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EXPERIMENT DESCRIPTION AND HAZARDS
EVALUATION FOR THE IRRADIATION OF FBR
PHYSICS CONSTANTS SAMPLES IN THE EBR-II

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ABSTRACT

The purpose of this experiment is to measure reactivity changes in mixed-oxide fuels as a result of irradiation in a typical fast neutron spectrum. Representative samples of such fuels were obtained and the EBR-II was used as the irradiation source because its flux spectrum resembled closely that of some fast breeder reactor concepts and was similar to the spectrum of the Coupled Fast Reactivity Measurement Facility in which the reactivity changes were to be measured.

This report describes the irradiation samples and documents the hazards evaluation required for such irradiations. The format is that specified in the Guide for Irradiation Experiments in EBR-II (Feb. 1971 edition).

Fabrication and evaluation of the samples are described in the report ANCR-1001 (Oct. 1971).

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EXPERIMENT DESCRIPTION AND HAZARDS EVALUATION FOR THE IRRADIATION OF FBR PHYSICS CONSTANTS SAMPLES IN THE EBR-II

by

Robert G. Nisle

I. INTRODUCTION

This report is being written to document the study that was made to evaluate the safety aspects of the irradiation of small mixed-oxide samples in the EBR-II. The data packages that were submitted to the Argonne National Laboratory to obtain approval for the irradiation of these samples were prepared in accordance with two different editions of the Guide For Irradiation Experiments in EBR-II. This report is a consolidation of the information contained in those two data packages in the interest of continuity. The format is that used by the February 1971 edition of that guide. Numbers in parentheses refer to the corresponding paragraphs in the Guide. Inapplicable paragraphs have been omitted.

Fabrication of the samples is described in a previous publication (ANCR-1001, dated Oct. - 1971). Figures 1 and 2 are pictorial representations of the individual samples and the assembled sample string.

II. PURPOSE OF EXPERIMENT

The purpose of the experiment is to measure the reactivity changes in mixed-oxide fuels as a result of irradiation in a typical fast neutron spectrum. The samples consist of a series of mixtures of PuO_2 and UO_2 . These mixtures were compacted, sintered, and machined to closely controlled dimensions. About six grams of these mixtures were loaded into stainless steel capsules as shown in the figures. Pre- and postirradiation measurements of reactivity will be made on these samples in the Advanced Reactivity Measurements Facility I (ARMF-I) and in the Coupled Fast Reactivity Measurements Facility (CFRMF). Measurements in the ARMF-I will be used to evaluate the changes in fissile and absorber content as a result of irradiation. Measurements in the CFRMF will be used to evaluate gross reactivity changes in a typical fast reactor neutron spectrum.

Approval in principle for these irradiations is contained in an AEC letter from E. E. Sinclair to C. M. Rice dated January 15, 1970.

III. EXPERIMENT DESCRIPTION

A. Objective of Experiment

The following is quoted from the Request for Approval in Principle that was approved by AEC letter RDT:RT:F-008 to C. M. Rice from E. E. Sinclair, dated January 15, 1970.

"As a part of a continuing program whose goals are to provide information on fission product cross sections, the present irradiation experiment addresses itself to the gross cross section of accumulated fission products. A series of reactivity measurements as a function of

FIGURE 1. Sample capsule for the irradiation portion of the FBR Physics Constants program.

FIGURE 2. Assembly view of sample capsules loaded in a B-19 type EBR-II irradiation capsule.

neutron fluence in a critical facility whose spectrum is a good approximation to the irradiation spectrums will provide nuclear designers with immediate data."

B. Prior Irradiation Experience

The following is quoted from the same Request for Approval in Principle.

"The experimenters have much experience in thermal reactor irradiation work. Specific examples of material involved in irradiation studies include ^{235}U , natural UO_2 , ^{239}Pu , ^{240}Pu , thorium, hafnium, dysprosium, and europium. All of these have combined the use of interim reactivity measurements as a non-destructive burn-up monitor."

A sample of 20% PuO_2 , 80% UO_2 is currently being irradiated in 7C3 in the EBR-II.

C. Irradiation Conditions

Three reactor cycles of about thirty days continuous irradiation for each of four sample strings are contemplated. The preferred location is 7N4. The calculated burn-up in that location is 2.14%, based on fission rates given in Table C-I in the Guide. The sample string is contained in a B-19 pin, and the remaining 18 locations in the subassembly are to be filled with dummy stainless steel rods. Table I lists the four different PuO_2 - UO_2 mixtures and the calculated heating rates.

