			Fast flux - thermal flux comparison (reference)
G-4	700	708	14.9	////	////	////	////	Burnup effect
G-7	600	612	13.8	////	////			Overlap with LMFBR and temperature effect
G-5	650	669	14.3	////	////			Temperature effect
G-2	750	753	15.5	////	////			Temperature effect
G-1	800	783	16.1	////	////			Temperature effect
First Replacement ^b								
G 8	700	702	15.5		////	////	////	Burnup effect
Planned Second Replacement ^c								
	(d)	15				////	////	Reproducibility at new reference
	(d)	15				////	////	Reproducibility at new reference
	(d)	15 + x ^e				////	////	Influence of higher heat-generation rate
	(d)	15 - x ^e				////	////	Influence of lower heat-generation rate
	(d) + 50	15 + x ^e				////	////	Influence of higher temperature plus higher heat-generation rate

^aThese temperatures are for the outer surface of the cladding.

^bDoes not get benefit of feedback from experiment.

^cGets feedback benefit.

^dNew reference temperature, based upon experimental results.

^eValue of x will be chosen to give reasonable variation from reference heat-generation rate.

Fig. 1 Flow strip for subassembly hex container

3. Flux Levels and Heat Ratings

The linear heat ratings ($\pm 10\%$), fuel burnup, and reactor residence times are:

Capsule	Peak Heat Ratings Beginning of Life (kW/ft)	Reactor Days (Full Power 62.5 MW)	Burnup	
			%FIMA Total	U + Pu (MWd/Te)
G-1	16.1	271	5.3	50,000
G-2	15.5	280	5.3	50,000
G-3	15.5	138	2.7	25,000
G-4	14.9	600	10.7	100,000
G-5	14.3	303	5.3	50,000
G-6	14.3	303	5.3	50,000
G-7	13.8	316	5.3	50,000
G-8	15.5	427	8.0	75,000

4. Reactor Startup Procedure

There is only one specific requirement for any special startup rate for the experiment: the fuel must come up to temperature slowly enough to allow fuel densification and restructuring (to improve the thermal conductivity) and thus preclude center melting of the fuel at full power. Most of such restructuring occurs within a few hours and therefore a normal EBR-II startup with an advance to power in increments of ~ 5 MW every 1-1/2 to 2 hr (indicated in Fig. 68, Run 39, Report of EBR-II Operations, October 16, 1969 through January 28, 1970) will be satisfactory, provided the reactor power rate increase does not exceed that for Run 39.

5. Shutdown Requirements

During the final shutdown before each capsule is removed from the core, the following conditions should apply to prevent the loss of information stored in the silicon carbide temperature monitors:

For three full power days immediately before shutdown, the power level of the reactor should be held at a constant level close to the mean power level during the entire irradiation period.

The shutdown should be rapid. During this shutdown period, the capsule should not be subjected to a total fluence greater than 1×10^{19} n/cm² or the accumulated dose received during 3 hr at full power. The temperature of the sodium in the outlet channel of the subassembly should drop to 700°F within the first hour after shutdown begins.

C. DESIRED OUT-OF-REACTOR SERVICES

Special Requirements for Fabrication of Subassembly

1. Flow Strips

The Mark B-7B subassembly hex container has flow strips installed to regulate the flow in each channel of the subassembly so that the outlet temperature from each channel will be uniform. These flow strips keep the sodium temperature of each outlet channel below 900°F, not including the mixing that occurs along the capsule lengths caused by the spiral wire wraps.

The flow strips (see Fig. 1) are made of 304L stainless steel; one flow strip is welded to the inside face of each of the hex flats. The flow strips have been installed by GGA as shown in Fig. 2.

2. End Fitting

The bottom tip element shown in Fig. 3 has been fabricated by GGA. Usually, this end fitting is furnished by the EBR-II facility, but it was necessary to change the angle of pitch on the tip to 45° instead of 30°; this change has been accepted by the EBR-II Project.

3. Capsule Handling

X-ray radiographs of fabricated fuel rods indicated that 2 capsules (G-4 and G-6) out of 8 had pellet separations due to handling the rods in the upside down position. This was corrected by holding the rod in an upright position and tapping it gently on the floor. After the capsules have been received and handled at EBR-II and have been radiographed, those capsules showing pellet separation should be treated as above and X-ray

Fig. 2 Subassembly hex container with flow strips installed

radiographed again. After the pellet separation is corrected, the capsules should remain in an upright position until they are inserted in the EBR-II core.

4. Wire Wrapping of Capsules

A wire must be helically wrapped along the length of each capsule (G-1 through G-8) to space the capsules properly in the subassembly. The wire, which is indicated in Fig. 4 and is identical to that shown on the other seven capsule drawings, is one continuous 0.020-in.-diam wire on each capsule. The EBR-II Project will install the wire on each of the eight capsules, as shown on the capsule assembly drawings, using wire supplied by Argonne National Laboratory but redrawn by GGA to the proper size.

5. Grid Plate

Capsules G-1 through G-8 have a smaller outside diameter, 0.026 in. than the regular Mark B-7B capsule. The smaller diameter of the GGA capsules will cause the center of each capsule to be in a different location than that of the usual Mark B-7B capsule, which means that the normal grid plate that holds the capsules will not fit GGA capsules G-1 through G-8. A new grid plate will be fabricated if necessary by the EBR-II Project with the centerlines as shown on Fig. 5. This description is also shown on all capsule G-1 through G-8 assembly drawings.

6. Flow Test

The Mark B-7B subassembly has flow channels in it that are different from any other subassembly in the EBR-II. The flow strips and capsule diameter and wire wrap size are specifically for this subassembly. The flow requirement for the F-1 experiment is, with the sodium at 0.304 specific heat, a flow of 17 gpm to achieve the cladding temperature required for the experiment. A flow test, which will be performed by the EBR-II facility, is necessary to confirm these conditions with the required 40.8 ΔP and selected orifice adaptor.

Fig. 4 Typical capsule with fission-gas plenums

LOCATION IN CORE - SECTOR "E" - 7E4

Fig. 5 Capsule positions and markings for the F-1 subassembly X094

D. IDENTIFICATION OF CAPSULES

Each of the eight capsules, G-1 through G-8, is clearly marked by die stamping with an identification number. Each capsule is also die stamped with an orientation mark to show how the capsule must be placed in the sub-assembly in relation to the center of the reactor core.

The identification mark, such as "G-1," and the dosimeter or orientation mark, such as "DOS," are inscribed together on the top of the plug end. The same marking is die stamped on the bottom of the capsule along the side of the tip element, which also must face the center of the core.

E. SUBASSEMBLY LOADING DESCRIPTION

The F-1 experiment consists of seven capsules inside a hex containment can (see Fig. 2). Each of the seven capsules has a 0.020-in.-diam wire helically wrapped along the length of the capsule.

The arrangement of the seven capsules is shown in Fig. 5. An eighth capsule for later insertion is also shown in position on this drawing. The drawing indicates that one set of the G-1 through G-8 markings are on the side of the capsule facing the centerline of the reactor core.

F. FUEL-ROD DESCRIPTION

Sectional views of the capsules are shown in Figs. 4 and 6. The principal differences in the capsules are the inclusion of active charcoal traps in capsules G-4, -6, and -7 (Fig. 6) and empty plenums in the other capsules (Fig. 4) with various thermal barrier thicknesses in the capsules. The description of the fuel rod and capsule components and other details are contained in Table 3.

The active charcoal traps were designed not only to provide ready access of gaseous fission products from the fuel rod to the trap but also to prevent passage of charcoal particles into the fuel rod. The charcoal particles are held in place in the trap by 40-mesh stainless steel screen that has an opening size one-half that of the minimum particle size. After the traps are loaded, they are welded to the clad standoff spacer and to the spacer. The traps were then held with the opening down and no charcoal or

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Table 3

SUMMARY OF FUEL-ROD AND CAPSULE DATA

Fuel-rod designations	G-1 through G-8
Rod type	Sealed, but designed for operation with low pressure differential across the cladding to simulate manifolded type; integral charcoal trap in low-temperature rods

Cladding data (one rod)

Material	316 SS, 20% cold worked
OD, in.	0.300
Wall, in.	0.0185
OD/ID ratio	1.143
Length, in.	23.5
Volume, cm ³	6.48
Weight, g	51.24

Stainless steel reflector and cladding standoff plug (2/rod)

Material	304L
Length, in.	3.61
Diameter, in.	0.263
Volume, cm ³	2.2
Weight, g	17.5

Plenum and trap tube

	G-1,-2,-3,-5,-8				G-4,-6,-7			
	Plenum		Trap		Plenum		Trap	
	Top	Bottom	Top	Bottom	Top	Bottom	Top	Bottom
Volume, cm ³	3.63	3.42		0.455	3.15	3.42	0.443	0.455
Weight, g	28.7	27.06		3.59	24.9	27.06	3.50	3.59
Material	316 SS							

Fuel rod plugs and spacers

Spacer plug	G-1,-2,-3,-8	G-4,-6,-7
	1 req. for each rod	2 req. for each rod
Volume, cm ³	0.435	0.435
Weight, g	3.434	3.434
Material	304L SS	304L SS

Table 3 (continued)

Cap	G-1 through G-8 <u>1 req. for each rod</u>							
Volume, cm ³	0.212							
Weight, g	1.675							
Material	304L SS							
Spacer tube	<u>G-1,-2,-3,-8</u>							
Volume, cm ³	1.51							
Weight, g	11.914							
Material	304 SS							
Thermal barrier tubes								
	<u>G-1</u>	<u>G-2</u>	<u>G-3</u>	<u>G-4</u>	<u>G-5</u>	<u>G-6</u>	<u>G-7</u>	<u>G-8</u>
Material	304	304	304	304	304	304	304	304
Volume, cm ³	202	175.5	123.1	151.4	163	100	65	123.1
Weight, g	1591.2	1382.8	970	1193	1289	790	513	970
Inconel springs, g, (7 rods)					0.827 ea x 7 = 5.789			
Capsule components, G-1 through G-8								
Tip element								
Volume, cm ³					6.48			
Weight, g					51.19			
Material					304L SS			
Adaptor top								
Volume, cm ³					4.11			
Weight, g					32.444			
Material					304L SS			
Plug end								
Volume, cm ³					7.69			
Weight, g					60.722			
Material					304L SS			
Containment and tip adaptor								
Volume, cm ³					5.37			
Weight, g					47.423			
Material					304L SS			
Outer containment tube								
Volume, cm ³					56.2			
Weight, g					443.2			
Material					304L SS			

Table 3 (continued)

Fuel pellets

Dimensions

OD, in.	0.2595
ID, in.	0.058
Length (U,Pu)O ₂ , in.	0.250
End-dish depth, in.	0.006
Total fuel stack height, in.	13.5 (for one rod)
Number of (U,Pu)O ₂ pellets	54 (for one rod)
Material	(U,Pu)O ₂ , sol-gel derived

Composition

UO ₂ , % (93% U-235)	85
PuO ₂ , % (Pu ²³⁸ , < 0.04; Pu ²³⁹ , 88.7; Pu ²⁴⁰ , 9.99; Pu ²⁴¹ , 1.193; Pu ²⁴² , 0.102; Pu ²⁴⁴ , <0.005) ^a	15
Oxygen-to-metal ratio	1.97-1.99
Density, % theoretical	90-92
Stack smear density of (U,Pu)O ₂ pellets, % theoretical	83-85
BET surface area, m ² /g	<0.05
Fuel weight at 84% TD (TD = 10.99 g/cm ³), g . .	115