TABLE I

ESTIMATED IRRADIATION CONDITIONS

<u>Experiment Number</u>	<u>Nominal Composition</u>	<u>Fission Rate (Table C-I)</u>	<u>Fission Plus Gamma Heating Rate</u>
I-11-0	100% UO ₂	1.2 x 10 ¹² fiss/sec/capsule	5.32 x 10 ³ $\frac{\text{Btu}}{\text{hr-in}^3}$
I-11-1	10% PuO ₂ in UO ₂	3.5 x 10 ¹² fiss/sec/capsule	11.88 x 10 ³ $\frac{\text{Btu}}{\text{hr-in}^3}$
I-11-2	20% PuO ₂ in UO ₂	6.3 x 10 ¹² fiss/sec/capsule	20.63 x 10 ³ $\frac{\text{Btu}}{\text{hr-in}^3}$
I-11-3	30% PuO ₂ in UO ₂	11.5 x 10 ¹² fiss/sec/capsule	35.67 x 10 ³ $\frac{\text{Btu}}{\text{hr-in}^3}$

With a coolant flow rate of 2 gpm the bulk temperature rise for the 20% sample was calculated to be 120°F. The measured flow rate in I-11-2 (ANL letter, W. J. Larson, dated Sept. 22, 1971) was found to be 16.5 gpm at this flow rate the bulk temperature rise for the 30% sample would be 14°F.

D. Out-of-Reactor Services

Standard procedures only to remove the center pin from the B-19 assembly will be required. The sample pin will then be shipped via cask to a Hot Cell on site to be designated at a later date.

E. Capsule Identification

The following is the sample identification system adopted for this series of irradiation:

I - XX - YZ

I = Letter designation assigned by Argonne to Aerojet Nuclear Company experiments.

XX = 11, Number assigned to irradiation pin with stainless dummy pins
in remaining 18 locations

XX = 12, Number assigned to irradiation pin with depleted uranium in
remaining 18 locations.

Y = 0, 100% UO_2

Y = 1, 10% PuO_2 in mixed oxide sample

Y = 2, 20% PuO_2 in mixed oxide sample

Y = 3, 30% PuO_2 in mixed oxide sample

Z = A, B, sequential irradiations of Y series

The sample currently being irradiated has the identification number
I-11-2A.

F. Subassembly Loading

The sample pin (B-19) is to be loaded in the center position of the
subassembly with the remaining 18 positions filled 304 stainless dummies.
A 7N4 location in the reactor is preferred.

G. Flux Monitor Description

A foil package for the measurement of flux level and spectrum is
inserted in the depression at one end of the individual sample capsules.
The foil package consists of minute quantities of the following elements:
copper, iron, nickel, chromium, titanium, indium, aluminum, and zinc. The
iron, nickel, and chromium are derived from small pieces of stainless steel
and the remaining elements are cast in a specially prepared alloy.

H. As-Built Description

1. The identification of the sample strings and chemical
composition are given in Table I (1a and 1b).

2. The oxygen/metal ratio is 0.134 in all samples (1c).
3. The isotopic composition is given in Table II (1d).

TABLE II

ISOTOPIC COMPOSITION

<u>Isotope</u>	<u>10% PuO₂ Sample</u>	<u>20% PuO₂ Sample</u>	<u>30% PuO₂ Sample</u>
²³⁸ Pu	0.09%	0.11%	0.03%
²³⁹ Pu	86.03%	88.27%	90.78%
²⁴⁰ Pu	11.18%	9.41%	8.31%
²⁴¹ Pu	2.35%	2.02%	0.80%
²⁴² Pu	0.28%	0.10%	0.07%
²³⁴ U	0.004%	0.003%	0.004%
²³⁵ U	0.414%	0.408%	0.414%
²³⁶ U	0.007%	0.007%	0.007%
²³⁸ U	99.575%	99.582%	99.575%

-
4. The active material is in the form of pellets 0.213" in diameter and 0.250" long. Four of these pellets are loaded into one capsule. Five such capsules are loaded into one sample string (H,1e).
  5. The stainless steel capsules shown in Figure 1 are loaded and evacuated in a glove box. The glove box is then flushed with He at atmospheric pressure (12.4 psia at the local altitude). The seal is then made by TIG welding in the helium atmosphere (H,1f).
  6. The pellet density is 95.7% of theoretical (H,1g).

7. Four pellets make one inch of fuel per capsule, five capsules per sample string (H, 1i).
8. Total fuel weights for each of the sample strings are (H, 1j)

I-II-0	30.630 g	UO ₂ (depleted U)
I-11-1	2.178 g PuO ₂	28.343 g UO ₂ (depleted U)
I-11-2	6.049 g PuO ₂	23.941 g UO ₂ (depleted U)
I-11-3	9.828 g PuO ₂	21.452 g UO ₂ (depleted U).
9. Sorbed gas and water vapor removed by evacuation as described in (H5) above (H, 1k).
10. The cladding is 304 SST (H, 1l).
11. Cladding was not cold worked (H, 1m).
12. Cladding outside diameter is 0.325" (H, 1n).
13. Cladding thickness is 0.053" (H, 1o).
14. Thermal bond is helium (H, 1p).
15. The fuel/cladding annulus is 0.003" wide.
16. The gas space has a volume of 0.0589 cc.
17. The ratio of gas volume to fuel volume = 0.1009.
18. Plenum gas is same as thermal bond (helium).
19. Other items in the sample string are (H, 1v).
  - a) Five empty capsules, identical except for loading, with loaded capsules; weight 10.5 g each.
  - b) Two spacer rods:

Bottom rod = 19.687" long	208.76 g.
Top rod = 19.240" long,	204.02 g.
  - c) Compression spring at top, compressed from 1.68" to 1.371".
20. Individual capsule length = 1.657" (H, 1x).
21. Individual capsule weights:

loaded = 16.3 g	}	(H, 1w)
empty = 10.5 g		

22. Gamma activity at one foot is less than background (H, 1y).
23. No spacer wires used on individual capsules (H, 1z).
24. Plenum volume inside B-19 pin = 12.23 cc (H, 2ab).
25. Plenum gas is helium (H, 2ac).
26. Final assembled weight to be measured by EBR-II personnel (H, 2ad).
27. Assembled B-19 pin length = 60 7/8"
diameter = 0.375" (H, 2ae).
28. Spacer wire to be added to B-19 pin by EBR-II personnel (H, 2af).

I. Thermal Analysis

Peak and total heat production rates are given in Table III (I, 1a, and 1b).

TABLE III

HEAT PRODUCTION RATES

<u>Experiment Number</u>	<u>Peak Rate Center of Pellet</u>	<u>Total Production for Entire String</u>
I-11-0	0.67 kW/ft	277.2 W = 946.8 Btu/hr
I-11-1	1.50 kW/ft	623.1 W = 2127.7 Btu/hr
I-11-2	2.58 kW/ft	1075.9 W = 3674.2 Btu/hr
I-11-3	4.47 kW/ft	1766.4 W = 6032.3 Btu/hr

In addition to these sources, gamma heat sources in the cladding and in the sodium coolant were included in the HEATING program to calculate internal temperatures, as follows:

304 SST (cladding)	1329.1 Btu/hr/in. ³
Sodium coolant	139.8 Btu/hr/in. ³

The calculated temperatures are given in Table IV.

The local coolant temperature varied from 700 to 900°F along the length of the sample string.

TABLE IV

INTERNAL TEMPERATURES CALCULATED BY THE HEATING CODE

Experiment Number	Center Line [2-30]	<u>Peak Temperatures^(a)</u>			
		Inner Cladding [7-30]	Outer Cladding [10-30]	Inner Capsule [12-30]	Outer Capsule [14-32]
I-11-0	1243°F	1037°F	1024°F	380°F	878°F
I-11-1	1364°F	1046°F	1018°F	584°F	880°F
I-11-2 ^(b)	1609°F	1011°F	931°F	323°F	900°F
I-11-3	2341°F	1386°F	1307°F	901°F	879°F

(a) See Figure 3 for the model used in the HEATING code. The numbers in [ ] are the R-Z mesh points shown in the figure.

(b) Apparent inconsistencies in temperatures shown for the I-11-2 experiment are due to the fact that a different model was used with the HEATING code. Some values shown are probably too low.

J. Thermal Analysis of Subassembly

1. Total heat generation rate in the subassembly includes fission and gamma heating in the sample and gamma heating in the 18 remaining dummy pins. The heating rates in the four subassemblies identified by the sample strings they contain are:

I-11-0	21,241.0 Btu/hr	
I-11-1	22,420.0 Btu/hr	
I-11-2	23,924.0 Btu/hr	
I-11-3	29,045.0 Btu/hr	(J,a).