Blanket pellets

Pellet material	UO ₂
Enrichment, % U ²³⁵	0.71 (natural)
Number of pellets	12 at top, 12 at bottom (for one rod)
OD, in.	0.2595
Length, in.	0.25
Stack height, in.	3.00 (for one rod)
Oxygen-to-uranium ratio	2.005
Density, % theoretical	88-91
Blanket weight at 90% TD (TD = 10.96 g/cm ³) 88% smear density, g	25.65 at each end

Active fission-product trap (one rod)
(in G-4,-6,-7)

Material	Activated coconut char- coal
Bed length, in.	10 (upper), 11 (lower)

Table 3 (continued)

Charcoal weight, g	3.902 (upper), 4.280 (lower)
Charcoal type	Barnebey-Cheney MI 6736
Particle size	10-20 mesh
Density	0.448 g/cm ³
BET surface area	1004 m ² /g
Charcoal impurity content, ppm	
Ba	4
B	<1
Fe	40
Ca	120
Sr	<80
Mn	<0.20
Al	20
Cu	4
Si	400
Mg	20
Ti	2
Na	<120
P	<200
Screen, 316 SS, 40 mesh, g	0.0405 ea x 7 = 0.2835
Support tube, 316 SS, g	0.144 ea x 7 = 1.008
Sealed charcoal traps (7 rods)	
Stainless clad tube	2.03 g x 14 = 28.4 g
Activated charcoal	0.30 g x 11 = 3.3 g 0.48 g x 3 = 1.45 g
Other data	
Gas in fuel rod	Helium
Gas volume in fuel rod, cm ³	
G-4, -6, -7	16.2
G-1, -2, -3, -5 (-8)	19.8
Preirradiation gas pressure in rod	4 psig Helium at 25°C
Temperature monitors (7 rods)	
SiC, g	0.082 x 0.400 long (in.) 0.083 ea x 24 x 7 = 13.94
Molybdenum tubes, g	0.4697 ea x 24 x 7 = 78.9

Table 3 (continued)

316 stainless plugs								
	<u>G-1</u>	<u>G-2</u>	<u>G-3</u>	<u>G-4</u>	<u>G-5</u>	<u>G-6</u>	<u>G-7</u>	<u>G-8</u>
Grams	0.432	0.4755	0.4914	0.4755	0.4853	0.5336	0.4724	0.4914
Grams x 24	10.36	11.4	11.79	11.4	11.65	12.77	11.34	11.79
316 stainless tubes, g					0.67 x 24 x 8 = 112.6			
Dosimeters (one rod)								
Outer tube, g					10.325 (1)			
3-15/16 in. spacer tubes, g					0.4234 ea x 3 = 1.2702			
1 in. spacer tubes, g					0.1065 ea x 2 = 0.2130			
Tungsten, g					1.563 (1)			
Inner tube, g, each containing the following .					0.3635 ea x 10 = 3.635			
ThC ₂ particles, g					<1 total			
Depleted UC ₂ particles, g					<1 total			
(Th, enr. U)C ₂ particles, g					<1 total			
Enriched UC ₂ particles, g					<1 total			
V-Fe, g					0.01557 each			
Al-Co, g					0.0136 each			
Sodium volume, cm ³								
	<u>G-1</u>	<u>G-2</u>	<u>G-3</u>	<u>G-4</u>	<u>G-5</u>	<u>G-6</u>	<u>G-7</u>	<u>G-8</u>
	73	91	128	106	98	147	178	128
Sodium level, in. (distance below reference level; see Fig. 3)								
	11.90	11.68	11.76	12.46	12.46	11.90	11.57	11.72
Volume of gas plenum (calculated), cm ³								
	<u>G-1</u>	<u>G-2</u>	<u>G-3</u>	<u>G-4</u>	<u>G-5</u>	<u>G-6</u>	<u>G-7</u>	<u>G-8</u>
Rod	19.8	19.8	19.8	16.2	19.8	16.2	16.2	19.8
Capsule	15.4	18.9	25.2	22.2	21.4	28.2	34.7	25.2
Gamma and neutron activity at 1 ft								
No detectable neutron activity.								
Maximum gamma activity less than 1 mrem/hr.								

^aDate of last chemical separation (7 lots of Pu(NO₃)₄):

3 lots	Feb. 1967
4 lots	May 1968

dust was seen falling out of the hole through the clad standoff spacer.

An experiment was performed on a portion of the batch of charcoal retained at GGA to determine the fraction of charcoal that might be generated due to vibration from shipping, handling, or reactor operation and would thus pass through the screen. A 10-g sample was placed in a large 35-mesh sieve (Tyler standard size, equivalent to the 40-mesh screen in the trap) and vigorously agitated for 30 min on a sieve shaker. The result was a loss of 0.06% charcoal. The charcoal is expected to shrink ~30% due to the neutron environment and temperature in the reactor. This condition was simulated by agitating a 10-g sample of the same lot of charcoal in a 28-mesh sieve (30% larger than the trap screen) for 30 min on a sieve shaker, resulting again in a loss of 0.06% charcoal.

The active traps contain an average of 8 g of charcoal per fuel rod. If the exposure of the fuel rods to vibration is as severe as in the laboratory experiment, a total loss of about 4.8 mg of charcoal would result. Since the fuel rods will be in an upright position most of the time, the upper trap (~4 g) would be the major source charcoal fines, and not much more than 3 mg is assumed to pass through the screens. Further, the traps are small in diameter and long—a much different geometry from the large sieve used in the experiment. The long column of charcoal in the trap with its several screens would retain some of the particulate charcoal. Thus, it appears likely that less than 1 mg of charcoal would sift down to the ends of the blanket pellets. Any charcoal passing beyond that point would have to fall through the (0.003 in. max) diametral gap between the solid blanket pellets and the cladding.

The 0.06% charcoal that passed through the 35-mesh sieve was sieved with a 200-mesh sieve having 0.0029-in. openings, with the result that 17% passed through. The charcoal retained on the 200-mesh sieve was sieved with a 60-mesh sieve having 0.0098-in. openings, with the result that 20% passed through. Thus, of the 0.06% charcoal passing through the screen size used in the active traps, 63% is larger than 0.010-in. and only 17% is smaller than the 0.003-in. maximum diametral gap between the blanket pellets and the cladding ID. Therefore, the diametral blanket-to-pellet gap would hold back about 63% of the charcoal. The charcoal that could work through the

3-in. long blanket-to-clad gap would be found on the periphery of the fuel pellets at a maximum temperature of 1015°C.

G. POSTIRRADIATION OUT-OF-REACTOR SERVICES

1. Capsule Removal and Disassembly

During cooling, sodium removal, storage, and disassembly of the sub-assembly or capsules, the silicon carbide (SiC) temperature monitors should not experience temperatures above 700°F. Therefore, the temperature of the capsule wall of capsules G-1 through G-8 should be lowered to below 700°F as soon as possible. Immediately after subassembly disassembly, a precision gamma scan should be made in the VAD machine of capsules G-4, G-6, and G-7 to identify isotope distribution in the charcoal traps before vibration or rotation to an out-of-vertical position causes mixing of the charcoal. The experiment is not expected to contain failed capsules.

2. Interim Examinations

Neutron radiography shall be performed on the fuel-rod capsules prior to startup of irradiation of the F-1 experiment, at EBR-II shutdowns corresponding to burnup exposures of 25,000, 50,000, 62,500, 75,000, 87,500, and 100,000 MWd/Te, and following the last EBR-II cycle of the fuel-rod irradiation. Neutron radiography shall be performed to best show the condition of the fuel-rod specimens within their capsule containments. Gamma scanning shall be performed and dimensional measurements shall be taken with a Bausch and Lomb DR-25B Optical Gage (± 0.0001 in.) of each capsule's external diametral dimension after each neutron radiographic examination. The weight (± 0.1 gram) of each capsule shall be obtained and recorded. Leak testing of the capsules and sodium-bond-level measurements shall also be performed. Photographs shall be taken to record the visual appearance of the capsules during each interim examination. Center-of-balance determinations should be made before irradiation and at each interim examination.

Conditions that must be met before reinsertion of the capsules into the reactor are:

1. No evidence of ruptures in either the fuel-element cladding or the capsule wall.

2. No detectable leakage of fission products.
3. Plastic strain of the fuel-element cladding not to exceed 1.6%, except for rod G-4 (see Section 2E).

In order to satisfy the third condition, it would be desirable to have a precision measurement of ± 0.0005 in. of the fuel-element diameter by neutron radiography.

H. SHIPPING PROCEDURE

If a decision is made at a later date that the postirradiation examination will not be conducted at EBR-II, the capsule(s) will be shipped intact from EBR-II to Oak Ridge National Laboratory (ORNL). In any event, should facilities not be available at EBR-II for certain desired postirradiation examinations, it is anticipated the examinations will be performed at ORNL.

The temperature requirements of Section G should be met. The capsule(s) shall be cleaned of sodium and contaminants and packaged in a containment furnished by ORNL. This will prevent the capsule or capsules from movement within the cask cavity, which is 5-3/4 in. ID by 69 in. long. The cask is 24 in. OD and 90 in. long, weighs about 16,000 lb, and is a lead-filled steel weldment. This ORNL cask has a DOT special permit number 5753, with an expiration date of June 30, 1971.

In the event that postirradiation examination is performed at EBR-II, the temperature monitors, dosimeters, and traps will be shipped to GGA for analysis. The GGA shipping cask for these parts is a cylindrical steel-covered, lead-shielded container 36 in. high and 16 in. in diameter and weighs 2,000 lb. The cavity is 1-1/2 in. in diameter and 12 in. deep. The cask is mounted on a pallet; the approval number is IC-55.

I. THERMAL PERFORMANCE OF CAPSULE

The fuel pellet and cladding arrangement is similar to that used in previous experimenters' assemblies, so no new methods were required for the thermal analysis of these parts. It was assumed that an effective contact resistance of 0.0015 in. of helium was maintained between the fuel pellet and the cladding. This is in spite of fuel restructuring, fuel swelling, and fission-gas production, which, in theory, will lower the gap

conductance. Testing has indicated, however, that there is no significant change in the gap conductance during the irradiation cycle,⁽²⁾ so it was assumed that the initial conductance will be maintained throughout the test cycle.

The cladding is of 20% cold-worked (CW) type 316 stainless steel. The stress analysis results noted in Section K uses correlations developed from various tests with 300 series steels and do not necessarily specifically apply to 20% CW type 316 stainless steel. However, it is anticipated that the correlations used will yield conservative results when applied to the 20% CW type 316 stainless steel, so the reported results should be safe for the condition of the test. There is a dearth of information on the properties of 20% CW type 316 stainless steel under the conditions that these capsules will experience. Because of this, no strict conclusions will be drawn regarding the results of the creep calculations noted later in Section K.