2. A 120°F bulk temperature rise was calculated for the 20% fuel and a coolant flow rate of 2 gpm. The measured flow rate in this same subassembly was found to be 16.5 gpm (ANL letter W. J. Larson, dated Sept. 22, 1971). When a bulk temperature rise was calculated for the 30% fuel and the 16.5 gpm rate, a value of 14°F was obtained. It appears, therefore, that a detailed calculation for each of the four cases will yield little more than an arbitrary set of numbers for the bulk temperature rise, depending on the coolant flow rate assumed. The above indicates then the range of temperatures available by adjusting the coolant flow rate (J,b).

K. Stress and Strain Analyses

1. Plenum pressures and cladding and capsule stresses were calculated from the following data.

Load pressure = 1 atm = 12.4 psia at local altitude.

Load temperature = 68°F = 528°R

Operating temperatures, volumes, and corrected pressures are given in Table V. In this table the operating pressures and differential pressures were calculated by use of the following equations.

$$\text{Gas volume} = \frac{\text{Atoms fissioned} \times 22.4 \times 10^3}{6.02252 \times 10^{23}}$$

$$\text{Gas pressure} = \frac{(\text{Gas volume} + 0.0589) \times 1 \text{ atm}}{0.0589}$$

$$\text{Temperature correction} = \frac{\text{Operating temperature}}{528}$$

Operating pressure = Gas pressure x temperature correction.

The thermal stress was calculated by means of formulas given in Thermal Stress Techniques by Zudans, et al., Elsevier, N. Y.

(1965). The resulting stresses were found to be given by

$\sigma_a = 167.36 \Delta T$ (Compression) at the inner surface, and

$\sigma_b = 129.18 \Delta T$ (Tension) at the outer surface, for the cladding; and

$\sigma_a = 168.23 \Delta T$ (Compression) at the inner surface,

$\sigma_b = 156.03 \Delta T$ (Tension) at the outer surface,

for the B-19 pin. ΔT is the temperature difference between inner and outer surfaces. The calculation of thermal and pressure stresses is outlined in Table VI.

(K, 1a, 1b, 1c, and K, 2f, 2g, 2h)

2. The Reactor Handbook, Vol. I, Materials quotes a creep rate of 1% in 10k hours at 1000°F and 12k psi stress. The worst case in Table VI is the 30% sample which show total stresses as follows:

Inner surface: $-13,221 + 1623 = -12,598$ psi (compression)

Outer surface: $10,205 + 1623 = 11,828$ psi (tension)

TABLE V

STRESS - STRAIN DATA

Experiment Number	Gas Volumes Std. Conditions	Operating Temperatures		Operating Pressures		Differential Pressures	
		Inner Capsule [6-30]	Annulus [11-20]	Inner Capsule	Annulus	Cladding	B-19
I-11-0	0.0518 cc	1557°R	1351°R	5.55 atm	2.56 atm	2.99 atm	1.56 atm
I-11-1	0.2044 cc	1567°R	1353°R	13.27 atm	2.71 atm	10.56 atm	1.76 atm
I-11-2	0.6137 cc	1380°R	1277°R	32.7 atm	2.4 atm	30.3 atm	1.4 atm
I-11-3	0.9551 cc	2030°R	1512°R	66.21 atm	2.86 atm	63.35 atm	1.86 atm

TABLE VI

CALCULATION OF PRESSURE AND THERMAL STRESSES

Experiment Number	Pressure Differences		Temperature Difference		Pressure Stress		Thermal Stress	
	Cladding	Capsule	Cladding	Capsule	Cladding	Capsule	Cladding	Capsule
I-11-0	2.99 atm	1.56 atm	13°	2°	77 psi	162 psi	2176 psi(C) 1679 psi(T)	336 psi(C) 312 psi(T)
I-11-1	10.56 atm	1.71 atm	27°	4°	271 psi	177 psi	4519 psi(C) 3488 psi(T)	1346 psi(C) 1248 psi(T)
I-11-2 ^(a)	30.3 atm	1.4 atm	4°	19°	775 psi	145 psi	669 psi(C) 517 psi(T)	3196 psi(C) 2964 psi(T)
I-11-3	63.35 atm	1.86 atm	79°	22°	1623 psi	193 psi	13221 psi(C) 10205 psi(T)	3701 psi(C) 3432 psi(T)

(a) Apparent inconsistencies probably due to the fact that a different HEATING Model was used.

These stresses are comparable to the 12k, and the temperature is somewhat above the 1000°F quoted above (about 1350°F). The irradiation time, however, is only about 2k hours. A 1% creep might, therefore, be an acceptable estimate of the creep. Hence, a 1% increase in circumference would result in about 1/3 of that as an increase in radius, i.e., about 0.001" diametral increase (K, 1d, and 2e).

3. Properties of Cladding Material

The parameters used in these calculations were obtained primarily from the Reactor Handbook, Vol. I, Materials. The values quoted here are the result of correcting those taken from the Handbook to place them all in consistent units.

Material of cladding and capsule - 304 SST

Density = 0.2861 lb/in.³

Thermal capacity = 0.12 Btu/hr/°F

Thermal conductivity = 1.001 Btu/hr/°F/in.

Coefficient of Linear Thermal Expansion (at 1200°F)

= $10.9 \times 10^{-6} (\text{°F})^{-1}$

Modulus of elasticity (at 1200°F) = 18.5×10^6 psi

Poisson's ratio = 0.32

Peak gamma heating rate = 1329.3 Btu/hr/in.³

(from Irradiation Guide)

(K, 1e, and 2j)

L. Effect of Diameter Changes on Coolant Flow

The effect of a diameter increase of 0.001" calculated above is to reduce the coolant flow rate. Since the flow will be reduced to adjust

the bulk temperature rise anyway, the expected reduction in flow rate is not considered to be excessive.

IV. SAFETY ANALYSIS

A. Normal Operation

1. Fuel pellets are individually encapsulated in groups of 4 per capsule, and capsule string is held in place by a spring at the upper end of the B-19 pin. Little or no movement is, therefore, to be expected.
2. The fuel expansion due to thermal and irradiation effects was shown in the original data package to be less than the He gap. Hence, no adverse effects are expected from fuel clad interactions.
3. Sodium is not used as a thermal bonding agent.
4. Estimated swelling of fuel cladding and capsule was shown not to be excessive.
5. Temperatures are all well below the melting points of both fuel and cladding. According to data on creep strengths given in the Reactor Handbook, Vol. I, Materials, p. 565 the ductility may be reduced by about 50% at the highest temperature in the 30% sample. This is not expected to result in a hazard, however, since the stresses are still below the reduced creep strength.
6. Flux foil packages have been placed at lower temperature points in the sample string. No adverse effects are, therefore, expected. Furthermore, the masses of these foils are very small of the order of tens of milligrams. Even if they were to melt, no harm would result. They all consist of non-fissile absorbers.

B. Abnormal Reactor Operation

1. Over Power Operation

The effect of increased operating power above normal is essentially one of increased heat production. As indicated in J(a) an increase in PuO_2 produces an increase in total heat production of about 37% in the 30% PuO_2 sample as compared with the 100% UO_2 sample, and about 30% as compared with the 10% sample. Hence, the normal operating conditions of the 30% PuO_2 sample reflect over-power conditions for the samples containing lesser amounts of PuO_2 . In calculating the operating conditions for the 30% sample at 1.25 times of normal power it is assumed that both fission heat and gamma heat are increased by 25%. The HEATING code was used as before to calculate sample temperatures. Peak temperatures were found to be as follows:

Peak center line temperature	2714°F
Cladding inside surface temperature	1519°F
Cladding outside surface temperature	1421°F
Local coolant temperature	900°F
Peak capsule inside temperature	915°F
Peak capsule outside temperature	996°F

The corresponding stresses were calculated to be

Hoop stress due to gas pressure	1765 psi
Thermal stresses σ_a	= 16,401 psi (compression)
in cladding σ_b	= 12,659 psi (tension)
Average gas pressure	= 3.17 atm
Differential pressure across B-19	= 2.17 atm
Hoop stress, B-19	= 225 psi
Thermal stresses σ_a	= 4879 psi (compression)
in capsule σ_a	= 4583 psi (tension)

2. Loss of Coolant Pumping Power

Values of flow rate and reactor power from Figures E-2 and E-3 were used to calculate bulk temperature rise and peak cladding and capsule temperatures. The SIMIR code was used for this purpose. Table VII shows the results for the 30% sample. It should be noted that a bulk temperature rise of 200° has been assumed. This would require an initial coolant flow rate of 1.6 gpm. With the measured flow rate mentioned above (16.5 gpm) the initial bulk temperature rise is only 14°F and the temperature transient would be correspondingly less severe. It would also be correspondingly less severe for the 10% PuO_2 and the 100% UO_2 samples. A significant point to note in Table VII is the fact that although the peak temperatures rise above the normal ones the temperature differences across the cladding and capsule drop below the normal temperature differences. This means the thermal stresses will be less. These differences reflect the effect of reducing the magnitude of the heat sources as reactor power drops. An anomalous temperature difference of 45° occurs at time step 18 seconds. At this point the sudden drop in reactor power has produced an unusually small bulk temperature rise. Interior temperatures, however, are still at relatively high values. The differences are thus exaggerated.