1. Fuel Pellets

It is assumed that the fuel pellets will restructure immediately upon attaining the necessary temperature, as indicated by Mikailoff, et al.⁽³⁾ This restructuring should occur during the specified startup. No transient analysis was made of the restructuring part of the cycle. After restructuring, the following correlations were used for the thermal conductivity of the fuel pellet material:

$$K = 0.011 + \frac{1}{T(0.4848 - 0.4465 D)}$$

where K = conductivity, watts/cm-°C,

T = °C,

D = fractional theoretical density.

It was assumed that the degradation of the fuel material that will occur during the test does not adversely affect the thermal properties of the pellet.

The calculated power generation was based on the assumption that the center rod of the subassembly would operate at an axial peak power of 14.9 kW/ft. The axial power profile of the fuel pellets results in the axial temperature profiles shown in Fig. 7. This plot is for the highest heat flux case, which was assumed to have an axial peak power of 16.1 kW/ft.

The temperatures noted were calculated with the heat-transfer code TAC2D and with BRITL,⁽⁴⁾ which are summarized in Appendix C.

2. Cladding Thermal Analysis

A significant thermal gradient exists across the fuel cladding because of the relatively high heat flux. The thermal gradients for all of the cases are plotted in Fig. 8. The average rod has a thermal gradient of 40°C (72°F) maximum in the cladding and is proportional to the relative power at any other location.

A temperature-dependent function for the thermal conductivity was used for calculating the cladding temperatures:

$$K = 8.625 + 4.375 \times 10^{-3} (T \text{ } ^\circ\text{F}) \quad (\text{Btu}/(\text{hr})(\text{ft})(^\circ\text{F})).$$

This function and the material properties were used in the TAC2D code for obtaining the design cladding temperatures. The thermal performance of the fuel rods was calculated by BRITL using the cladding outer surface temperatures resulting from TAC2D calculations. The results of the thermal performance of the fuel rods and capsules are given in Table 4. The uncertainty factors for these two codes are listed in Table 5.

Extensive experience in the thermal design and analysis of the irradiation capsules with the use of TAC2D, and its predecessor, RAT, has shown that with the exclusion of reactor power uncertainty, the net inaccuracy is significantly less than ~11, and a value of ±5% is a reasonable estimate for this type of configuration.

The arrangement of the capsules in the subassembly are illustrated in Fig. 5.

Fig. 7 Axial temperature profile, 800°C nominal, for 16.1 kW fuel rod

Fig. 8 Thermal gradients in the fuel-rod cladding

Table 4

THERMAL PERFORMANCE OF FUEL RODS AND CAPSULES

Capsule Number	Cladding OD Peak Nominal Temperature (°C)	Heat Generation		Peak Surface Temperatures (°C)						Peak Local Coolant Temperature (°C)
		Peak Design (kW/ft)	Total (kW)	Fuel		Clad		Capsule		
				Inner	Outer	Mid-wall	Outer	Inner	Outer	
G-1	800	16.1	16.7	3101	1125	802	783	504	477	476
G-2	750	15.5	16.1	2952	1083	772	753	498	472	471
G-3	700	15.5	16.1	2877	1033	722	702	498	471	471
G-4	700	14.9	15.4	2783	1026	727	708	494	469	467
G-5	650	14.3	14.8	2622	975	691	669	488	463	462
G-6	700	14.4	14.9	2699	1026	740	721	491	466	465
G-7	600	13.8	14.3	2452	909	632	612	464	442	459
G-8	700	15.5	16.1	2877	1033	722	702	486	459	471

Total heat generation for
seven capsules

108.3

Table 5

UNCERTAINTY FACTORS

Material thermal properties	±5%
Thermal model (geometric factors)	±3%
Program calculation	±2%
Fuel loading	±1%
Reactor power	<u>±10%</u>
TOTAL	~±21

The design cladding temperatures are the maximum temperatures on the outer surface of the cladding. The thermal gradient in the cladding causes the maximum inner surface temperature to be 40°C (72°F) higher than the outer surface or design temperature. This radial gradient causes a thermal stress, which is discussed in Section K.

Cladding swelling has no significant thermal effect because a minor reduction in the sodium gap will have only a negligible effect on the capsule temperatures, and the small decrease in thermal conductivity will not cause a significant increase in thermal gradient.

There is only a negligible circumferential variation in the cladding temperatures. The variations across the diameter due to the power gradient is on the order of 4°C. Circumferential thermal variations on the capsule wall are not evident on the cladding wall because of the smoothing effect of the sodium gaps and the stainless-steel thermal barrier.

J. THERMAL PERFORMANCE OF THE SUBASSEMBLY

1. Mixed Mean Coolant Temperature

The division of flow between the three basic coolant channels was calculated with a small computer program, WRIT, which was written specifically

for this purpose. The coolant channels were defined as noted on Fig. 9. The significant parameters are tabulated in Table 6. Each capsule is assumed to have equal coolant flow; however, the distribution in the channels around a capsule are unequal (see Table 6). The basic purpose of the program (WRIT) is to determine the pressure drop and the individual coolant channel exit temperatures for a given total flow and fuel rod power. The channel exit temperatures are calculated with the simplifying assumption that the adjacent fuel rods develop exactly the same power, which is, of course, not correct. Since the basic purpose is, however, to equalize the channel coolant temperatures, the final result will be reasonably satisfactory.

Table 6
FLOW CHANNEL PARAMETERS FOR ALL CAPSULES

	Area	Perimeter	Hydraulic Radius, D_H	Reynolds No.	Flow (lb/hr)	Exit Tem- perature ^a (°F)
Channel 1	0.0086 in. ²	0.44 in.	0.00648 in.	14188.	85.3	868
Channel 2	0.0239 in. ²	1.01 in.	0.00789 in.	19491.	270.6	859
Channel 3	0.0127 in. ²	0.408 in.	0.01039 in.	30721.	171.1	867

Note: Channel Δp is 6.9 psi with an assembly flow of 17 gpm, including entrance and exit losses but not any orificing.

^aCalculated for capsule G-4 (average case).

The computer program uses the following functions for determining the pertinent coolant parameters:

$$\text{Viscosity } (\mu) = (0.8499 \times 10^{-6} \times T - 0.002) \times T + 1.753 \text{ lb/hr-ft}$$

$$\text{Density } (\rho) = 59.8 - 8.6 \times 10^{-3} (T - 32) \text{ lb/ft}^3$$

$$\text{Friction factor } (f) = 0.046/\text{Re}^{0.2}$$

where T = temperature, °F

Re = Reynolds number

Fig. 9 Model section

With no mixing, the sodium in the coolest channel of the subassembly exits at 858°F and in the hottest channel, at 888°F. Thus, they are within the Row 7 range of 950°F ±100°F. The spiral wire wrap around each capsule will force a certain amount of mixing to occur. Thus, the noted maximum will probably not occur in the actual model.

The design coolant flow rate of 7280 lb/hr is easily attained with the design. A channel pressure drop of 6.9 psi, including entrance and exit losses, was calculated for the nominal, or 14.9 kW/ft, rod. It is assumed that the remainder of the available core pressure drop will be used in the orificing.

K. STRESS AND STRAIN

The stresses and strains for the fuel cladding and capsule cladding are given in Table 7. The following subsections give examples of the calculations leading to the tabulated values.

1. Fuel Cladding

The principal stress in the cladding is the thermal stress that results from the radial temperature gradient in the cladding wall. This radial gradient causes a temperature gradient of 37° to 43°C in the various rods. A second stress is produced by the differential expansion of capsule components and the helium fill gas in the plenum and the gradual buildup of fission gases.

(a) Thermal Stress and Strain

The thermal stress is computed as follows and is listed in Table 7:

$$\sigma_{th} = \frac{E\alpha\Delta T}{2(1 - \nu)}$$

where $E = 29.5 \times 10^6 (1 - 2.37 \times 10^{-4} T)$

$\nu = 0.3$

ΔT = temperature gradient

$\alpha = 10.5 \times 10^{-6}$

T = peak axial temperature, °F

Table 7
ROD AND CAPSULE PRESSURE, STRESS, STRAIN

G	Rod								Capsule			
	Internal Pressure BOL, ^a psia	Internal Pressure EOL, ^b psia	Pressure Stress ^c EOL, psi	Pressure Stress ^d EOL, psi	Thermal Stress, psi	Creep Strain EOL, %	Swelling EOL, %	Total Diameter Increase EOL, %	Internal Pressure Hot, psia	Pressure Stress, psi	Thermal Stress, psi	Pressure With Failed Rod EOL, psi
1	28	156	874	8,600	10,600	0.70	0.0	0.72	70	617	12,600	1,622
2	28	147	782	11,050	10,700	0.63	0.0	0.66	70	617	12,200	869
3	28	90	167	1,930	11,420	0.001	0.04	0.045	71	630	12,200	738
4	28	305	2,070	15,100	10,900	3.20	0.35	3.59	70	617	11,700	2,143
5	28	123	493	16,250	11,000	0.12	0.23	0.38	67	577	11,400	1,100
6	28	153	722	12,420	10,350	0.19	0.43	0.48	71	630	11,300	1,072
7	28	126	502	20,020	11,420	0.11	0.28	0.41	69	603	10,650	1,069
8	28	203	1,161	15,130	11,420	1.59	0.27	1.92	71	630	12,200	1,405

^aBeginning of life.

^bEnd of life.

^cDue to internal gas pressure.

^dDue to internal gas pressure plus fuel swelling.

The stresses are tensile on the outer surface and compressive on the inner surface and are balanced.

(b) Pressure Stress and Strain

The capsule and rod are backfilled with helium during the loading process and are sealed at a pressure of 4 psig with the components at room temperature. When the assembly is brought to operating temperature, expansion of the fuel causes a reduction of the free volume of the plenum gas. Gaseous fission products produced during irradiation will add substantially to the internal rod pressure. The internal pressure and resultant radial strains were computed with BRITL and are tabulated in Table 7.

The hoop stress in the cladding (cladding pressure stress) is determined from

$$\sigma = \frac{P D_i}{2t} ,$$

where σ = stress,

P = gas pressure, or gas pressure plus fuel swelling pressure,

D_i = cladding ID,

t = effective cladding thickness, in.

For G-1 and G-2,

$$t = 0.013 ,$$

which was reduced 0.005 in. from the original dimension to allow for fuel cladding reactions, manufacturing tolerances, and defects less than the detectable standard size. For G-3 through G-8,

$$t = 0.015 ,$$

which was reduced 0.003 in. to allow for fuel cladding reactions, manufacturing tolerances, and defects less than the detectable standard size. It is assumed that static sodium corrosion on the clad OD is negligible.

On the outer surface of the cladding, the pressure and thermal stresses are both tensile. On the inner surface, the pressure stress is tensile and the thermal stress is compressive. Thus, the highest stresses are on the outer surface.