The values shown in Table VII are considered to be maximum ones because of the large initial bulk temperature rise resulting from the small coolant flow rate that was assumed.

TABLE VII

TEMPERATURE TRANSIENT ON LOSS OF COOLANT PUMPING POWER

<u>Time</u> <u>(seconds)</u>	<u>Δt_b</u>	<u>Cladding</u>			<u>Capsule</u>		
	<u>$^{\circ}\text{F}$</u>	<u>(R-Z)</u>	<u>$t^{\circ}\text{F}$</u>	<u>$\Delta t^{\circ}\text{F}$</u>	<u>(R-Z)</u>	<u>$t^{\circ}\text{F}$</u>	<u>$\Delta t^{\circ}\text{F}$</u>
0	200 $^{\circ}\text{F}$	(7-30)	1362 $^{\circ}\text{F}$	80 $^{\circ}\text{F}$	(12-30)	879 $^{\circ}\text{F}$	23 $^{\circ}\text{F}$
6	307 $^{\circ}\text{F}$	(7-30)	1369 $^{\circ}\text{F}$	75 $^{\circ}\text{F}$	(12-30)	946 $^{\circ}\text{F}$	14 $^{\circ}\text{F}$
12	471 $^{\circ}\text{F}$	(7-30)	1391 $^{\circ}\text{F}$	64 $^{\circ}\text{F}$	(12-30)	1065 $^{\circ}\text{F}$	10 $^{\circ}\text{F}$
18	123 $^{\circ}\text{F}$	(7-30)	1368 $^{\circ}\text{F}$	62 $^{\circ}\text{F}$	(12-30)	857 $^{\circ}\text{F}$	45 $^{\circ}\text{F}$
24	147 $^{\circ}\text{F}$	(7-30)	1256 $^{\circ}\text{F}$	55 $^{\circ}\text{F}$	(12-30)	812 $^{\circ}\text{F}$	8 $^{\circ}\text{F}$
42	262 $^{\circ}\text{F}$	(7-30)	1026 $^{\circ}\text{F}$	20 $^{\circ}\text{F}$	(12-30)	913 $^{\circ}\text{F}$	10 $^{\circ}\text{F}$
54	413 $^{\circ}\text{F}$	(7-30)	1018 $^{\circ}\text{F}$	5 $^{\circ}\text{F}$	(12-30)	1020 $^{\circ}\text{F}$	4 $^{\circ}\text{F}$
66	344 $^{\circ}\text{F}$	(7-30)	1018 $^{\circ}\text{F}$	7 $^{\circ}\text{F}$	(12-30)	970 $^{\circ}\text{F}$	6 $^{\circ}\text{F}$
90	282 $^{\circ}\text{F}$	(7-30)	967 $^{\circ}\text{F}$	8 $^{\circ}\text{F}$	(12-30)	907 $^{\circ}\text{F}$	4 $^{\circ}\text{F}$
120	211 $^{\circ}\text{F}$	(7-30)	907 $^{\circ}\text{F}$	6 $^{\circ}\text{F}$	(12-30)	860 $^{\circ}\text{F}$	4 $^{\circ}\text{F}$

C. Combined Cladding and Capsule Failure (C2 in the Guide)

- a. Not applicable since adjoining pins are stainless steel dummies.
- b. Fission gas volumes at standard conditions are calculated in Appendix I. Table VIII lists these volumes together with the volumes at the reactor temperature of 900°F.

TABLE VIII

FISSION GAS VOLUMES

<u>Experiment No.</u>	<u>Std. Conditions</u>	<u>Fission Gas Volume</u>
		<u>900°F</u>
I-11-0	0.0518 cc	0.1334 cc = 4.7×10^{-6} ft ³
I-11-1	0.2044 cc	0.526 cc = 1.85×10^{-5} ft ³
I-11-2	0.6137 cc	1.58 cc = 5.58×10^{-5} ft ³
I-11-3	0.9551 cc	2.46 cc = 8.67×10^{-5} ft ³

The volume of the fuel pellets is 0.0589 cc per capsule. If all fuel were released to the primary coolant, the volume released would be $5 \times 0.0589 \times 3.35 \times 10^{-5} = 1.04 \times 10^{-5}$ ft³. For the 30% sample this would mean the release of 8.6 grams of Pu to the primary coolant.

- c. Rupture of both inner capsule and B-19 pin would be required to expose primary coolant to sample fuel. (see above)

D. Criticality Considerations

The ²³⁹Pu and ²³⁸U contents of the 20 pellets are

4.8162 g (²³⁹Pu + ²⁴¹Pu)

21.0156 g ²³⁸U

25.8318 g

The ratio

$$\frac{^{239}\text{Pu}}{^{239}\text{Pu} + ^{238}\text{U}} = \frac{4.8162}{25.8218} = 0.1863$$

Since the densities of PuO_2 and UO_2 are nearly equal (11.46 for PuO_2 and 10.96 for UO_2) the volume ratio will be nearly equal to the weight ratio as calculated above. For this mixture a weight of about 35 kg is required for criticality (Reactor Handbook, 2nd Ed., Vol. III, Part A, Physics, p. 276, Fig. 6.4). The 4.8 g of ^{239}Pu in the proposed sample assembly is well below the critical mass even when the subassembly is flooded. (Critical mass for a typical thermal reactor such as the ARMF-I is about 5 kg.)

The above calculation is for the 20% samples. For the 30% sample the Pu content is increased by 50%. For the other samples it is proportionately less. Criticality is hence no problem with these samples.

E. Reinsertion Criteria

Reinsertion of these experiments is not contemplated.

V. CONCLUSION

The studies described in this report cover the heat transfer, stress analysis, and hazards analysis required for the irradiation of four sample strings in the EBR-II. Each experiment consists of one of the four sample strings described herein located in the center position of a B-19 subassembly. The remaining 18 positions are filled with solid 304 SST rods.

Four more sample strings are also contemplated that will be identical to those described above. These latter are to be irradiated in a B-19 subassembly in which the remaining 18 positions will be filled with B-19 pins containing depleted uranium rods. The results of similar studies on these samples will be reported as a supplement to this report.

VI. APPENDIX I

Fission Gas Pressure

Assumptions: Burn-up fraction = 0.0214

1 atmosphere = 12.4 psia at local altitude

Capsule gas space = 0.0589 cc

Gas fraction of fission products = 0.3

100% gas release

Loading temperature = 68°F

100% UO₂ Sample

Weight UO₂ 6.130 g

U/UO₂ weight fraction = 0.8815

U-238 weight fraction = 0.99575

Weight U-238 = 5.3806 g

U-238 loss by fission and capture = 3.41×10^{-4}
(calculated by SINBAD burn-up code)

Assume all U-238 loss fissioned

$$\begin{aligned}\text{Atoms fissioned} &= \frac{5.3806 \times 3.41 \times 10^{-4} \times 6.02252 \times 10^{23}}{238} \\ &= 4.6429 \times 10^{18}\end{aligned}$$

Gas fraction = 1.3928×10^{18}

$$\begin{aligned}\text{Gas volume (standard conditions)} &= \frac{1.3928 \times 10^{18} \times 22.4 \times 10^3}{6.02252 \times 10^{23}} \\ &= 0.0518 \text{ cc}\end{aligned}$$

10% PuO₂ Sample

Pu-239 weight = 0.3307 g

Pu-241 weight = 0.0090 g

$$\text{Gas atoms (239)} = \frac{0.3307 \times 0.0214 \times 0.3 \times 6.02252 \times 10^{23}}{239}$$

$$= 5.33499 \times 10^{18}$$

$$\text{Gas atoms (241)} = \frac{0.0090 \times 0.0214 \times 0.3 \times 6.02252 \times 10^{23}}{241}$$

$$= 1.4439 \times 10^{17}$$

$$\text{Total gas atoms} = 5.4943 \times 10^{18}$$

$$\text{Gas volume (standard conditions)} = \frac{5.493 \times 10^{18} \times 22.4}{6.02252 \times 10^{23}} = 10^3$$

$$= 0.2044 \text{ cc}$$

20% PuO₂ Sample

$$\text{Fissile material per sample} = 1.02 \text{ g}$$

$$\text{Gas atoms} = \frac{1.02 \times 0.0214 \times 0.2 \times 6.02252 \times 10^{23}}{239.0552}$$

$$= 1.65 \times 10^{19}$$

$$\text{Gas volume} = \frac{1.65 \times 22.4 \times 10^3 \times 10^{19}}{6.02252 \times 10^{23}}$$