2. Capsule

(a) Thermal Stress

The thermal gradients in the capsules are somewhat less than those in the cladding because the capsules have a larger heat-transfer area. The highest-temperature rod has a thermal gradient of 37°C (67°F) across the capsule wall compared to 38°C (69°F) across the cladding thickness. The maximum thermal stress in the capsule is

$$\sigma_{th} = \frac{E\alpha\Delta T}{2(1 - \nu)} = 12,070 \text{ psi}$$

where $E = 29.5 \times 10^6 (1 - 2.37 \times 10^{-4} T) = 24 \times 10^6 \text{ psi}$

$$\alpha = 10.5 \times 10^{-6} \text{ } ^\circ\text{F}$$

$$\Delta T = 67^\circ\text{F}$$

$$\nu \approx 0.3$$

$$T \approx 790^\circ\text{F}$$

The thermal stress is higher than in the cladding due to the much higher value of E obtained at the lower operating temperature. The resulting strain is

$$\epsilon_{th} = \frac{\alpha}{E} = \frac{12,070}{24 \times 10^6} = 0.00050 \text{ .}$$

(b) Pressure Stress

The pressure stress under normal conditions is nominal. The capsule is sealed at room temperature with 4 psig of helium in the capsule plenum. The cold and hot capsule plenum volumes are determined by calculation of the internal component volumes and the differential expansion of the components. The hot plenum pressure is determined from the decrease in plenum volume and rise in temperature. The hot plenum pressures are listed in Table 5. The external pressure on the capsules is ~24 psia, so the net pressure on each

capsule is reduced by that amount. The resulting stress for a capsule wall of nominal thickness is computed by

$$\sigma_p = \frac{PD}{2t}$$

where P = pressure differential

D = internal diameter

t = wall thickness

and is listed in Table 7. The thickness was reduced by 0.0028 for fabrication tolerance, minimum detectable defect size, and corrosion allowance.

(c) Capsule Pressure with a Failed Rod

If the fuel rod fails, the fission gases can equalize between the rod and capsule plenums. The final capsule pressure is then

$$P_r = \frac{P_r V_r + P_c V_c}{V_r + V_c}$$

where P_r = rod pressure

P_c = capsule pressure

V_r = rod plenum volume

V_c = capsule plenum volume

At the hottest point in the capsule the external pressure is ~24 psia, so the pressure is reduced by that amount. The pressure stress is calculated as in (b) and is listed in Table 5. The yield stress is ~18,000 psi at 888°F.⁽⁵⁾ The pressure stress with a rod failure is then well under the yield stress for the material. The capsule would have to erode to about 0.003 in. before the pressure stress equaled the yield stress in this case.

L. EFFECT OF INCREASED DIAMETER OF CAPSULES ON COOLANT FLOW

A slight increase in the diameter of the capsule is anticipated due to the internal pressure of the capsule and the swelling of the stainless steel. A correlation for the swelling is provided⁽⁶⁾ as follows:

$$\frac{\% \Delta V}{V_o} = 10^{-48.31} (\phi t)^{1.71} 10 \left(\frac{1.549 \times 10^4}{T} - \frac{5.988 \times 10^6}{T^2} \right)$$

where V_o = initial volume,

ϕt = fluence = 6×10^{22} ,

T = temperature = 427°C

$$\frac{\Delta V}{V_o} = 3.54\%$$

Then

$$\frac{1}{3} \frac{\Delta V}{V_o} = \frac{\Delta D}{D_o} \approx 1.2\%$$

The increase in the thickness of the capsule wall will be negligible and the increase in the diameter will be $0.012 \times 0.779 = 0.009$ in., so the new diameter will then be $0.779 + 0.009 \approx 0.788$ in. The channel size is thus reduced by this increase in diameter.

The internal pressure in the capsule will be small. Under normal conditions, the (hot) free volume is 15 to 37 cm^3 . The capsule is sealed with 4 psig of helium over the sodium. The final pressure, when at the operating temperature, will be ~67 to 71 psia.

This pressure is insignificant as far as any dimensional effect on the capsule is concerned.

The channel area is a function of $1/OD^2$ (of the capsule), if it can be assumed that the capsules are free to expand within the hexagonal cluster. The wire wrap will tend to prevent free expansion but, at worst, if the expansion can freely take place, the area is reduced by the ratio of the square of the design and expanded capsule diameters; thus

$$\text{Percent area reduction} = \frac{0.779^2}{0.788^2} = \frac{0.6068}{0.6209} = 0.977$$

A channel area decrease of ~2% will not have a noticeable effect on the capsule performance and is within the normal approximation of the analysis.

2. SAFETY ANALYSIS

The potential hazards for capsules G-1 through G-8 have been reviewed in accordance with the EBR-II experimenters guide⁽⁶⁾ and are discussed in the following sections. No personnel hazards are anticipated other than those normally associated with doubly contained (U,Pu)O₂ and singly contained sodium.

A. NORMAL OPERATION

Under conditions of normal reactor operation at 62.5 MW plus 10%, the inner radius of the fuel will be operating at or above the fuel melting point. About 12% of the inner radius of the fuel may be molten; nevertheless, no gross fuel movement is expected at maximum burnup. Fuel swelling will result in 3.8% axial growth. The fuel pellets will restructure during the reactor startup, as specified in Section 1.B.4, and all calculations are based on properties of the restructured fuel.

Fuel-cladding reactions of up to 0.005 in. of the wall thickness have been observed in the cladding inner surface in thermal reactor irradiations at 735°C; however, such reactions in a fast reactor have not yet been determined. One of the objectives of the F-1 experiment is to gain more information about that behavior. The thermal barrier and limited interconnection between the inner and outer sodium annuli serve to contain fuel released in the event of a rod failure and to prevent attack of capsule cladding. Furthermore, the capsule cladding will be at much lower temperatures where no reactions have been observed. Sodium-cladding reactions in clean static sodium are assumed to have a negligible effect on cladding integrity because long-term rupture and creep properties are essentially the same in helium or in clean static sodium.

The effect of swelling on the capsule cladding will be only a 1% decrease in flow channel area, which is not significant. Fuel-rod radial strain estimated with the BRITL code has been found to be less than 2.0% except for the 700°C, 10.7% burnup rod, which will undergo approximately 3.6% total radial expansion. This degree of expansion will not cause any interaction with other capsule components or significantly affect radial temperature distribution. The effect of the environment on the fuel mechanical behavior is not well known for mixed-oxide fuel; however, some in-pile information on UO_2 indicates an enhancement of the creep rate below about 1200°C. The effect of neutron irradiation on the capsule cladding at its normal operating temperature will be an increase in yield point to near the ultimate tensile strength and a decrease in ductility down to 1 to 5%. The fuel cladding will experience a range of property changes because of the greater range of temperatures. The higher-temperature rods are expected to undergo a smaller increase in yield and a smaller decrease in ductility.

B. ABNORMAL REACTOR OPERATION

1. Overpower Condition

a. Nuclear Excursion in a Core at Zero Power

This is case 1 of Appendix E of "Guide for Irradiation Experiments in EBR-II," Rev. 3, July 1969, and is defined as the accidental insertion of a control rod in a delayed-critical core with full coolant flow. The power curve in Appendix E provides for a total time of 250 sec, which is adequate for the assembly to attain a steady-state condition. The results of the transient are summarized in Fig. 10.

During the transient, the thermal gradients in the fuel and cladding will remain below the steady-state values. Thus, no stresses above those obtained during normal full power operation are expected to be encountered.

b. Accidental Insertion of a Control Rod in a Core at Normal Power

This is case 2 of Appendix E in the EBR-II experimenters guide. At 50 sec, which is at the end of the defined transient, the cladding outer surface will have increased 125°C (225°F) to ~900°C and the peak fuel temperature increased 950°C (1720°F) to beyond the melting point. The fuel-rod pressures

Fig. 10 Capsule temperatures following an accidental insertion of a control rod in delayed critical core

and strains were not calculated, but it is evident that failure would be imminent. The capsule wall shows only a very modest increase of 35°C (60°F), which will not significantly reduce its strength. Thus, the capsule should not fail during this accident condition if the power excursion is terminated with 50 sec. The calculated temperatures are summarized in Fig. 11. Continued irradiation after this transient would be subject to a review of each capsule condition.

2. Loss of Coolant Flow

A loss of coolant pumping power and subsequent reactor scram causes a sharp drop in the capsule temperatures. The calculated results are shown on Fig. 12, where the capsule temperatures at the axial point of maximum temperature are given as a function of time. The thermal gradients in the capsule, cladding, and fuel are all less than during normal steady-state operation.

C. CLADDING OR CAPSULE FAILURE

1. Fuel Cladding Failure with Intact Capsule

The effect of a rupture of the fuel cladding would cause an increase in capsule pressure to 172 psia in the worst case (G-4). The net capsule hoop stress, after allowing for external pressure due to primary sodium, would be 2143 psi, which is considerably below the yield stress of 18,000 psi. The capsule pressure stress with fuel cladding failure is well below the yield stress, and the effect of creep strain will be negligible for the times involved, as can be seen by comparison with similar calculations for the fuel cladding. This results in negligible capsule diameter increase or flow channel decrease.

Failure of the fuel cladding that will allow ingress of sodium will increase the rate of fuel-cladding reactions and this would possibly lead to an extension of the area of failure; however, the capsule cladding operates at a lower temperature where rates of attack are negligible.

2. Combined Fuel Cladding and Capsule Failure

a. Gas Blanketing

A release of gas from the capsule would be possible if combined failure

Fig. 11 Capsule temperatures following an accidental insertion of a control rod in a normal core

Fig. 12 Capsule temperatures following a loss of coolant flow followed by scram

occurred. The released gas should not affect the adjacent assemblies for two reasons: (1) the upflow coolant stream will sweep any gas bubbles in the direction that they will tend to go anyway and thus any bubbles should not remain in the core region, and (2) the effect of the stainless-steel thermal barrier and the two sodium annuli is such that a small region of zero heat transfer on the capsule surface does not cause a significant hot spot. Previous calculations have shown that a short region ($\sim 7^\circ$ circumferential) of zero heat transfer affects the capsule by less than 5°C (10°F).

b. Primary Coolant Contamination

The maximum radioactivity concentrations(s) (in $\mu\text{C}/\text{cm}^3$) in the cover gas and in the Na coolant of EBR-II that could result from failure of one fuel rod and a single capsule containment was calculated and is shown in Fig. 13 as a function of time after irradiation. The results are based on 16.7 kW(t) total power generated in the highest-powered rod during continuous irradiation to 570 full power days. The cover-gas radioactivity is the sum of the activities of all of the Kr and Xe isotopes diluted in the cover gas volume (625 ft^3), assuming 100% release from the fuel to the cover gas. The solid-fission-product radioactivity concentration is the sum of the activities of all the fission products in the 269 isotope library except those of Kr and Xe homogeneously dispersed in the coolant volume ($11,500\text{ ft}^3$) of EBR-II, assuming 100% release from the fuel of capsule G-1 to the coolant.

c. Exposure of Fuel to Primary Coolant

The fuel would not be directly exposed to primary coolant because the thermal barrier would limit access of primary sodium to the failed fuel rod.

d. Capsule Sodium Level

After failure of the cladding capsule wall, the sodium level will be lowered. The worst case is capsule G-1 with the smallest internal void space. The internal pressure of G-1 is ~ 70 psia and the coolant pressure is ~ 24 psi with the result that the sodium level would be reduced ~ 20 in. to about the level of the top of the fuel column which may result in fuel-clad melting. This assumes that failure occurs below this point. If failure occurs near the top of the capsule at the maximum temperature region, the fill gas would be vented and the sodium level would not be lowered.

Fig. 13 Cover gas and coolant contamination following rupture of highest rated rod under EOL conditions

In the event of combined rod and capsule failure below the top of the fuel column at the end of the scheduled irradiation period, the conditions in rod G-1 with the smallest internal void volume and rod G-4 with the greatest pressure due to fission gas buildup would result in the sodium being expelled from the capsule down to the level of rupture in the capsule. These conditions are expected to result in fuel rod melting. This situation is expected to result in molten cladding, fuel, and thermal barrier being quenched when contacting the capsule wall or dropping to the lower part of the capsule, as was experienced with the ANL experiment HOV-4.⁽⁷⁾

e. Failed Capsule Sodium Removal

In the event of capsule and cladding failure, the capsule should be able to go through the normal sodium removal process. A continual reaction of sodium with moist argon or air might be expected at the point of capsule failure. Loss of fuel from the capsule is considered unlikely.

f. Unencapsulated Fuel Element Cladding Failure

Not applicable.

g. Nonfuel Materials Capsule Failure

Not applicable.

D. CRITICALITY

If the subassembly of 7 capsules were immersed in water in isolation from other fissionable material, it would achieve a safe, noncritical level of reactivity, $k_{\text{eff}} = 0.25$. This result is based on a two-dimensional x-y calculation using transport theory and a detailed energy representation.

E. CRITERIA FOR REINSERTION

It is planned to perform an interim examination at about 25,000, 50,000, 62,500, 75,000, and 87,500 MWd/Te on those capsules scheduled for greater exposure. This examination shall consist of disassembly of the subassembly and visual examination, dimensional measurements, gamma scan, neutron radiograph, and leak check on each capsule.

A 1.5% $\Delta D/D$ fuel rod increase shall be cause to terminate exposure of a capsule, with the exception of G-4, which is expected to reach $\sim 3.6\%$ $\Delta D/D$ at 100,000 MWd/Te and will be continued to that amount of diametral increase. Any evidence of fission-gas leakage shall also be cause for termination.

3. QUALITY ASSURANCE

The quality assurance plan for capsules G-1 through G-8 in the F-1 experiment includes the fuel-rod components, capsule components, thermal barrier components, dosimeters, and temperature monitors. Chemical analysis has been performed on a representative sample of each component to certify its composition. Metallography has been performed to determine the microstructural characteristics of each component. There are several additional quality assurance procedures that are unique to each component; these are discussed below.

A. FUEL RODS

1. Quality Assurance at GGA

The outside diameter (OD), the inside diameter (ID), and the wall thickness of the cladding tubing for the fuel rods were dimensionally inspected by air gauging. Any tube with a wall thickness measurement greater than a $\pm 10\%$ tolerance was rejected. Ultrasonic inspection of all cladding tubing was performed. Any tubing containing flaws in excess of 0.0015 in. deep by 0.030 in. long was rejected. Dye penetrant inspection for surface-connected cracks or flaws was made.

In addition to the general quality assurance indicated for all the components, the natural UO_2 blanket pellets were characterized for density and oxygen-to-uranium ratio.

2. Quality Assurance at ORNL

- a. All closure welds were dye-penetrant and X-ray radiographic inspected. Also, all welds passed a mass spectrometer helium leak test. The limit of leak detector sensitivity is $2 \times 10^{-10} \text{ cm}^3/\text{sec}$.

- b. The fuel column in the rod was inspected with a gamma-ray densitometer; density variations were recorded as a function of axial location.
- c. The completed specimens were X-ray radiographed to confirm proper fuel and component positions and dimensions.
- d. The end fittings were subjected to dye penetrant and dimensional inspection.

B. CAPSULES

1. Quality Assurance at GGA

The OD of the capsule tubing was dimensionally inspected using standard micrometers and the ID was inspected using Interimiks; the wall thickness was determined from $(OD - ID)/2$. The capsule tubing was inspected by an ultrasonic technique, and any tube with a flaw indication greater than the standard notch of 0.0015 in. deep by 0.030 in. long was rejected.

a. Thermal Barriers. The OD of the thermal-barrier tubing was dimensionally inspected using standard micrometers and the ID was inspected using Interimiks; the total indicated wall thickness was determined from $(OD - ID)/2$.

The standoffs that hold the thermal barrier in the center of the capsule tube and that also contain the SiC temperature monitors (within a molybdenum envelope which is seal-welded in a helium atmosphere and leak checked) were helium leak checked after being seal-welded in a helium-filled glove box. A special welding sequence was employed in welding the standoffs to the thermal barrier so that distortion of the thermal barrier was minimized. Upon completion of the welding of the standoffs, the thermal barrier is checked for distortion and bow and for proper assembly fit by passing the thermal-barrier assembly through a full length section of the capsule tubing. The dosimeter-container tube is then tack-welded to the thermal barrier. The entire assembly is then X-ray radiographed to ensure that the welds are adequate and that the temperature monitors and dosimeter materials are in their proper positions.

b. Fuel Rod Fission-product Traps. The activated charcoal was characterized by determining particle size, density, BET surface area, and impurity content. The charcoal was outgassed and placed in sealed vials until it was loaded. Two types of fission-product traps are employed: active charcoal traps that communicate with the fission gases released from the fuel are used in three of the eight rods and helium-filled sealed charcoal traps are placed in all of the rods.

The active traps are separated into several sections by stainless steel screens. The amount of charcoal loaded into each separate section was weighed to ± 0.01 gram and the weights and identity of the rod and section recorded. Loading was performed in a helium-filled box (< 100 ppm moisture) to minimize adsorption of moisture in the charcoal. The traps were then welded and helium leak checked. The active traps were X-ray radiographed to determine the proper positioning of the charcoal and the stainless steel screen separators and the adequacy of the welds.

The sealed traps were also loaded in a helium atmosphere box with a preweighed and prerecorded amount of charcoal and then seal-welded. The welds were helium leak checked and X-ray radiographed.

2. ORNL Capsule Assembly

The ORNL quality assurance program for the capsules included:

- a. All capsule end fittings dye-penetrant inspected for surface-connected flaws.
- b. Weld qualification and welding procedures were established.
- c. All closure welds were dye-penetrant inspected and X-ray radiographed. All welds had to pass a mass spectrometer helium leak test at ambient pressure. The weld between the bottom end adapter and the capsule tubing also had to pass a helium leak test with an internal helium pressure of 1910 psi (pressure required to attain a capsule wall stress of 16,000 psi). The limit of leak detector sensitivity under this condition is $2 \times 10^{-10} \text{ cm}^3/\text{sec}$.

- d. All capsules were sodium filled in accordance with the procedures normally used by ORNL.
- e. The sodium bonding was eddy-current tested to detect voids greater than 1/16 in. in diameter.
- f. The completed capsules were X-ray radiographed to verify the location of the various components and to further inspect the sodium-bond integrity.
- g. The completed capsules were dimensionally inspected and weighed.

3. GGA Quality Assurance on Mark B-7B Subassembly Hex Container

When the flow strips were installed and heliarc welded to the Mark B7-B hex can, the assembly was inspected to ensure that the flow strips were installed tightly against the hex can wall. Dimensional measurements within a precision of ± 0.001 in. were made across the flats of the hex can. Bow measurements within a precision of ± 0.001 in. were also made to ensure that the maximum bow did not exceed 0.030 in. over the entire length of the sub-assembly. X-ray radiography was performed on practice flow strip heliarc welds to determine adequate penetration of at least a thickness of 0.5. Metallography and shear pull tests were performed on practice welds to assure adequate shear strength.

4. POSTIRRADIATION EXAMINATION

The postirradiation examination operations and procedures to be performed on the fuel-rod capsules irradiated in the F-1 experiment are outlined below. In addition to examining the fuel-rod capsules for fuel, cladding, fission-product trap, temperature monitor, and dosimeter performance, observations specifically pertaining to reactor safety and the experimental environment will be made.

A. REVIEW OF IRRADIATION HISTORY

Review of all preirradiation data and note any deviations from original specifications and original objectives of test.

Review irradiation history and note deviations from original plans (such as temperatures, heat-generation rates, times, and pressure and power cycles); especially note any unusual in-pile behavior, such as

- (1) High-activity release from primary containment that would indicate a leak; any indication of abnormal pressure buildup inside fuel-rod specimens; any indication of abnormal power distribution based on interim gamma scans.
- (2) Temperature instabilities (occurrence of power transients, abnormal temperatures in a particular region that would indicate instabilities in the sodium capsule bond.
- (3) Deformations that may have appeared in interim examination and in neutron radiographs.

Determine decay-heat production for the fuel based on irradiation history to establish that a sufficient cooling period has been allowed for safe capsule disassembly. Computational methods are described in Ref. 1.

B. CAPSULE EXAMINATION AND DISASSEMBLY

1. Nondestructive Testing (To be performed at EBR-II facility)

Note following information when removing capsule from subassembly:

- (a) Any unusual circumstances, such as need for forcing the specimens out of the capsule, that would indicate growth or wedging.
- (b) The orientation of capsule to the subassembly side nearest to the reactor face.
- (c) Any indication of possible capsule failure or deformation.

Perform neutron radiography to show condition of capsules and fuel elements within the capsules.

Perform gamma scanning and dimensional measurements of capsule external wall.

Record balance point of each capsule.

Record weight of each capsule.

Leak test capsule and measure sodium-bond level.

Photograph capsule assembly to record visual appearance.

Take sample of sodium bond for fission-product analysis.

2. Disassembly (To be performed at EBR-II or ORNL)

Remove SiC temperature monitors from locations within capsule, being careful to retain location identification and orientation of each monitor or monitor capsule. Note that most monitors are doubly contained within a stainless steel capsule and a molybdenum capsule. The stainless capsules must be removed. Irradiated SiC and Mo of high purity does not show long-term radioactivity and can be handled out of a hot cell. It is therefore anticipated that determination of the temperature information stored in the SiC will be made at GGA (see Section 4I, below).

Remove sealed capsules containing dosimeter materials, retaining location identification and orientation. Ship to GGA for analysis (see Section 4H, below). To facilitate shipping, the dosimeter outer capsule

can be crimped and cut at several of the tube spacers. The individual dosimeter subcapsules can be uniquely identified within any one of the G capsules; it will have the same identification in each G capsule.

Remove fuel rod from capsule.

(a) A fiducial mark must be applied to each fuel rod to indicate 0° angular orientation to correlate with the dosimeter or orientation mark "DOS" and the arrow die stamps on end hardware of capsule.

(b) Perform following nondestructive tests:

Record general observations of specimen appearance.

Photograph specimens to record appearance.

3. Determination of Cladding Integrity

Review in-pile data.

Check activity in sodium bond. Analyze sodium for presence of fission products.

Examine surfaces; if leak is present, determine location by

(a) Visual and optical examination of all suspect areas (peak temperature region, welds, wrinkles, or dents).

(b) Dye check.

(c) Observation of bubbles from specimen under liquid while evacuating atmosphere above liquid.

4. Visual Examination of Specimens in Detail

Integrity and unusual areas that show evidence of incipient failure.

Gross change in shape or ovality.

Bowing.

Wrinkling.

Ridging at pellet interfaces.

Ratcheting.

Surface condition (such as scratches, pits, and dents).

Heat marking.

Fuel swelling, particularly if swelling caused localized deformation of the cladding.

5. Gamma Scans of Rods and Active Traps

Use gross and qualitative scans above 0.55 MeV to determine overall distribution of fission products.

Use specific energy band for specific radioisotope, particularly non-volatile isotopes such as Zr^{95} , Nb^{95} , or Ce^{144} to obtain information on burnup.

Interpret gamma scans for axial power generation distribution, location of components, pellet cracking, and axial redistribution (swelling or sintering).

Use gamma Zr^{95}/Nb^{95} (nonvolatile species) scan as preliminary basis for selecting metallography and burnup samples.

Use cobalt standard along with gamma scan to obtain fuel-column length measurements.

6. Dimensional Measurements

Take dimensional measurements and compare with preirradiation measurements by tabulating and charting the

Diameter (OD) change (± 0.0001 in.)

Length change (± 0.001 in.)

Bowing change (± 0.001 in.)

Note preirradiation 0° orientation marks and define top or starting location. Correlate locations of postirradiation measurements to preirradiation measurements. Note if different equipment is used.

7. Cladding Surface Changes

Record changes in cladding surface condition as follows:

- (a) Surface will be examined under stereomicroscope. Observations of edge will be made on optical comparator. Any unusual areas will be photographed.

- (b) Areas with heat markings, discolorations, or change in cross section will be marked for further observation (by metallography) and measurements.
- (c) A profilometer will be used to detect ridges or dents in the 0.1- to 1-mil range.

8. Volume Measurements

Measure volume ($\pm 0.1\%$) of specimen and compare with preirradiation volume to determine overall cladding and fuel-swelling effects on cladding strain.

C. FUEL-ROD DISASSEMBLY AND EXAMINATION (To be performed at EBR-II facility or ORNL)

Remove the fission-product traps from the fuel rod and forward the traps to GGA for analysis. This operation should be done in an inert atmosphere for G-4, -6, -7, which contain active traps, to avoid contamination of the charcoal.

Section the fuel rod, employing the following precautions:

- (1) Note method (saw, cut, mill, shear, etc.) and assure that method will not destroy the usefulness of sample.
- (2) Note location and orientation of cuts on sketch with detailed dimensions showing exact location of transverse and longitudinal samples.
- (3) Observe each face of a cut. If conditions exist that might change with time (e.g., fission products present and reacting with atmosphere), take special precautions to store and prepare sample in nonreactive atmosphere.
- (4) Take microphotographs of each sample after sectioning for records.
- (5) If the cladding is slit and fuel pellets are removed, note the appearance of both and take photographs for the record.
- (6) Cut sections of fuel-element cladding suitable for determination of irradiation temperature by Kr⁸⁵ release technique (see 4I2), remove any fuel residue, and forward to GGA for analysis.

- (7) During disassembly of the fuel rods, it is desired that both the sealed and the active traps be separated from the remainder of the element and be returned to the GGA Hot Cell facility in sealed containers. This is not of critical importance for the sealed traps since it is only necessary that care be taken to ensure that the seal welds on these capsules are not broken during disassembly of the fuel rod.
- (8) It is expected that disassembly of the F-1 subassembly will take place with the rods in a horizontal position. The rotation of the fuel rods to this attitude and vibration associated with handling the assembly will most likely result in axial mixing of the charcoal contained in each of the sections of the active traps. However, to avoid excessive disruption of the axial fission-product profiles, it is desirable to impregnate the active traps with paraffin as soon as possible. This operation should be carried out on the active traps of rods G-4, G-6, and G-7 at the earliest time possible after disassembly of the rod begins. Impregnation with the paraffin can be accomplished through the inlet fitting of the trap. The initial operations (until impregnation with paraffin has been completed) on the fuel rods should be carried out in an inert or dry nitrogen atmosphere to prevent ingress of oxygen or moisture, which might lead to the release or movement of the materials sorbed on the charcoal. The traps should then be canned in stainless steel pipe, capped, and shipped to the GGA Hot Cell facility.
- (9) The sealed traps are separated from each fuel rod and their position in the rod is noted on the package (either upper or lower sealed trap). The sealed traps are returned to the GGA Hot Cell facility for disassembly. The traps will be opened in the Hot Cell and the contents transferred to clean glass containers for transfer to the laboratory for sorption measurements.
- (10) Select samples of fuel and blanket (usually entire pellet) for burnup analysis. Samples will be photographed after

sectioning and then submitted for fission-product and isotopic analysis (see Section 4G, below).

D. METALLOGRAPHIC EXAMINATION OF FUEL AND CLADDING

Perform metallography and record the following:

Grinding and polishing techniques

Etchant used.

Changes in transverse sections that are to be recorded should include

(1) Fuel rod:

Gross shape, such as distortion or wrinkling and cladding creep.

Fuel-cladding radial gap; compare with as-fabricated gap.

Smear density (estimated); compare with as-fabricated smear density.

Microstructural changes that would indicate fuel-cladding compatibility problem areas; if such changes are evident, microprobe analyses will be performed to identify the constituents of the reaction layers.

(2) Fuel changes:

Crack pattern formed during irradiation.

Pellet hole size, shape, and location.

Fuel radial movement.

Fuel grain growth and size.

Pore structure, size, and shape (round, lenticular).

Fuel structure and evidence of melting (location), if any.

Fuel structure, columnar grains (radius).

Fuel structure, equiaxed grains (radius).

Fuel density versus radius or structure.

Phase changes, stoichiometric changes, and Pu segregation or redistribution in the fuel.

Fission-product behavior via fission-product analysis, direct gamma counting techniques, microprobe analyses, spectro-photometric and mass spectrometric techniques.

(3) Cladding changes:

Dimensional measurements (± 0.0001 in.) (wall thickness—uniform or local thinning, particularly in regions where failure occurred, if any).

Wrinkling or ridging.

Grain structure, grain size, precipitates, and phases; diffusion of oxygen and/or fission products from the fuel at grain boundaries.

OD surface (roughness, decarburization, attack from sodium bond, etc.).

ID surface (chemical interactions with fuel or oxygen released from the fuel, fission-product effects on fuel-cladding reactions, etc.).

Cracked or failed areas.

Volume changes.

Changes in longitudinal sections that are to be recorded should include

(1) Fuel element changes:

Gross shape.

Hole size, longitudinal variations.

Fuel-cladding radial gap (± 0.0001 in.).

Pellet interface, cladding region, ratcheting, or longitudinal gapping.

Fuel-cladding compatibility problem areas, if any.

(2) Fuel changes:

Crack pattern, appearance of dish on pellets.

Hole shape, longitudinal.

Fuel axial movement, slumping.

Evidence of fuel plasticity.

Fuel structure.

Fuel swelling.

Phase changes (same as for transverse sections).

(3) Cladding changes:

Gross shape and dimensional measurements, evidence of cladding strain as a result of swelling.

Ridging, if any, at pellet interfaces.

Grain structure as a function of longitudinal location and operating temperature (especially at pellet interface regions).

OD surface (roughness, decarburization, attack from sodium bond, etc.).

E. MICROPROBE ANALYSIS

Perform electron or laser microprobe analysis to determine compositional changes as a result of migration of Pu or fission products or reaction of cladding with fuel, fission products, or sodium bond.

F. TENSILE TESTS

Perform ring-pull tests on cladding in the temperature range from room temperature to 800°C. Compare the results with controls aged at the irradiation temperature and heated for the same length of time as the duration of the irradiation test.

G. BURNUP ANALYSIS

Perform burnup analysis of the $(\text{Pu}, \text{U})\text{O}_2$ fuel, employing

- (1) Fission-product analysis; total Zr, and Nd^{148} .
- (2) Plutonium and uranium isotopic composition change analysis and determination of the plutonium-to-uranium ratio.

- (3) Comparison of burnup analysis with flux monitor dosimetry and thermal analysis data.
- (4) Study results of burnup analyses to determine spatial variations.
- (5) Gamma-scan analyses (see Section 4B5).

H. ANALYSIS OF FLUX MONITORS

Review flux monitors and perform dosimetry analyses to determine spatial variations in fluence (to be performed at GGA). Fluence measurements are to be performed by analysis for Mn^{54} activity in Fe-V alloy. Mn^{54} is formed by (n,p) reaction in Fe^{54} . The Mn^{54} emits a 840-keV gamma ray that can easily be monitored by an appropriate gamma detection system. Other dosimeters incorporated for measurements of neutron fluences are depleted U in the form of UC_2 , fully enriched U in the form of UC_2 particles and $(\text{Th,U})\text{C}_2$ particles, and Th in the form of ThC_2 particles. Both of these isotopes will fission with fast neutrons (> 1 MeV). Measurement of select long-lived fission-product gamma emitters (e.g., Cs^{137} , Ce^{144} , and Ru^{106}) will provide data for calculating the incident fluence. Al^{27} reacts with energetic neutrons to yield Na^{24} , via (n, α), which in turn decays to stable Mg^{24} . The magnesium can be determined by conventional chemical analysis, probably by atomic absorption techniques.

I. ANALYSIS OF PASSIVE TEMPERATURE MONITORS

1. Silicon Carbide

As described in Appendix A, the SiC temperature monitors store information relative to the average temperature to which they have been exposed during the last few full-power days of irradiation. Extracting this information from the monitors requires determination of dimensional change or lattice parameter change as a function of postirradiation annealing temperature. The radiation-induced changes start to anneal out when the annealing temperature passes the irradiation temperature. Since the high-purity SiC used will not be highly active, this work need not be done in a hot cell.

2. Kr⁸⁵ Release

As described in Appendix A, sections of fuel-rod cladding may be analyzed by Kr⁸⁵ release to determine the maximum temperature to which they have been exposed. As in the case of the SiC, this analysis is made by postirradiation annealing the sections. Release of Kr⁸⁵ introduced into the cladding by fission recoil will begin when the annealing temperature exceeds the irradiation temperature. It is necessary to remove all traces of fuel from the cladding prior to the annealing treatments. This can be done using concentrated nitric acid. This work may have to be performed in a hot cell because of the activity level in the stainless steel cladding.

J. FISSION-PRODUCT-TRAP ANALYSIS*

1. Postirradiation Examination of Charcoal in Sealed Traps

The F-1 capsule assembly contains sixteen sealed charcoal traps irradiated to estimated fast fluences (> 0.11 MeV) ranging from 6.8×10^{20} to 1.7×10^{22} neutrons/cm². The purpose of the sealed traps is to determine the effect of irradiation on the sorption properties of the charcoal independent of the effects of fission-product loading. It is the purpose of this section to describe the studies to be carried out on the charcoal material at the conclusion of the irradiation.

It is desirable to determine the effects of fast neutron fluence on the following properties of the charcoal. (It would be very desirable to measure the sorption of strontium or barium on the charcoal, but satisfactory techniques for sorption measurements with these species have not been developed.)

- (a) The Xe sorption capacity of the charcoal at 300°, 350°, and 400°C.
- (b) The Kr sorption capacity of the charcoal at 300°, 350°, and 400°C.
- (c) The Cs sorption capacity of the charcoal at 300°, 350°, and 400°C.

* A more detailed description of fission-product behavior is given in Appendix B.

Each charcoal sample can be utilized for measurements of the sorption of each of the three species but only at a single temperature.

It is planned to measure, in sequence, the krypton, xenon, and cesium sorption on each of the charcoal samples under the above temperature conditions for the various neutron fluences experienced by the capsules. This matrix is designed to provide the maximum reliability of data on the temperature dependence and fast fluence while still providing for duplicate samples at several irradiation levels and temperatures.

2. Postirradiation Examination of the Active Charcoal Traps

The F-1 irradiation capsule contains three fuel rods (G-4, -6, -7) in which charcoal traps approximately 10 in. long are exposed to the gaseous and volatile fission-product effluents from the fueled portion of the rod. These traps will form a permanent "sink" for the volatile species. Thus, the long-lived (and stable) I, Cs, and Te, which leave the fuel and diffuse past the blanket region to reach the charcoal trap by gaseous diffusion, are expected to deposit close to the entrance to the trap and be retained permanently. The volatile oxides of long-lived Ru, Tc, and Mo may also diffuse to the charcoal and deposit in the trap.

The behavior of the volatile fission products is in contrast to that of the short-lived and the long-lived (and stable) gaseous fission products. The latter will equilibrate throughout the internal clad volume and will exert a pressure of ~10 to 15 atm (depending on the relative amount sorbed in the charcoal).

The short-lived noble gases following release from the fuel will migrate rapidly by convection and thermal diffusion out of the fuel zone to the carbon traps. The diffusional delay in the trap material will be substantially greater than in the fuel or other regions because of sorption and tortuosity effects, and also because the temperature is lower. Because of the long delay times in the trap, short-lived noble gases will tend to decay therein, depositing solid daughter products and exhibiting an exponential axial concentration gradient, which is dependent on the species half-life and the effective diffusion coefficient. In principle, therefore, measurement of

the axial concentration gradient of the solid radioactive daughters in the trap will yield information on the effective diffusion coefficients of the gaseous parent species. The expected profiles for the solid daughter products of the short-lived gaseous fission products will be calculated with the SLIDER code⁽⁸⁾ using an extension of the procedure employed for the 04-P9 capsule⁽¹⁾ and any additional information gained from irradiation of the P9 capsule. It is an objective of postirradiation analyses of the active traps in the F-1 capsules to compare the actual profiles with the calculated profiles.

The potential for nonuniform mixing of the charcoal in each section during postirradiation handling and disassembly of the capsules is a complication which may perturb the data. Thus, a gamma scan of the active trap regions of the individual fuel rods at the earliest possible moment during disassembly is desirable. This gamma scan should be made directly on each rod through a narrow slit after separation of the rod from the bundle. During this gamma scan, the rod should be in its normal vertical orientation. The gamma scan should be made with a 4096-channel analyzer using a lithium-drifted germanium detector.

Disassembly of the active charcoal traps will be carried out at the GGA Hot Cell facility. The traps are removed from their containers and again gamma-scanned with a 4096-channel analyzer using a lithium-drifted germanium detector. The purpose of this second gamma scan is to determine whether there has been any significant movement of charcoal during handling, impregnation, and shipping. The traps will then be opened, the paraffin removed by a solvent and the charcoal sections placed in vials for radiochemical analysis of the individual sections.

Appendix A

TEMPERATURE MONITORING

A.1. SILICON CARBIDE TEMPERATURE MONITORS

Principle of SiC Temperature Monitoring

In 1961, Pravdyuk, et al.,⁽⁸⁾ reported that the radiation-induced lattice expansion of SiC starts to anneal out when the postirradiation annealing temperature exceeds the irradiation temperature and suggested that this property might be used as the basis for an irradiation temperature monitor. Measurements of dimensional change during postirradiation annealing, reported by Thorne, et al.,⁽⁹⁾ confirmed the observation and established that isochronal annealing times of 1 to 2 hr at each temperature were sufficient to establish the shape of the isochronal annealing curve as essentially linear with respect to temperature once the irradiation temperature is passed (Fig. A1). This property enables the position of the break in the annealing curve to be found graphically with good precision. Subsequent instrumented irradiations in the Dounreay Fast Reactor (DFR)⁽¹⁰⁾ and the Oak Ridge Reactor⁽¹¹⁾ confirmed that the break occurs at the irradiation temperature. Reported data from instrumented samples are summarized in Fig. A2. Results from uninstrumented samples irradiated in the Dounreay Fast Reactor⁽¹²⁾ showed that when the reactor was shut down slowly (over an 11 hr period), the break in the postirradiation annealing curve took place in two steps, with the higher temperature step indicating the full-power operating temperature.

There is no theoretical reason why postirradiation isochronal annealing curves should be linear with a sharp "knee." The kinetics of annealing radiation damage are complex and involve processes, with a range of frequency factors or activation energies or both. Under these conditions isochronal annealing curves are normally S-shaped, with the width of the

Fig. A1 Lattice parameter of irradiated silicon carbide as a function of postirradiation annealing temperature

Fig. A2 Irradiation temperature indicated by silicon carbide monitor vs. temperature measured by thermocouple

sloping part depending on the width of the spectrum of frequency factors or activation energies. It has been reported that when dimensional changes in irradiated silicon carbide are recorded continuously during a linearly programmed temperature rise,⁽¹⁰⁾ the "knee" becomes poorly defined, making the curve closer to the expected sigmoidal annealing curve. Thus, the presence of a sharp "knee" at the irradiation temperature in isochronal annealing curves, for annealing times of about 1 hr, must be regarded as a lucky coincidence, which nevertheless has been well established empirically in a number of irradiation facilities with widely different neutron fluxes.

Reliability of the Measurement

In cases where the actual operating temperature was known, the indicated temperature was remarkably close to the measured temperature (see Fig. A2). All data points fell within 30°C of the measured temperature and the mean deviation of the indicated temperature for the measured temperature was only 13°C. At higher temperatures the radiation-induced property changes are smaller and the slope of the isochronal annealing curves past the "knee" is small, reducing the precision with which the irradiation temperature may be found. At irradiation temperatures of 800° to 1000°C, the precision has been estimated to be $\pm 100^\circ\text{C}$ if length changes are used, and possibly $\pm 25^\circ\text{C}$ if lattice parameter measurements are used.⁽¹⁰⁾ However, there do not appear to be any reported measurements from instrumented samples irradiated above 760°C.

A slow shut down during which neutron damage is accumulated at temperatures below the operating irradiation temperature would seriously affect the accuracy of the measurement. In a Dounreay Fast Reactor experiment in which the power was reduced over 11 hr,⁽¹²⁾ the postirradiation annealing curves of the silicon carbide monitors showed a multi-stepped annealing curve from which the true irradiation temperature was difficult to determine. In EBR-II it is probable that a final shut down extended over more than a few hours would reduce significantly the reliability of the temperature measurement, and if extended over longer periods (a day or two) the information in the monitors would be lost.

A.2. Kr⁸⁵ RELEASE

The inner surface of fuel-rod cladding is subject to bombardment by recoiling Kr⁸⁵ during irradiation. During postirradiation annealing, the release of Kr⁸⁵ starts when the irradiation temperature in the last few hours is exceeded. As a backup or to gain more knowledge of temperatures in the fueled region where the location of SiC monitors is difficult owing to the large thermal gradients across the monitors and the possible perturbation in radial heat transfer, this technique seems well worth pursuing. The method is less well established experimentally than SiC monitors, and the samples will be very radioactive, necessitating shielding or hot cell analysis; however, some experiments on DFR fuel cladding have indicated a 25°C agreement with temperatures determined from a coolant channel thermocouple and known reactor power profiles.⁽¹³⁾ Some experience would need to be gained with this technique and samples could be readily exposed in the King furnace in the GGA TRIGA reactor with accurately measured temperatures, and then subjected to annealing.

Appendix B
FISSION-PRODUCT-TRAP BEHAVIOR AND ANALYSIS

B.1. PURPOSE AND OBJECTIVES

The transport and release of volatile and gaseous fission products is of considerable importance to the design and operation of a power plant. In the normal operating condition of the plant, the distribution of volatile fission products within the elements will be governed first by solid-state diffusion of the fission products through the fuel lattice into the pores and interstices of the fuel and then by gaseous diffusion through the holes and clearances in the fuel, axial blanket, and internal trap, if any. The F-1 capsules are designed to provide information on the desirability of incorporating activated carbon traps into fuel rod designs and as a tool to determine the release and distribution of gaseous and volatile fission products within the fuel rods.

The F-1 capsules contain two types of activated carbon traps: "active" traps, which are included in three of the eight fuel rods, and "passive" or sealed traps, which are included in all eight rods.*

The active traps which are approximately 10 in. long are placed above and below the blanket regions of the rods. They are exposed to the gaseous and volatile fission-product effluents from the fueled portion of the rod and provide a large surface area for sorption and condensation of fission products.

The charcoal provides a site within the reactor core for the sorption and permanent retention of condensable fission products. The objective of this type of trap is to provide information on the ability of hot activated carbon to remove condensable fission products and to delay the transport of short-lived gaseous fission products. These data are needed for the proof-of-principle of the utility of such traps and for detailed design of charcoal traps in fast breeder fuel rods.

*Only seven fuel rods are loaded in the fuel bundle at a given time.

The inclusion of the passive traps in each of the fuel rods will permit evaluation of the effect of the radiation environment on the sorption characteristics of charcoal, independent of the loading of the charcoal with condensable fission products. Little is known about the sorption capacity for gases of activated carbon that has been subjected to irradiation. Some data are available for the cesium sorption capacity of irradiated graphite, which is known to increase under irradiation.⁽¹⁴⁾ The Cs sorption capacity of activated carbon appears to decrease under irradiation, but the data are too sparse for firm conclusions to be drawn.⁽¹⁴⁾

A logical step in the determination of fission-product migration and release is the measurement of fission-product distribution in irradiation capsules. Vented rods in which the fission-gas release is continuously monitored would yield the most information on noble-gas release. The P9 capsule experiment presently being irradiated in the ORR is an example of this approach. The ORR, however, does not supply the fast-flux environment necessary for fast breeder evaluation.

The EBR-II reactor does fulfill this important criteria. The first fast flux test of the F-1 rods, however, is of the nonvented variety. This means that information on fission-product migration, diffusion, and sorption must be gleaned solely from postirradiation examination.

This does not detract entirely from the usefulness of such a test. In fact, a sealed rod experiment will provide data on transport of fission products strictly by gaseous diffusion, unfettered by problems associated with pressure cycling (with subsequent enhanced release), which will undoubtedly occur in vented rod tests.

B.2. FISSION-PRODUCT BEHAVIOR IN THE F-1 CAPSULE

In principle, the behavior of condensable fission products, such as I, Cs, Te, Ru, and Tc, in the sealed rod in EBR-II is expected to be essentially identical with that in the vented rod test. These fission products, will, in general, be sorbed or condensed in the first centimeter or so of the carbon trap. Thus, the long-lived (and stable) I, Cs, and Te, which reach the trap by gaseous diffusion from the fuel through the blanket region to

the trap, are expected to deposit close to the entrance to the trap and be retained permanently. The volatile oxides of long-lived Ru, Tc, and Mo may also diffuse to the charcoal and deposit in the trap.

The extent and rate of migration of the volatile fission products from the fuel, past the blanket zone, and into and within the carbon trap are essential design information. These data are required to (1) assess the decay heat load expected in the charcoal traps, (2) to assess the potential for plugging of the system by deposition of volatile species in the colder regions, and (3) to determine the effect that loading of the traps by condensable species would have on the Kr and Xe sorption.

The behavior of the volatile fission products is in contrast to that of the short-lived and the long-lived (and stable) gaseous fission products.

The noble gases will diffuse throughout the trap material. According to diffusion theory, each gaseous isotope diffuses separately; therefore, internal carbon traps will act as sinks for the short-lived Xe and Kr isotopes and have little effect on the long-lived or stable species. The latter will equilibrate throughout the internal cladding volume and will exert a pressure of ~10 to 15 atm (depending on the relative amount sorbed in the charcoal).

The short-lived noble gases, following release from the fuel, will migrate rapidly by convection and thermal diffusion out of the fuel zone to the carbon traps. The diffusional delay in the trap material will be substantially greater than in the fuel or other regions because of sorption and tortuosity effects and also because the temperature is lower. Because of the long delay times in the trap, short-lived noble gases will tend to decay therein, depositing solid daughter products and exhibiting an exponential axial concentration gradient, which is dependent on the species half-life and the effective diffusion coefficient. In principle, therefore, measurement of the axial concentration gradient of the solid radioactive daughters in the trap will yield information on the effective diffusion coefficients of the gaseous parent species.

In practice, however, in the F-1 rods the gaseous diffusion coefficients will vary considerably throughout capsule life, as the gas composition and pressure varies. The rods will initially be seal-welded with 1.3 atm He inside. When the rods are just brought to power, the diffusing medium will be He at about 2 atm (assuming the average rod temperature is 450°C). Near the end of the irradiation, the rod pressure is calculated to be ~200 psi or 13.6 atm (75% Xe, 7% Kr, and 18% He). Calculations of the interdiffusion coefficients of Xe and Kr under the varying conditions of the F-1 rods indicate that noble gas diffusion rates could vary by up to a factor of 50 throughout the capsule life.

These calculations are the results of a preliminary analysis and do not take into account conditions within the rods that may cause the effective diffusion coefficients at end of life to be higher than predicted and, hence, decrease the discrepancy. Factors which could increase D_{eff} at end of life are (1) a decrease in specific adsorption, K' , due to high sorbate partial pressures, solid fission-product contamination or fast neutron damage, (2) increase in trap temperature as fission products build up therein, and (3) increase in axial trap temperature gradient causing an increase in thermal diffusion effects.

Appendix C
COMPUTER PROGRAMS

C.1. TAC2D

TAC2D is a code for calculating steady-state and transient temperatures in two-dimensional problems by the finite difference method. It is written entirely in Fortran V language. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular (polar) coordinate system by orthogonal lines of constant coordinate called grid lines. The grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its capacitance, heat generation, and heat flux paths to neighboring nodal points. A system of these equations is solved by an implicit method, which is the most efficient known at this time.

Some advantages of the code are that

1. The geometrical input is simple.
2. The input of thermal parameters is by Fortran V arithmetic statement functions. Many of the calculation variables (time, local temperature, local position, etc.) are available for use in these functions.
3. Internal and external flowing coolants may be used.
4. There may be internal and external thermal radiation.
5. There is a wide selection of optional output.

The principal limitations of the code are that

1. The grid-line system must be orthogonal in the rectangular, cylindrical, or circular coordinate system. Therefore, the sides of the nodal elements must also be orthogonal. The entire problem must be bounded by four grid lines in one of the coordinate

systems. Difficulties in treating irregular boundaries can be overcome to some extent through the use of materials having specially chosen properties.

2. All radiation is treated one-dimensionally.
3. There are no provisions for thermal expansion or change of phase. Such special heat-transfer situations could be included by extensions of the existing programming.

TAC2D has been assigned operational status. The machine requirement is a 65K Univac 1108, or equivalent. In addition to input-output, a maximum of four and a minimum of no tapes are required depending upon the code options being used. The operating system under which the code has been successfully used is in EXEC II as modified for Gulf General Atomic. Running time depends upon the size of the problem and is not easily defined.

A related code, TAC3D, is three-dimensional and has all the features of TAC2D described in this abstract. All symbolic elements, test cases, a users manual, and a descriptive report are available through the Argonne National Laboratory Code Center for each of the two codes.

The thermal model used for obtaining the temperatures of the internal parts of the capsule is illustrated on Fig. C1. The capsule dimensions were obtained from the detailed capsule drawings. It was assumed that all of the heat generated in the capsule is removed by its own coolant and that there is no heat loss to surrounding assemblies.

The axial power distribution was taken from "Guide for Irradiation Experiments in EBR-II." The γ heating was 1.0 watt/g for all of the materials. The flux variations across the seven-rod subassembly were obtained from Table C-I in the EBR-II experimenters guide by using the values for U^{235} . The coolant inlet temperature was assumed to be 700°F with a subassembly flow rate of 17 gpm.

The heat transfer between the capsule wall and the coolant was calculated using the following equation:

$$h(\text{Btu/hr-ft}^2\text{-}^\circ\text{F}) = \frac{K}{D} [4.9 + 0.0175 (\frac{G \times D}{K})^{0.8}]$$

Fig. C1 R-Z model of F-1 capsule

which is obtained from Eq. (9-7A) of Ref. 16, page 215, and is low for use with sodium.

Thermal radiation is incorporated into the calculation of the ΔT across the fuel-cladding gap. Grey-body characteristics were assumed for the surfaces of these materials.

C.2. BRITL

The BRITL computer program was developed to study interactions between fuel and cladding in cylindrical oxide fuel elements under the action of either a thermal or fast neutron flux. It differs from other existing codes principally in the allowance for initial cracking to relieve the thermal tensile stresses resulting from the high internal temperature gradients.

The fuel rod is composed of fuel pellets, which may contain a central hole, inside a cylindrical metallic cladding. After heating to a steady temperature distribution, the fuel is assumed to restructure as shown in Fig. C2. Fuel heated above 2150°C forms an inner zone of columnar grain structure. Outside this zone to a temperature of about 1650°C the grain structure is equiaxial. The densified material in these two regions moves outward to produce a larger central hole. The outer zone of material is subjected to stresses because of differential thermal expansion, resulting in radial cracks as shown. The initial hot gap between the fuel and the cladding is determined by the installed gap and the relative expansions of the outer fuel region and the cladding.

The fuel matrix expands, as a result of retained gaseous and solid fission products, until the initial gap between fuel and cladding disappears. After this, the expansion is resisted by cladding pressure. The cladding is distorted by swelling and by creep and elastic deflection under the influence of the fuel pressure and internal gas pressure. In the fuel region near the cladding, in which radial cracks assure no circumferential

Fig. C2 Fuel rod after heating to steady temperature

stresses, the stress in the fuel will be purely radial. That is, each wedge-shaped piece of fuel will be in equilibrium under the effect of radial pressures on its inner and outer surfaces. The inner densified region is assumed to be homogeneous.

A completely general treatment of the fuel distortion problem would include swelling, densification or sintering, thermal stress, elastic and plastic strain, and viscous or creep flow. The situation is further complicated by the well-recognized phenomenon of brittle cracking. A number of simplifications have been made in this analysis to allow inclusion of cracking effects:

1. Radial cracking is assumed to relieve all thermal stresses at the beginning of the irradiation. Since the various effects of radiation damage and fuel densification appear to result in little net change in thermal conductivity, it is further assumed that the temperature distribution will not change during the irradiation. Therefore, thermal stress effects are eliminated.
2. Swelling of fuel is introduced as a function of fission rate and temperature, based on a correlation of available data. A fraction of the fuel swelling can be assigned at the start to fill pores. The fraction will be reduced as the pores are filled up. Cladding swelling is introduced as an empirical function of neutron fast flux and temperature.
3. Sintering volume change is calculated from the normal stresses, fuel viscosity, and initial porosity by the following method.⁽¹⁵⁾

$$\frac{d\rho}{dt} = - \frac{3}{4} \frac{\sigma}{\eta} (1 - \rho)$$

in which σ is the average normal stress, ρ is the fraction of theoretical density, and η is the effective fuel viscosity.

4. In the axial direction it is assumed that the fuel expands uniformly with zero net axial force until the fuel-cladding gap disappears. After this time it is assumed that fuel and cladding

expand axially at the same rate. This would be strictly true for a long fuel column with no slip between fuel and cladding.

5. Since relatively low stresses occur in most of the fuel, it has been assumed that plastic strain would not be important. Creep strain rate, at these low stress levels, has been assumed to be proportional to deviatoric stress, occurring without change in volume.

REFERENCES

1. Lindgren, J. R., et al., "Planned Thermal Irradiation of Manifolded-Vented (U,Pu)O₂-fueled Rod in ORR Capsule P-9," USAEC Report GA-9896, Gulf General Atomic, March 15, 1970.
2. Craig, C. N., G. R. Hull, and W. E. Baily, "Heat Transfer Coefficients between Fuel and Cladding in Oxide Fuel Rods," USAEC, Report GEAP-5748, General Electric Co., January 1969, p. 2.
3. Mikailoff, H., et al., "Irradiation Results of Mixed UO₂-PuO₂ Test Pieces for Fast Neutron Reactor Fuel Elements," US-Euratom Fast Reactor Exchange Program, Report EUFNR-707 (EUR 4323 f) October 1969.
4. Thompson, W. I., "BRITL, A Fuel Rod Analytical Model for Thermal or Fast Flux Irradiations," USAEC, Report GAMD-10007, Gulf General Atomic, April 1, 1970.
5. Simmons, W. F., and J. A. Van Echo, Report on the Elevated-temperature Properties of Stainless Steel, ASTM, Data Series Publications DS5-S1.
6. Nuclear Applications and Technology, Vol. 9, July 1970.
7. Neimark, L., Argonne National Laboratory, private communication.
8. Pravdyuk, N. F., et al., Properties of Reactor Materials and the Effects of Radiation Damage, D. J. Littler, ed., Butterworths, London, 1962, p. 57.
9. Thorne, R. P., V. C. Howard, and B. Hope, Proc. Brit. Ceram. Soc., No. 7, 1967, p. 449.
10. Bramman, J. I., A. S. Fraser, and W. H. Martin, International Conference on Fast Reactor Irradiation Testing, Thurso, Scotland, April 14-17, 1969, Paper No. 7/7.

11. Kennedy, C. R., Oak Ridge National Laboratory, private communication.
12. Martin, W. H., and A. M. Price, J. Nucl. Energy, 21, 1967, p. 359.
13. Clarke, D., "The Determination of Reactor Fuel Cladding Temperature by Recording Krypton-85 Release During Postirradiation Heating," UKAEA, TRG Report 1736 (D), 1968.
14. Public Service Company of Colorado, "330 MW(e) HTGR Reactor, Research and Development Program Quarterly Progress Report for the Period Ending June 30, 1967," USAEC, Report GA-8038, Gulf General Atomic, July 28, 1967, p. 175.
15. Wolfe, R. A., and S. F. Kaufman, "Mechanical Properties of Oxide Fuels," USAEC Report WAPD-TM-587, Westinghouse, Atomic Power Division, October 1967.
16. McAdams, William H., Heat Transmission, 3d ed, McGraw-Hill, New York, 1954.