$$= 0.6137 \text{ cc}$$

30% PuO₂ Sample

$$\text{Pu-239 weight} = 1.5736 \text{ g}$$

$$\text{Pu-241 weight} = 0.0139 \text{ g}$$

$$\text{Gas atoms (239)} = \frac{1.5736 \times 0.0214 \times 0.3 \times 6.02252 \times 10^{23}}{239}$$

$$= 2.5457 \times 10^{19}$$

$$\text{Gas atoms (241)} = \frac{0.0139 \times 0.0214 \times 0.3 \times 6.02252 \times 10^{23}}{241}$$

$$= 2.2300 \times 10^{17}$$

$$\begin{aligned}\text{Gas volume (standard conditions)} &= \frac{2.568 \times 10^{19} \times 22.4 \times 10^3}{6.02252 \times 10^{23}} \\ &= 0.9551 \text{ cc}\end{aligned}$$

Gas pressures (standard temperatures)

100% UO₂

$$\begin{aligned}\frac{(0.0589 + 0.0518) \times 1}{0.0589} &= 1.88 \text{ atmosphere} \\ &= 23.3 \text{ psia}\end{aligned}$$

10% PuO₂

$$\begin{aligned}\frac{(0.0589 + 0.2044) \times 1}{0.0589} &= 4.47 \text{ atmosphere} \\ &= 55.4 \text{ psia}\end{aligned}$$

20% PuO₂

$$\begin{aligned}\frac{(0.0589 + 0.6137) \times 1}{0.0589} &= 11.42 \text{ atmosphere} \\ &= 141.6 \text{ psia}\end{aligned}$$

30% PuO₂

$$\begin{aligned}\frac{(0.0589 + 0.9551) \times 1}{0.0589} &= 17.22 \text{ atmosphere} \\ &= 213.5 \text{ psia}\end{aligned}$$

VII. APPENDIX II

Thermal Stress Formulas

Ref: Thermal Stress Techniques; Zenons-Zudan; Isi Chu-Yen, Wm. H. Steigelmann. Amer, Elsevier Publ. Co. Inc. New York (1965)
pp. 235, 236.

$$\sigma_a = \frac{\sigma E \Delta T}{2(1-\nu)} \left[\frac{1}{\ln(b/a)} - \frac{2b^2}{b^2 - a^2} \right] \quad \text{Eq. (3.89)}$$

$$\sigma_b = \frac{\sigma E \Delta T}{2(1-\nu)} \left[\frac{1}{\ln(b/a)} - \frac{2a^2}{b^2 - a^2} \right] \quad \text{Eq. (3.90)}$$

VII. APPENDIX III

Figures 3 and 4 show the R-Z models used to calculate internal temperatures. Figure 3 was used with the HEATING code in the case of the 20% sample, and Figure 4 was used with both HEATING and SIMIR codes in the case of the other samples. A comparison of results obtained by the use of the two codes in the case of 30% sample is shown in Table IX.

TABLE IX

COMPARISON OF TEMPERATURES CALCULATED BY HEATING AND SIMIR CODES

Mesh	Points	Cladding Temps.		Peak Temps		Temperature Gradients			
						Cladding		Capsule	
<u>R</u>	<u>Z</u>	<u>HEATING</u>	<u>SIMIR</u>	<u>HEATING</u>	<u>SIMIR</u>	<u>HEATING</u>	<u>SIMIR</u>	<u>HEATING</u>	<u>SIMIR</u>
1	10			2240°F	2257°F				
7 - 10	10					77°F	79°F		
12 - 14	10							23°F	23°F
7	10	1284°F	1263°F						
1	15			2263°F	2279°F				
7 - 10	15					79°F	78°F		
12 - 14	15							22°F	23°F
7		1307°F	1385°F						
1	20			2289°F	2305°F				
7 - 10	20					78°F	79°F		
12 - 14	20							23°F	23°F
7	20	1334°F	1311°F						
1	25			2314°F	2331°F				
7 - 10	25					79°F	79°F		
12 - 14	25							23°F	23°F
7	25	1359°F	1337°F						
1	30			2341°F	2358°F				
7 - 10	30					78°F	80°F		
12 - 14	30							23°F	23°F
7	30	1386°F	1364°F						

Note: Above temperatures calculated for the 30% sample.

Figure 3. R-Z model used with the HEATING heat transfer code.

Figure 4. R-Z model used with the HEATING and the SIMIR heat transfer codes: