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LOS ALAMOS SCIENTIFIC LABORATORY
of the
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LOS ALAMOS • NEW MEXICO

Description of Experiments
LASL-36, -37, -38, -39, -40, -41, and -42
and Evaluation of the Hazards Involved
with Their Irradiation in EBR-II

UNITED STATES
ATOMIC ENERGY COMMISSION
CONTRACT W-7405-ENG. 36

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Description of Experiments
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by

John O. Barner

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IN REPLY
REFER TO:

February 12, 1968

TO: Holders of LA-3768-MS

SUBJECT: ERRATA TO LA-3768-MS

Page 14, first column, table at top of page, thermal conductivity of Hastelloy-X should be

$$5.21 + 6.25 \times 10^{-3} ({}^{\circ}\text{F}_{\text{mean}}).$$

Page 14, second column, Item 14

$$(\alpha_T)_R \text{ should be } (\sigma_T)_R.$$

Page 16 $(\alpha_T)_{R_O}$ should be $(\sigma_T)_{R_O}$

and $(\alpha_T)_{R_i}$ should be $(\sigma_T)_{R_i}$

Page 17, line 4, $T = 112.8^{\circ}\text{F}$.

Page 17, first equation should be

$$\begin{aligned} (\sigma_T)_{R_O} &= \frac{(22.9 \times 10^6)(8.56 \times 10^{-6})(112.8)}{2(1 - 0.32)(0.0695)} \left[1 - \frac{(2)(0.140)^2(0.0695)}{0.0029} \right] \\ &= +14,000 \text{ psi} \end{aligned}$$

Page 17, second equation should be

$$\begin{aligned} (\sigma_T)_{R_i} &= \frac{(22.9 \times 10^6)(8.56 \times 10^{-6})(112.8)}{2(1 - 0.32)(0.0695)} \left[1 - \frac{(2)(0.150)^2(0.0695)}{0.0029} \right] \\ &= -18,240 \text{ psi} \end{aligned}$$

TABLE OF CONTENTS

	<u>PAGE</u>
INTRODUCTION	5
EXPERIMENT DESCRIPTION AND HAZARDS	5
A. Purpose and Objectives of the Experiments	5
B. Description of the Fuel and Cladding; Summary of Prior Irradiation Experience	6
C. Fuel Pin and Capsule Data	8
D. Other Materials Data	8
E. Drawings of Fuel Elements, Capsules, and Other Devices	8
F. Fabrication and Testing of Capsules Prior to Irradiation	8
G. Subassembly Data	9
H. Nuclear Effects of the Subassembly in EBR-II	10
I. Performance of the Subassembly During Irradiation	10
J. Postirradiation Characteristics of Subassembly and Capsules	18
K. Handling and Shipment of Capsules	18
L. Hazards Evaluation	18
REFERENCES	21
APPENDIX	23

NOMENCLATURE

Pin, clad	test element.
Capsule	ANL-supplied outer container.
MWD/T	measure of fuel burnup, megawatt days per tonne.
Grams fissioned/cm ³	measure of burnup.
P	power, kW/cm or kW/ft.
F	fissions/sec/g.
M	grams of fissionable isotope/cm.
M'	weight of structural material, g/cm.
H' _Y	gamma heating, W/g.
P _e	Peclet's number.
Nu	Nusselt's number.
C _p	heat capacity, Btu/lb-°F.
D _e	equivalent flow diameter, ft.
ρ	density, g/cm ³ .
K	thermal conductivity, Btu/sec-°F-ft.
h	film coefficient, Btu/sec-°F-ft ² .
T	temperature, °F.
Q	heat flux, Btu/h-ft.
l	length, ft.
R _o	outside radius, ft.
R _i	inside radius, ft.
W	internal heat generation rate, Btu/h-ft ³ .
α	thermal expansion, in./in.-°F.

DESCRIPTION OF EXPERIMENTS LASL-36, -37, -38, -39, -40, -41, AND -42
AND EVALUATION OF THE HAZARDS INVOLVED WITH THEIR IRRADIATION IN EBR-II

John O. Barner

INTRODUCTION

This report provides information required to evaluate the hazards involved in the irradiation in EBR-II of seven capsules designated LASL-36, -37, -38, -39, -40, -41, and -42. These capsules are fueled with single-phase (U,Pu)C, clad in either Type 316 stainless steel or Hastelloy-X with a sodium bond between fuel and clad.

The section letters and numbers in this report correspond to the outline for the preparation of hazards reports as contained in "Guide for Irradiation Experiments in Experimental Breeder Reactor II (EBR-II), Rev. 2."¹ References 2-7 are also used in conjunction with Ref. 1. Evaluation of the subassembly which will contain the Los Alamos Scientific Laboratory (LASL) capsules will be performed by Argonne National Laboratory (ANL) after the final loading has been determined. Sections of this report which deal solely with the subassembly are so noted.

EXPERIMENT DESCRIPTION AND HAZARDS

A. Purpose and Objectives of the Experiments

1. The objectives of the LASL fast breeder experimental program are:

- a. To determine the chemical stability and mechanisms of mass transfer in the fuel element systems, single-phase uranium-plutonium carbide (also mixed nitrides), sodium bond, and high-strength alloy claddings.
- b. To contribute sufficient design information to the Liquid Metal Fast Breeder Reactor (LMFBR) program, from irradiation experiments utilizing prospective fuel element systems for design of a retentive

mixed carbide fuel element power reactor system.

The objectives of the former part of the program will be realized by out-of-pile testing. The objectives of the latter part will be completed through a continuing irradiation program utilizing fuel element mockups which will be irradiated primarily in a fast flux. Heating rates will be approximately 30 kW/ft to burnups up to 1 g fissioned/cm³ (~ 91,000 MWD/T).

The seven test assemblies which comprise the initial set of experiments are designed to operate in EBR-II at the conditions listed in Table I.

2. The following tests are planned for postirradiation evaluation of the seven assemblies:

- a. Capsule, element, and fuel dimensional and warpage measurements.
- b. Fission gas sampling and analysis from capsule annulus (if any) and fuel pin.
- c. Fuel burnup determination by mass spectrographic analysis for stable neodymium and molybdenum isotopes, radiochemistry, and uranium and plutonium depletion.
- d. Fission product concentrations in the pin sodium bond.
- e. Metallographic examination of the fuel and cladding.
- f. Electron microscopic examination of the fuel and cladding.
- g. Gamma scan of the capsule and pin.
- h. Beta-gamma and alpha radiography of the fuel and cladding.
- i. Fuel density determinations.
- j. Microprobe analysis of clad and fuel, if possible.

Table I
General Design Operating Conditions of LASL, EBR-II Test Capsules

Operating Conditions	Capsules						
	LASL-36	LASL-37	LASL-38	LASL-39	LASL-40	LASL-41	LASL-42
Nominal Maximum Power (kW/ft)	30	30	30	30	30	30	30
Maximum Burnup (g fissile fissioned/cm ³)	0.22	0.44	0.66	0.22	0.44	0.66	0.44
Maximum Burnup, MWD/T (fissile)	20,000	40,000	60,000	20,000	40,000	60,000	40,000
Required MWD	3,555	7,110	10,665	3,555	7,110	10,665	7,110
Est. Time in Reactor at 50% Efficiency (months)	5.2	10.4	15.6	5.2	10.4	15.6	10.4
Maximum Capsule Surface Temp (°F)	~ 1020	~ 1020	~ 1020	~ 1020	~ 1020	~ 1020	~ 1020
Reactor Position (Sector E)	4n2-2	4n2-4	4n2-5	4n2-3	4n2-6	4n2-8	4n2-1
Reactor Power (MWt)	45	45	45	45	45	45	45

B. Description of the Fuel and Cladding; Summary of Prior Irradiation Experience

1. A description of the fuel and cladding to be irradiated is as follows:

Fuel:

Composition	(U _{0.8} Pu _{0.2})C, single phase	
Thermal conductivity, cal/sec/cm/°F	0.035	
Btu/h/ft/°F	8.55	
Thermal expansion, in./in./°F	6.15 x 10 ⁻⁶	
Melting point (solidus), °F	4442	
Theoretical density, g/cm ³	13.6	
Crystal structure	Face-centered cubic	
Phase transformations	None	

Clad:

	316 SS	Hastelloy-X
Melting point, °F	~ 2550	2300-2420
0.2% yield strength @ 1200°F, psi	25,000	40,000
Tensile strength @ 1200°F, psi	55,000	84,000
Stress rupture, 10,000 h @ 1200°F, psi	19,000	24,000

2. Summary of Irradiation Data

Irradiation experiments on UC and (U,Pu)C fuels have been completed in thermal reactors during the past few years.^{8,9} Several more experi-

ments on mixed carbide in fast and thermal reactors are either in test or planned by United Nuclear Corporation (UNC), ANL, and Westinghouse. The great majority of these tests use nonstoichiometric carbides which are complex multiphase systems. The choice of single-phase mixed carbide for the proposed LASL experiments is based upon observations of increasing stability as carbides containing free metal approach the stoichiometric composition.⁹ Also, in a sodium-bonded element containing higher order carbides (especially the dicarbide), carbon would undoubtedly be transferred to the cladding with a deleterious effect.

Irradiation tests on UC and mixed carbide fuels indicate that a diametral increase of approximately 0.7% per 10,000 MWD/T might be expected. This diametral increase can be accomplished by the displacement of a sodium bond; hence, excessive strains of embrittled claddings will not occur to cause cladding rupture. Sodium bonding will also reduce fuel operating temperatures. This temperature reduction will reduce fuel swelling and improve gaseous fission product retention.¹⁰ Gaseous fission product release values for sintered mixed carbides irradiated in a thermal reactor range up to 30% at > 100,000 MWD/T.⁸ Release values at ~ 40,000 MWD/T are ~ 1%.

The high lineal heating rate proposed for the

seven experiments is near the maximum power rating that is being considered for nonvented fuel pins.¹¹ Relatively large thermal stresses and high temperatures are developed in the stainless steel test cladding. High-strength materials which are compatible with sodium and (U,Pu)C are required for containment. The production of hydrogen and helium by (n,p) and (n, α) reactions will undoubtedly embrittle nonrefractory metal claddings. Therefore, designs of fuel pins must not incorporate highly strained claddings. In addition, the stress-rupture strength of the claddings must be sufficiently high to provide for adequate creep strength after long periods of operation under thermal and hoop stresses (caused by released gaseous fission products). The relative decrease in ductility is dependent upon the amount of helium and hydrogen that is allowed to collect in grain boundaries.¹² The use of precipitation-hardened or age-hardened (Hastelloy-X) alloys may prevent the accumulation of gases in the grain boundaries by providing precipitation points in the grains of the alloy.¹³

Irradiation testing of cladding materials has been completed in thermal reactors by several investigators. While the data may not be directly extrapolatable to fast reactors,^{12,14,15} some results can be observed. In general the 0.2% yield strength and ultimate tensile strength at high temperature remain approximately the same. There is generally a large decrease in elongation to failure and normally a decrease in stress-rupture strength. No significant irradiation results from a fast reactor to high exposures (i.e., $\sim 10^{23}$ nvt) have been reported or completed for the cladding materials of interest. However, similar results would be expected. Table II illustrates some of the typical results from thermal reactors on cladding materials.

In addition to the above general results, irradiation experiments in EBR-II are either underway or are planned for Type 316 stainless steel and Hastelloy-X. The results of these experiments will be of significant value for determining appropriate claddings for future irradiations.

Table II
Irradiation Damage to Clad Materials

Material	Irradiation Temperature (°F)	Dose	Yield Strength @ °F	Tensile Strength @ °F	Elongation @ °F	Reference
316 SS	----		13K @ 1290	36 @ 1290	48% @ 1290	14
	1200	$2.5 \times 10^{19} > 1 \text{ MeV}$	25K @ 1290	67 @ 1290	46% @ 1290	
Hastelloy-X	----		42K @ 1100	----	45% @ 1100	15
	700-800	$4 \times 10^{20} > 1 \text{ MeV}$	41K @ 1100	----	13% @ 1100

Fig. 1. LASL, EBR-II irradiation capsule.

C. Fuel Pin and Capsule Data

Component	LASL 36, 37, 38	LASL 39, 40, 41	LASL 42
1. Chemical composition, fuel	Single phase ($U_{0.8}Pu_{0.2}$)C		
2. Uranium enrichment, %	93	93	93
3. Fuel pellet dimensions, in.			
Diameter	0.265	0.265	0.265
Height	0.250	0.250	0.250
Column height	13.250	13.250	13.250
4. Total fuel weight, g	146.2	146.2	138.5
5. Fuel volume, cm^3	11.98	11.98	11.98
6. Fuel density, g/cm^3	12.23 ± 0.1	12.23 ± 0.1	11.27 ± 0.1
7. Blanket composition (depleted)	UC	UC	UC
8. Blanket pellet dimensions, in.			
Diameter	0.265	0.265	0.265
Height	0.250	0.250	0.250
Stack height			
Lower	1.000	1.000	1.000
Upper	0.250	0.250	0.250
9. Blanket weights, g	15.5	15.5	15.5
10. Blanket volume, cm^3	1.13	1.13	1.13
11. Blanket density, g/cm^3	13.0	13.0	13.0
12. Clad material	316 SS	Hastelloy-X	316 SS
13. Clad, o.d., in.	0.300	0.300	0.300
14. Clad wall thickness, in.	0.010	0.010	0.010
15. Radial sodium bond annulus, in.	0.0075 ± 0.0025	0.0075 ± 0.0025	0.0075 ± 0.0025
16. Sodium level (in.) above upper blanket in pin at operating temperature	5.25	5.25	5.25
17. Gas in pin	He	He	He
18. Gas volume and pressure in pin at room temperature, and at operating temperature, cm^3 , psia	18.27, ~ 10 16.9, ~ 33	18.27, ~ 10 16.9, ~ 33	18.27, ~ 10 16.9, ~ 33
19. Pin sodium weight, g	5.63	5.63	5.63
20. Capsule sodium level	Same as pin sodium level		
21. Gas in capsule annulus	He	He	He
22. Gas volume and pressure in capsule annulus at operating temperature, cm^3 , psia	5.26, ~ 32	5.26, ~ 32	5.26, ~ 32

D. Other Materials Data

Not applicable.

E. Drawings of Fuel Elements, Capsules, and Other Devices

A drawing of a typical LASL capsule is illustrated in Fig. 1. The LASL drawing number corresponding to this figure is 58Y-119163.

F. Fabrication and Testing of Capsules Prior to Irradiation

1. Fabrication

a. General

The experimental pins and capsules are fabricated by tungsten inert gas welding procedures. Care is taken to ensure that no radioactive contamination is present on the weld areas and the outer surfaces.

b. Filling of the Capsule Annuli with Sodium

Both the fuel-clad and clad-capsule annuli

are filled with sodium to a level of 5.25 in. (at operating temperature) above the upper blanket. The space above the sodium, in both annuli, is filled with helium.

A calculated weight of sodium is extruded in a helium atmosphere containing less than 10 ppm oxygen plus water. The sodium source is purified, and the final billet contains less than 5 ppm oxygen. For the pin annulus, the sodium extrusion is inserted into the gas space above the fuel, and the pin upper cap is welded in place under a partial atmosphere of helium. For the clad-capsule annulus, the sodium extrusion is inserted into the capsule and melted. The pin is then inserted and the capsule nose plug is welded in place under a partial atmosphere of helium. The bond is then heated to 400° to 550°C for less than 1 h and concurrently vibrated by a magnetostrictive device. The capsule is removed from the heat source so that the sodium freezes in an upward direction.

2. Testing

The following tests are done on the pins and capsules:

- a. Radiography to determine fuel position, sodium level, voids in the sodium bond, and weld integrity.
- b. Eddy-current and ultrasonic tests to determine clad defects, voids in the sodium bond, and shrinkage voids.
- c. Helium leak checks to determine weld integrity.
- d. Visual inspection.
- e. Measurements of fuel, cladding, and capsule to determine diametral and axial dimensions and warpage.
- f. Chemical analyses of the sodium.

3. Detailed Inspection Procedures

a. Eddy-Current and Ultrasonic Tests

The ultrasonic and eddy-current inspections are being used to determine the following conditions:

- (1) Nonbonds between the capsule wall

and the sodium

- (2) Nonbonds between the sodium and the fuel pellets.

- (3) Voids in the sodium.

The first two defects are nonwetting phenomena, while the third is a gross defect affecting the heat transfer.

The ultrasonic inspection is used to detect nonbonds. The inspection system consists of a Sperry Model 721 or 800 Reflectoscope using a high resolution pulser-receiver and a 20-MHz lithium sulfate transducer. Experience with this system has shown that nonbonds of about 0.075-in. diam should be detectable. These defects should not cause any serious heat transfer problems and, if present, should disappear during the first few hours of operation.

The eddy-current inspection is used primarily to detect voids in the sodium. Two axially-positioned eddy-current coils are placed near the rotating capsule. The coils are connected differentially so that a zero output signal is received when they sample areas of equal conductance. The conductivities of the capsule materials indicate that a frequency of about 250 Hz should be sensitive to defects in the sodium bonds. This system was tested utilizing a brass tube with various sized holes drilled in it. This tube was inserted in a 0.010-in.-thick stainless steel tube. The test indicated that a 0.075-in.-diam hole could be detected, while a 0.050-in.-diam hole could not. The sensitivity, therefore, lies between these two values.

G. Subassembly Data

The following information pertains only to the seven LASL capsules that will be included in the subassembly. It is LASL's intention that all seven capsules be included in a single subassembly, and that the low burnup capsules be removed at the proper time by utilizing the planned EBR-II re-insertion equipment.

1. Capsule Materials and Weights

Materials	Capsule Weights (g)		
	LASL 36,37,48	LASL 39,40,41	LASL 42
(U _{0.8} Pu _{0.2})C	146.5	146.5	138.5
UC	15.5	15.5	15.5
304 SS	146.3	146.3	146.3
316 SS	44.2	--	44.2
Hastelloy-X	--	45.5	--
Nb	1.5	1.5	1.5
Pin sodium	5.63	5.63	5.63
Annular sodium	2.54	2.54	2.54

2. Desired Reactor Position

Based on a reactor power of 45 MWt, a 10% increase in the corrected fission rates, and 3.55×10^{10} fissions/W-sec, the desired capsule positions correspond to a Sector E, row 4n2, subassembly positions 1, 2, 3, 4, 5, 6, and 8 for capsules LASL-42, -36, -39, -37, -38, -40, and -41, respectively. Any other position corresponding to the above positions (i.e., approximately 30 kW/ft maximum) is acceptable.

H. Nuclear Effects of the Subassembly in EBR-II

Calculations for this item will be done by ANL.

I. Performance of the Subassembly During Irradiation

1. Heat Generation Rate of the Fuel-Bearing Capsules

$$P = (2.8 \times 10^{-14})(F)(M) + H_Y \text{ kW/cm, [Ref. 1]}$$

F = fissions/sec-g of fissile material (Table V, Ref. 1, adjusted to 45 MWt and increased by 10% as per latest estimates)¹⁶

$$H_Y = \text{gamma heating} = H'_Y(M')$$

$$H'_Y = \text{gamma heating rate, W/g}$$

$$M' = \text{weight of gamma-heated material, g}$$

The calculation is based upon the region of the capsule from the bottom to the top of the fueled region. Gamma heating and fission heating in the blankets and structural materials above and below this region are negligible. The shape of the gamma heating curve in the region of interest is assumed to be the same as that for the power density. A calculation for LASL-38 (Sector E, 4n2-5) is illustrated. Calculations for the final reactor positions will be made by ANL.

a. Calculation of the average fission rates

Isotope	Fissions/g $\times 10^{13}$			
	62.5 MWt	45 MWt	45 MWt $\times 1.10$	Av, 45 MWt $\times 1.10 \div 1.15$
²³⁵ U	1.070	0.770	0.847	0.736
²³⁸ U	0.698	0.050	0.055	0.048
Pu	1.293	0.930	1.023	0.890

b. Calculation of grams of each fissile isotope per cm

$$\text{cm}^3 \text{ of fuel/cm of length} = \frac{\pi}{4} \frac{(0.265 \text{ in.})^2 (16.39 \text{ cm}^3/\text{in.}^3)}{2.54 \text{ cm/in.}} = 0.356 \text{ cm}^3/\text{cm}$$

$$M_{\text{Pu}} = \frac{(13.6 \text{ g/cm}^3)(0.90)(0.20)(239)(0.356 \text{ cm}^3/\text{cm})}{250.5} = 0.831 \text{ g/cm}$$

$$M_{\text{235U}} = \frac{(13.6)(0.90)(0.80)(0.93)(235)(0.356)}{250.5} = 3.040 \text{ g/cm}$$

$$M_{\text{238U}} = \frac{(13.6)(0.90)(0.80)(0.07)(238)(0.356)}{250.5} = 0.232 \text{ g/cm}$$

c. Calculation of gamma heating for each material

Material	H'_Y, W/g (@ 14.34 cm)	
	@ 45 MWt	Averaged over Length
Heavy metals	4.30	3.75
Capsule and clad	2.60	2.26
Na and C	Negligible	Negligible

d. Calculation of material weights

$$M'_{\text{claddings}}/\text{cm} = \frac{\pi}{4} \frac{(16.39 \text{ cm}^3/\text{in.}^3)(7.9 \text{ g/cm}^3)}{(2.54 \text{ cm/in.})} [(0.375^2 - 0.335^2)\text{in.}^2 + (0.300^2 - 0.280^2)\text{in.}^2]$$

$$= 2.01 \text{ g/cm}$$

e. Power generation

Material	F or H _Y	M or M'	kW/cm
²³⁵ U	0.736×10^{13}	3.040	0.6270
²³⁸ U	0.048×10^{13}	0.232	0.0031
Pu	0.890×10^{13}	0.831	0.2075
Heavy metals	3.75×10^{-3}	4.103	0.0154
Capsule and clad	2.26×10^{-3}	2.010	0.0045
Total			0.857

$$P_{\text{av}} = 0.857 \text{ kW/cm} = 25.85 \text{ kW/ft (av/pin)}$$

2. Maximum Linear Heat Generation

$$P_{\text{max}} = (P_{\text{av}})(1.15) = 0.985 \text{ kW/cm} = 29.75 \text{ kW/ft}$$

3. Maximum Capsule Mean Wall Temperature

a. $\text{Power/pin} = \frac{(25.85 \text{ kW/ft})(13.25 \text{ in.})}{12 \text{ in./ft}} = 28.55 \text{ kW/pin} = 1.62 \times 10^3 \text{ Btu/min/pin}$

b. $\text{Allowable } \Delta T_{\text{subassembly}} = \frac{(45 \text{ MWt})(263^\circ\text{F})}{62.5 \text{ MWt}} + 100^\circ\text{F} = 290^\circ\text{F}$

c. $c_{p_{990^\circ\text{F}}}$ of sodium = 0.3017 Btu/lb/°F

d. $\text{Flow/pin} = \frac{1.62 \times 10^3 \text{ Btu/min}}{(0.3017 \text{ Btu/lb/}^\circ\text{F})(290^\circ\text{F})} = 18.55 \text{ lb/min}$

$$\text{Flow/pin} = \frac{(18.55 \text{ lb/min})(7.48 \text{ gal/ft}^3)}{(0.815 \text{ g/cm}^3)(62.43 \text{ lb cm}^3/\text{g ft}^3)} = 2.73 \text{ gal/min/pin}$$

e. $\text{Flow area} = \frac{\pi(0.474^2 - 0.375^2)}{(4)(144)} = 4.53 \times 10^{-4} \text{ ft}^2$

f. $\text{Coolant velocity} = \frac{(2.73 \text{ gal/min})(0.1337 \text{ ft}^3/\text{gal})}{(60 \text{ sec/min})(4.53 \times 10^{-4} \text{ ft}^2)} = 13.42 \text{ ft/sec}$

g. $P_e = \frac{D_e V_p C_p}{K} \quad [\text{Ref. 1}]$

where

$$D_e = \frac{4 \text{ area}}{\Sigma \text{ perimeters}} = \frac{4\pi(0.474^2 - 0.375^2)(12)}{4\pi(0.474 - 0.375)(144)} = 8.15 \times 10^{-3} \text{ ft}$$

$$\rho = (0.815 \text{ g/cm}^3)(62.43 \text{ lb-cm}^3/\text{g ft}^3) = 50.8 \text{ lb/ft}^3$$

$$c_p = 0.3017 \text{ Btu/lb/}^\circ\text{F}$$

$$K = \frac{(0.157 \text{ cal/sec-}^\circ\text{C-cm})(244 \text{ Btu-sec-}^\circ\text{C-cm/h-}^\circ\text{F-ft-cal})}{(3600 \text{ sec/h})} = 1.06 \times 10^{-2} \text{ Btu/sec-}^\circ\text{F-ft}$$

$$\therefore P_e = \frac{(8.15 \times 10^{-3} \text{ ft})(13.42 \text{ ft/sec})(50.8 \text{ lb/ft}^3)(0.3017 \text{ Btu/lb-}^\circ\text{F})}{(1.06 \times 10^{-2} \text{ Btu/sec-}^\circ\text{F-ft})} = 158$$

$$h. \quad N_u = 2.3 + 0.23 (P_e)^{1/2} = 2.3 + 2.89 = 5.19 \text{ [Ref. 1]}$$

$$i. \quad h = \frac{N_u K}{D_E} = \frac{(5.19)(1.06 \times 10^{-2} \text{ Btu/sec-}^\circ\text{F-ft})}{8.15 \times 10^{-3} \text{ ft}} = 6.75 \text{ Btu/sec-}^\circ\text{F-ft}^2$$

$$j. \quad \Delta T_{\text{film at top of fuel}} = Q/hA = 0.75 Q_{\text{av}}/hA$$

$$= \frac{(1.62 \times 10^3 \text{ Btu/min})(144 \text{ in.}^2)(.75)}{(13.25 \text{ in.})(6.75 \text{ Btu/sec-}^\circ\text{F-ft}^2)(\pi)(0.375 \text{ in.})(60 \text{ sec/min})} = 27.7^\circ\text{F}$$

$$k. \quad \text{A similar calculation for the reactor midplane results in } \Delta T_{\text{film at midplane}} = 41.3^\circ\text{F.}$$

$$l. \quad \text{Assume that the maximum mean wall temperature will occur in the top 1 in. of fuel and will equal } T_{\text{coolant}} + \Delta T_{\text{film}} + 1/2 \Delta T_{\text{wall}}$$

$$\Delta T_{\text{wall}} = \frac{Q}{2\pi k l} \ln \frac{R_o}{R_i} + \frac{W}{4K} (R_o^2 - R_i^2) \ln \frac{R_o}{R_i}$$

where

$$K = \text{thermal conductivity} = 10.6 \text{ Btu/hr-}^\circ\text{F-ft}$$

$$l = \text{length of section} = 1.00 \text{ in.} = 1/12 \text{ ft}$$

$$R_o = \text{outside radius of tube} = 1.565 \times 10^{-2} \text{ ft}$$

$$R_i = \text{inside radius of tube} = 1.398 \times 10^{-2} \text{ ft}$$

$$\ln \frac{R_o}{R_i} = 0.1132$$

$$W = \text{internal heat generation rate in wall}$$

$$= \frac{(2.26 \text{ W/g})(7.9 \text{ g/cm}^3)(16.39 \text{ cm}^3/\text{in.}^3)(3.41 \text{ Btu/h-W})}{5.787 \times 10^{-4} \text{ ft}^3/\text{in.}^3}$$

$$= 1.73 \times 10^6 \text{ Btu/hr-ft}^3$$

$$Q = \text{heat flux through the wall}$$

$$= H_{\text{Pu}} + H_{235} + H_{238} + H_{\text{fuel}} + H_{\text{clad}}$$

$$= (56.92 \text{ Btu/min-kW})(2.54 \text{ cm/in.})(60 \text{ min/h})(0.75)$$

$$[0.6270 + 0.0031 + 0.2075 + 0.0154 + \frac{0.116}{0.0502} (0.0045)] \text{ kW/cm}$$

$$= 5.55 \times 10^3 \text{ Btu/h-ft}$$

$$\therefore \Delta T_{\text{wall}} = \frac{(5.55 \times 10^3)(12)(0.1132)}{(2\pi)(10.6)(1)} + \frac{(1.73 \times 10^6)}{(4)(10.6)} [(1.565 \times 10^{-2})^2 - (1.398 \times 10^{-2})^2]$$

$$- (2)(1.398 \times 10^{-2})^2 (0.1118)]$$

$$= 113.5 + 1.2 = 114.7$$

$$\therefore T_{\text{max mean wall}} = 990 + 27.7 + 57.3 = 1075^\circ\text{F}$$

However, refined computer calculations which consider end losses give a value of 1060.2°F for the maximum mean wall temperature.

4. Maximum Heat Flux at the Capsule-Coolant Interface

The maximum heat flux will occur at the core midplane.

$$P = 29.75 \text{ kW/ft}$$

$$\text{Capsule o.d.} = 0.375 \text{ in.}$$

$$\text{Maximum heat flux} = P/A$$

$$= \frac{(2.975 \times 10^4 \text{ W/ft})(3.41 \text{ Btu/h-W})(12 \text{ in./ft})}{(\pi)(0.375 \text{ in.})}$$

$$= 1.033 \times 10^6 \text{ Btu/h-ft}^2$$

5. Total Heat Generation Rate of the Subassembly

The total heat generation for the subassembly will be computed by ANL. A typical LASL capsule will contribute:

$$\frac{(25.85 \text{ kW/ft})(13.25 \text{ in.})}{12 \text{ in./ft}} = 28.55 \text{ kW/pin}$$

$$= 9.75 \times 10^4 \text{ Btu/h-pin}$$

6. Coolant Flow Rate to Meet Design Criteria

The computed coolant flow rate of sodium at 990°F (i.e., outlet flow rate) of a typical LASL capsule (LASL-38) in a Sector E, 4n2-5 subassembly position is 2.73 gal/min (see I-3-D). If all the capsules in the subassembly require the same flow, the subassembly flow will be

$$(2.73 \text{ gpm/pin})(19 \text{ pins}) = 51.9 \text{ gpm/subassembly.}$$

The heating rates and flow rates for all the LASL capsules would be as follows for the hypothetical loading described in section G-2.

7. Orifice Size

This item will be determined by ANL.

8. Coolant Temperature Rise Through the Shroud Tubes

Two methods were used to calculate capsule temperatures. The first utilized a radial heat transfer mode with no end losses, the second utilized a two-dimensional, steady-state heat transfer code with a fluid node solution.

The first method indicated that with a 290°F coolant temperature rise the maximum mean wall temperature would be exceeded by 16°F (including uncertainties). The second method indicated that the maximum mean wall temperature would be 1105°F ± 1°F (including uncertainties) with a coolant temperature rise of 290°F. The second method results in temperatures which more closely approximate the actual conditions than the first method. However, if the ANL maximum mean wall temperature restriction is based upon a no-end-loss model, then the coolant outlet temperature should be reduced to 975°F and the coolant flow reduced by 5.4%.

9. Mixed Mean Coolant Temperature

This item will be calculated by ANL.

10. Thermal Conductivity of the Capsule Materials

The following thermal conductivity values were used to calculate operating temperatures of the capsule components.

Capsule	Position Sector E	% Heating Rate of LASL-38	Maximum Heating Rate (kW/ft)	Required* gpm of Na990 per Subassembly
LASL-36	4n2-2	101.7	30.20	52.7
LASL-37	4n2-4	101.7	30.20	52.7
LASL-38	4n2-5	100	29.75	51.9
LASL-39	4n2-3	100	29.75	51.9
LASL-40	4n2-6	98.3	29.75	51.0
LASL-41	4n2-8	100	29.75	51.9
LASL-42	4n2-1	98	29.15	50.9

*The coolant flow rates are based on exceeding the 1105°F maximum mean capsule wall temperature by 16°F (including uncertainties) as illustrated in section I-11, if the analysis is based upon a heat transfer model with no end losses. If based upon the computer calculation, the flow rates shown will produce a maximum mean wall temperature of 1105°F ± 1°F.

Material	Thermal Conductivity (Btu/h-ft ² -°F)
304 SS	$7.8 + 4.95 \times 10^{-3} (^{\circ}\text{F}_{\text{mean}})$
316 SS	$8.0 + 4.3 \times 10^{-3} (^{\circ}\text{F}_{\text{mean}})$
Hastelloy-X	$62.5 + 7.5 \times 10^{-2} (^{\circ}\text{F}_{\text{mean}})$
Sodium	$52.7 - 1.54 \times 10^{-3} (^{\circ}\text{F}_{\text{mean}})$
(U,Pu)C	8.55

11. Temperature Distributions in Capsules

Figure 2 illustrates the temperature distribution in a LASL capsule operating at 30 kW/ft

Fig. 2. Typical temperature distribution in LASL capsule operating at 30 kW/ft (max), based upon a no-axial conduction model.

(maximum at the midplane) with a coolant outlet temperature of 990°F. The calculation is based upon a model utilizing no end losses (i.e., only radial heat losses), and is therefore conservative. The computer calculation resulted in a maximum fuel temperature of 2130°F as compared to the illustrated 2230°F and a maximum mean wall temperature (including uncertainties) of 1105°F with a coolant tempera-

ture rise of 290°F and a flow rate of 51.9 gpm/sub-assembly. Uncertainties are shown only for critical areas of the capsule. Table III illustrates a sample uncertainty calculation based upon the no-end-loss model.

12. Predicted Expansion of the Fuel

At operating temperature, the diameter of the fuel at the midplane is expected to increase by 0.0034 in. The corresponding increase in clad diameter is 0.0033 in.

The irradiation-induced diametral expansions of the fuel in LASL-38 and -41 are expected to be less than 4% at 0.66 g fissioned/cm³. The sodium bond provides for a diametral fuel expansion of 5.65%. Expansions in the other capsules are expected to be less than 3%.

Axial irradiation-induced expansion is expected to be less than 4% for all capsules.

13. Gas Pressure in Pin Due to Fission Gas Release

The effect of fission gas release upon internal pin pressure is shown in Fig. 3. The fission gas generation after 365 days of irradiation under the conditions of temperature as outlined in I-11 and fission rates as outlined in I-1 are assumed. Actual fission gas release rates are expected to be less than 50%.

14. Calculated Hoop and Thermal Stresses in Pin Wall

The hoop stress in the pin wall as a function of fission gas release is shown in Fig. 3.

The thermal stresses at the reactor midplane (i.e., highest heat flux) are as follows for the Type 316 stainless steel pin:

$$(\alpha_T)_R = \frac{E \alpha T}{2(1-\nu) \ln R_o/R_i} \left[1 - \frac{2 X^2}{R_o^2 - R_i^2} \ln R_o/R_i \right]$$

where

$$E = 2.8 \times 10^7 \text{ psi}$$

$$\alpha = 10.3 \times 10^{-6} \text{ in./in.}$$

$$T = 107.6^{\circ}\text{F}$$

$$\nu = 0.25$$

$$R_i = 0.140 \text{ in.}$$

$$R_o = 0.150 \text{ in.}$$

$$\ln R_o/R_i = 0.0695$$

$$R_o^2 - R_i^2 = 0.0029$$

Table III
Calculation of Capsule Temperature Uncertainties
Based Upon a No-axial Conduction Model

Thermo-physical Quantity	$\theta_{\Delta T, 290(170)}$		$\theta_{\text{film}, 28(41)}$		$\theta_{3C4, 103(147)}$		$\theta_{Na, 37(50)}$		$\theta_{316, 78(107)}$		$\theta_{Na, 19(25)}$		$\theta_{\text{Fuel}, 715(988)}$		Top Ed to Mean wall	Top Ed ² to Mean wall	Midplane Ed to Centerline	Midplane Ed ² to Centerline
	F	d	F	d	F	d	F	d	F	d	F	d	F	d				
Heating Rate	1.07	20.3 (11.6)	1.07	2.0 (2.9)	1.07	7.2 (10.3)	1.07	2.6 (3.5)	1.07	5.4 (7.5)	1.07	1.3 (1.8)	1.07	50.0 (69.2)	25.9	670.8	106.1	11,257.2
Flow Rate	1.05	14.5 (8.5)													14.5	210.3	8.5	72.3
Flow Profile	1.05	14.5 (8.5)													14.5	210.3	8.5	72.3
Tolerances - Clad					1.05	5.1 (7.4)									2.5	6.2	7.4	54.8
Bond 1							1.05	1.8 (2.5)									2.5	6.3
Pin									1.05	3.9 (5.3)							5.3	28.1
Bond 2											1.05	1.0 (1.3)					1.3	1.7
Fuel													1.05	35.8 (49.4)			49.4	2,440.4
Conductivity - Clad					1.10	10.3 (14.7)									5.1	32.3	14.7	216.1
Bond 1							1.10	3.7 (5.0)									5.0	25.0
Pin									1.10	7.8 (10.7)							10.7	114.5
Bond 2											1.10	1.9 (2.5)					2.5	6.3
Fuel													1.10	71.5 (98.8)			98.8	9,761.4
Heat Transfer Coefficient			1.20	5.6 (8.2)											5.6	31.4	39.6	1,568.2
Power level	1.02	5.8 (3.4)	1.02	0.6 (0.8)	1.02	2.1 (2.9)	1.02	0.7 (1.0)	1.02	1.6 (2.1)	1.02	0.4 (0.5)	1.02	14.3 (19.8)	7.4	54.8	30.5	930.3
Transient Overload	1.05	14.5 (8.5)	1.05	1.4 (2.0)	1.05	5.1 (7.3)	1.05	1.8 (2.5)	1.05	3.9 (5.3)	1.05	1.0 (2.3)	1.05	35.8 (49.4)	18.4	886.1	77.3	5,975.3
																2,102.2		32,530.2

Top of fuel, mean capsule wall (see Section H-3-1 for computer results)

Midplane, fuel centerline

*Numbers in parentheses refer to values at the reactor midplane

$$\sqrt{2,102.2} = 45.8, \text{ Max. Temp.} = 1120.8^{\circ}\text{F}$$

$$\sqrt{32,530.2} = 180.5, \text{ Max. Temp.} = 2440^{\circ}\text{F}$$

Fig. 3. Capsule and pin hoop stresses and pin sodium displacement as a function of fission gas release and resulting pressure after 365 days of irradiation. Example: 60% gas release produces 610 psi of pressure in the pin. The resulting pin hoop stress is 8,100 psi. If the pin now fails, the resulting capsule hoop stress is 3,750 psi. The sodium displacement will be 4.75 in. of sodium from the reservoir over the upper blanket.

For $r = R_o$, $X = R_i$

$$(\alpha_T)_{R_o} = \frac{(2.8 \times 10^7)(10.3 \times 10^{-6})(107.6)}{(2)(1 - 0.25)(0.0695)} \left[1 - \frac{(2)(0.140)^2(0.0695)}{(0.150^2 - 0.140^2)} \right]$$

$$= +18,400 \text{ psi}$$

For $R = R_i$, $X = R_o$

$$(\alpha_T)_{R_i} = \frac{(2.8 \times 10^7)(10.3 \times 10^{-6})(107.6)}{(2)(1 - 0.25)(0.0695)} \left[1 - \frac{(2)(0.150)^2(0.0695)}{0.150^2 - 0.140^2} \right]$$

$$= -23,800 \text{ psi}$$

The 0.2% yield strength of Type 316 stainless steel at 1150°F is ~ 30,000 psi.

For Hastelloy-X the thermal stresses are as follows:

$$\begin{aligned}
 E &= 22.9 \times 10^6 \text{ psi} \\
 \alpha &= 8.56 \times 10^{-6} \text{ in./in.} \\
 T &= 9.4^\circ\text{F} \\
 \mu &= 0.32 \\
 R_i &= 0.140 \text{ in.} \\
 R_o &= 0.150 \text{ in.} \\
 \ln R_o/R_i &= 0.0695 \\
 R_o^2 - R_i^2 &= 0.0029
 \end{aligned}$$

For $r = R_o$, $X = R_i$

$$\begin{aligned}
 (\alpha_T)_{R_o} &= \frac{(22.9 \times 10^6)(8.56 \times 10^{-6})(9.4)}{2(1 - 0.32)(0.0695)} \left[1 - \frac{(2)(0.140)^2(0.0695)}{0.0029} \right] \\
 &= +1170 \text{ psi}
 \end{aligned}$$

For $r = R_i$, $X = R_o$

$$\begin{aligned}
 (\alpha_T)_{R_i} &= \frac{(22.9 \times 10^6)(8.56 \times 10^{-6})(9.4)}{2(1 - 0.32)(0.0695)} \left[1 - \frac{(2)(0.150)^2(0.0695)}{0.0029} \right] \\
 &= -1520 \text{ psi}
 \end{aligned}$$

The 0.2% yield strength for Hastelloy-X at 1150°F is ~ 41,000 psi.

15. Effect of Irradiation on Cladding Mechanical Properties

Very little data from fast flux exposures are available for Type 316 stainless steel. However, the yield strength and ultimate tensile strength would be expected to remain approximately the same. A severe loss in ductility to approximately 1% would be expected. The stress rupture of Type 316

stainless steel would be expected to decrease a moderate amount. Similar effects would be expected in Hastelloy-X.

16. Calculated Hoop and Thermal Stresses in the Capsule Wall

The hoop stress in the capsule wall as a result of pin failure is illustrated in Fig. 3.

The thermal stresses at the reactor midplane are as follows:

$$(\sigma_t)_R = \frac{E \alpha T}{2(1 - \mu) \ln R_o/R_i} \left[1 - \frac{2 X^2}{R_o^2 - R_i^2} \ln R_o/R_i \right]$$

where

$$\begin{aligned}
 E &= 2.8 \times 10^7 \text{ psi} \\
 \alpha &= 10.6 \times 10^{-6} \text{ in./in.} \\
 T &= 148.9^\circ\text{F} \\
 \mu &= 0.25 \\
 R_i &= 0.168 \\
 R_o &= 0.188 \\
 \ln R_o/R_i &= 0.1132 \\
 R_o^2 - R_i^2 &= 0.0071
 \end{aligned}$$

For $r = R_o, X = R_i$

$$(\sigma_t)_{R_o} = \frac{(2.8 \times 10^7)(10.6 \times 10^{-6})(148.9)}{(2)(1 - 0.25)(0.1132)} \left[1 - \frac{(2)(0.0283)(0.1132)}{0.0071} \right]$$

$$= +25,000 \text{ psi}$$

For $r = R_i, X = R_o$

$$(\sigma_t)_{R_i} = \frac{(2.8 \times 10^7)(10.6 \times 10^{-6})(148.9)}{(2)(1 - 0.25)(0.1132)} \left[1 - \frac{(2)(0.0354)(0.1132)}{0.0071} \right]$$

$$= -29,400 \text{ psi}$$

The 0.2% yield strength for Type 304 stainless steel at 950°F equals ~ 35,000 psi.

17. Effect of Irradiation on Capsule Mechanical Properties

Very little data are available from fast flux irradiations. The expected results would be similar to Type 316 stainless steel (see I-15).

J. Postirradiation Characteristics of Subassembly and Capsules

1. Decay Heat Production

The target burnups for the LASL capsules are 20,000 MWD/T for LASL-36 and -39, 40,000 MWD/T for LASL-37, -40, and -42, and 60,000 MWD/T for LASL-38 and -41.

There is a total of 10.4 g of uranium and plutonium per in. of fuel.

The required days of irradiation per 10,000 MWD/T for pins operating at 30.20 kW/ft and 90% of theoretical density are:

$$\text{Days} = \frac{(10,000 \text{ MWD/T})(10^3 \text{ kW/MW})(10.4 \text{ g/in.})(12 \text{ in./ft})}{(30.2 \text{ kW/ft})(10^6 \text{ g/T})}$$

$$= 41.32 \text{ days of irradiation/10,000 MWD/T.}$$

The decay heat after completion of the final cycle of irradiation is illustrated in Table IV for LASL-36, -37, and -38. The calculation is based upon the total irradiation time plus the decay time after the final cycle. Decay time between prior cycles is neglected. Values are determined from Fig. 20, Ref. 1. LASL-39, -40, -41, and -42 will have slightly lower decay heats because of slightly lower fission rates than LASL-36, -37, -38, and -37, respectively.

2. Cooling Period in EBR-II Storage Basket

The cooling period in the storage basket should be of sufficient duration to provide a maximum capsule wall temperature of 350°F during transfer and disassembly.

3. & 4. Estimated Gamma Radiation Levels of Subassembly

Items 3 and 4 will be provided by ANL.

5. Isotopic Production

The total isotopic production of the LASL capsules is described in Table V.

K. Handling and Shipment of Capsules

Figure 4 illustrates the shipping container in which the seven capsules will be delivered to ANL. The 2R container shown in the figure will be packed in an ICC-approved wooden shipping crate. Each

capsule is contained in an individual hole in the shock-absorbing liner. The LASL drawing number for the container is 58Y-119163 D1. The postirradiation shipping cask is currently being designed. When completed it will be an ICC-approved shipping container.

L. Hazards Evaluation

1. Effects of Fuel Redistribution

(U,Pu)C will not slump or experience central

Table IV

Decay Heat After Irradiation of Three LASL Capsules

(a)

Capsule LASL-36 (26.2 kW/ft (av) 82.5 days of irradiation)

Time After Final Shutdown	P/P_0 Fraction of Operating Power	Decay Heat (kW)
1 min	0.0220	0.635
1 h	0.0090	0.260
12 h	0.0046	0.133
24 h	0.0035	0.101
5 days	0.0019	0.055
20 days	0.0008	0.023
40 days	0.0003	0.001

(b)

Capsule LASL-37 (26.2 kW/ft (av) 165 days of irradiation)

1 min	0.0240	0.695
1 h	0.0096	0.278
12 h	0.0048	0.139
5 days	0.0021	0.061
20 days	0.0009	0.026
40 days	0.0006	0.017

(c)

Capsule LASL-38 (25.9 kW/ft (av) 247.5 days of irradiation)

1 min	0.0245	0.700
1 h	0.0098	0.280
12 h	0.0490	0.140
5 days	0.0040	0.114
20 days	0.0011	0.031
40 days	0.0070	0.020

melting under the design operating conditions.

Fuel cracking will not result in any fuel redistribution.

2. & 3. Effects of Subassembly

Items 2 and 3 pertain to the whole subassembly and will be provided by ANL.

4. Effect of Sodium Bond Discontinuities

There are no sodium bond voids greater than 0.075 in. in diameter in any of the sodium bonds of the seven capsules. Voids smaller than this should have little effect on capsule operation.

5. Nuclear Effect of Flooding the Subassembly

Item 5 will be provided by ANL.

6. Effects of Fuel Clad Failure

Figure 3 illustrates the capsule hoop stress in the event of fuel clad failure because of fission gas release after 365 days of full power irradiation. This irradiation period corresponds to a period of 1.475 times the maximum design period (i.e., 88,500 MWD/T). Even at this level of fission gas generation, the maximum hoop stress in the capsule wall is only about 6000 psi (100% fission gas release assumed). The fission gas release is expected to be < 50%. Therefore, the capsule hoop stress will be ~ 3000 psi. The capsule is fully capable of withstanding even 100% fission gas release.

7. Probability and Consequence of Fuel Contacting Capsule Wall as a Result of Clad Failure

(U,Pu)C does not disintegrate in sodium. In addition, the available clad expansion space and resulting hole in the event of clad failure would be small. Therefore, the probability of fuel contacting the clad wall is very slight. However, in the event of clad failure, if the fuel did come in contact with the capsule wall, a slight carburization of the wall might occur. This carburization would be expected to be very slight because the fuel

Fig. 4. Preirradiation shipping container.

is single phase and extremely close to the stoichiometric composition.

8. Effect of Leak at Bottom of Capsule

The pressure in the capsule annulus as a result of the temperature increase is about 32 psia. In the event of a leak at the bottom of a capsule, a portion of the sodium in the annulus may be expected to be forced out. If the sodium is forced to a level below the top of the fuel, melting of clad and fuel will occur in the exposed area. As a result, fission gas leakage and pin sodium displacement will occur and will probably result in reactor coolant contamination.

The occurrence of a leak at the bottom of the capsule is unlikely in view of the temperature distribution along the length of the capsule and the provisions for maintaining the clad-fuel sodium bond in the event of clad failure.

9. Materials Effects on Reactor as a Result of Capsule Failure

Particles of fuel and fission products may be expected to be redistributed to various areas in the reactor as a result of total capsule failure. The magnitude is difficult to predict because it depends upon the extent to which the capsule wall is damaged as a result of clad and fuel meltdown. Failure of both the clad and the capsule must occur before the reactor coolant becomes contaminated.

10. Consequences of Capsule Failure

a. Coolant and cover gas contamination

The isotopes present at the end of irradiation in LASL-38 (~ 60,000 MWD/T) are presented in the appendix. Table V illustrates the curie level of the elements present in LASL-38 as a function of decay time.

b. Chemical effects on reactor coolant

In the event of a failure of the capsule and subsequent failure of the pin clad, fuel will be exposed to the reactor coolant. There is a slight chance that the single-phase (U,Pu)C will carburize the stainless steel in the area where it is finally deposited. However, the amount of carburization should be extremely low in relation to the large mass of reactor

stainless steel, the small mass of fuel, and the stoichiometry of the fuel.

No other capsule components would cause a chemical reaction with reactor components.

c. Capsule to capsule failure propagation

Each capsule is contained within its own shroud tube. Therefore, the probability of one capsule failure causing failure of adjacent capsules by mechanical or chemical mechanisms or by fuel redistribution in sufficient amounts to cause melting is remote.

d. Subassembly

Item d will be evaluated by ANL after the subassembly loading has been assigned.

11. Consequence of IBC Cleaning of Defective Capsules

The cleaning of a failed or defective capsule in the inter-building coffin (IBC) will probably result in radioactive contamination of the IBC and partial oxidation of the sodium bonds and mixed carbide fuel.

12. & 13. Subassembly

Items 12 and 13 relate to the subassembly and will be evaluated by ANL after the loading has been determined.

14. Failure of Pin Cladding

In the event of pin cladding failure (without capsule failure), the amount of sodium in the two bonds has been adjusted so that the cladding-fuel bond will remain intact. This condition will apply for either a failure in the gas annulus or the bottom of the pin. Figure 3 illustrates the primary sodium displacement as the result of failure at the bottom of the pin.

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Table V
Fission Products in the Fuel Pin*
(See Appendix)

Group of Isotopes	Curies @ time						
	0	1 h	12 h	1 d	7 d	14 d	30 d
Xenon	8,947	3,664	2,889	2,216	807	311	38.5
Krypton	5,882	1,482	112	13	3.2	3.2	3.2
Total gas	14,829	5,146	3,001	2,229	810	314	42
Iodine	9,498	6,318	3,302	2,476	656	274	74
Bromine	2,592	173	3.8				
Total halogen	12,090	6,491	3,306	2,476	656	274	74
Sr	6,414	3,184	1,425	999	669	609	493
Y	6,928	4,015	1,475	541	24	24	24
Zr	2,385	2,325	1,798	1,439	824	763	643
Nb	6,922	3,693	2,666	1,946	809	805	777
Ru	767	693	522	488	438	391	302
Rh	838	767	695	647	449	391	303
Sn	490	78	8	6	3	2	1.3
Te	6,325	3,083	1,264	1,115	363	150	69
Cs	5,522	812	99	97	77	60	40
Ba	7,053	2,627	1,269	1,233	900	627	280
La	5,860	3,789	1,433	1,252	993	692	295
Ce	4,846	3,078	2,717	2,441	1,438	1,264	1,026
Pr	3,546	2,858	1,931	1,731	1,433	1,154	774
Nd	893	693	483	465	319	205	75
Pm	486	484	425	367	125	94	90
Eu	6.5	6.4	5.7	4.4	3.8	3.3	1.8
Total long-lived metals	59,282	32,185	18,216	14,771	8,868	7,234	5,194

*See appendix for isotopic breakdown.

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APPENDIX

Curie Level in LASL-38 After 250 Days of Irradiation as a Function of Decay
Time After Last Cycle (50% Duty Factor Assumed)

Isotope	Curies						
	0	1 h	12 h	1 d	7 d	14 d	30 d
⁷⁸ Ge	5.1	3.6					
^{77A} As	1.3	1.3	1.06				
⁷⁸ As	5.1	4.8					
⁷⁹ As	14.2						
^{79M} Se	14.2						
^{81M} Se	2.1	1.0					
⁸¹ Se	35.5	3.7					
^{83M} Se	73.5						
⁸³ Se	55.8	10.6					
⁸⁴ Se	233						
⁸⁵ Se	279						
⁸⁷ Se	507						
^{83A} Br	73.5	54.8	2.0				
^{83B} Br	55.8	48.0	1.8				
^{84M} Br	4.8						
⁸⁴ Br	233	70.1					
⁸⁵ Br	279						
⁸⁷ Br	631						
⁸⁸ Br	661						
⁸⁹ Br	654						
^{83M} Kr	129	124	12.3				
^{85M} Kr	329	285	50.2	7.6			
⁸⁵ Kr	3.2	3.2	3.2	3.2	3.2	3.2	3.2
⁸⁷ Kr	631	375	1.1				
⁸⁸ Kr	890	695	45.6	2.3			
⁸⁹ Kr	1160						
⁹⁰ Kr	1270						
⁹¹ Kr	874						
⁹² Kr	474						
⁹³ Kr	122						
⁸⁸ Rb	905	769	51.0	2.6			
⁸⁹ Rb	1210	95.6					
⁹⁰ Rb	1460						
⁹¹ Rb	1380	71.1					
⁹² Rb	1340						
⁹³ Rb	1550						
⁸⁹ Sr	710	710	705	700	645	585	469
⁹⁰ Sr	24.4	24.4	24.4	24.4	24.4	24.4	24.3
⁹¹ Sr	1470	1400	640	272			
⁹² Sr	1340	1040	55.4	2.3			

APPENDIX (continued)

Isotope	Curies						
	0	1 h	12 h	1 d	7 d	14 d	30 d
⁹³ Sr	1550	9.8					
⁹⁴ Sr	1320						
⁹⁰ Y	24.0	24.0	24.0	24.1	24.3	24.3	24.3
^{91M} Y	884	865	411	175			
⁹² Y	1530	1470	356	43.6			
⁹³ Y	1550	1460	684	298			
⁹⁴ Y	1370	164					
⁹⁵ Y	1570	31.7					
⁹⁵ Zr	885	885	880	876	822	763	643
⁹⁷ Zr	1500	1440	918	563	1.6		
^{95M} Nb	16.3	16.3	16.4	16.5	16.4	15.5	13.2
⁹⁵ Nb	779	779	780	781	789	789	764
^{97M} Nb	1430	1380	881	541	1.6		
⁹⁷ Nb	1540	1510	989	607	1.7		
⁹⁸ Nb	16.2	7.2					
⁹⁹ Nb	1540						
¹⁰⁰ Nb	1600						
⁹⁹ Mo	1530	1520	1350	1190	267	46.5	
¹⁰¹ Mo	1270	84.9					
¹⁰² Mo	1040	22.9					
¹⁰⁵ Mo	228						
^{99M} Tc	1330	1330	1260	1130	255	44.4	
¹⁰¹ Tc	1270	779					
^{102M} Tc	520	14.3					
¹⁰² Tc	520	23.2					
¹⁰³ Tc	760						
¹⁰⁴ Tc	456	52.5					
¹⁰⁵ Tc	228	4.4					
¹⁰⁷ Tc	48.1						
¹⁰³ Ru	461	461	457	453	408	362	274
¹⁰⁵ Ru	228	202	35.9	5.4			
¹⁰⁶ Ru	30.1	30.1	30.0	30.0	29.7	29.3	28.4
¹⁰⁷ Ru	48.1						
^{103M} Rh	461	461	457	453	409	362	275
^{105M} Rh	45.6	39.0	6.9	1.1			
¹⁰⁵ Rh	228	228	201	163	10.2		
¹⁰⁶ Rh	30.1	30.1	30.0	30.0	29.7	29.3	28.4
¹⁰⁷ Rh	48.1	8.9					
¹⁰⁸ Rh	17.7						
¹⁰⁹ Rh	7.6						
¹⁰⁹ Pd	7.6	7.2	4.1	2.2			
¹¹¹ Pd	4.8						
¹¹⁰ Pd	2.5	2.5	1.7	1.2			

APPENDIX (continued)

Isotope	Curies						
	0	1 h	12 h	1 d	7 d	14 d	30 d
¹⁰⁹ M _{Ag}	7.6	7.2	4.1	2.2			
¹¹¹ M _{Ag}	4.8						
¹¹¹ Ag	4.4	4.4	4.2	4.0	2.3	1.2	
¹¹² Ag	2.5	2.5	2.0	1.3			
¹¹³ Ag	4.1	3.6					
¹¹⁴ Ag	3.6						
¹¹⁵ M _{Ag}	2.5						
¹¹⁵ Ag	2.0						
¹¹⁵ Cd	2.5	2.4	2.1	1.8			
¹¹⁷ M _{Cd}	2.8	2.2					
¹¹⁷ Cd	1.7	1.6					
¹¹⁸ Cd	3.6	1.3					
¹¹⁵ M _{In}	2.5	2.5	2.3	2.0			
¹¹⁷ M _{In}	1.7	1.5					
¹¹⁷ In	1.5	1.4					
¹¹⁸ In	3.6	1.3					
¹¹⁹ In	3.6						
¹¹⁹ M _{Sn}	1.4	1.4	1.4	1.4	1.4	1.3	1.3
¹²¹ Sn	3.8	3.7	2.8	2.1			
¹²⁵ M _{Sn}	2.0						
¹²⁵ Sn	2.8	2.8	2.7	2.6	1.7	1.0	
¹²⁷ Sn	32.9	25.2	1.3				
¹²⁸ Sn	93.8	45.2					
¹³⁰ Sn	50.7						
¹²⁷ Sb	32.6	32.6	30.6	27.9	9.1	2.4	
¹²⁸ M _{Sb}	12.7	11.9	5.4	2.3			
¹²⁸ Sb	91.2	53.0					
¹²⁹ Sb	203	195	50.9	8.5			
¹³⁰ Sb	507	2.3					
¹³¹ Sb	659	118					
¹³² Sb	836						
¹³³ Sb	1010						
¹³⁴ Sb	456						
¹²⁷ M _{Te}	4.5	4.5	4.5	4.5	4.5	4.3	3.9
¹²⁷ A _{Te}	26.1	26.1	25.3	23.7	7.9	2.1	
¹²⁷ B _{Te}	4.5	4.5	4.5	4.5	4.3	4.1	3.7
¹²⁹ M _{Te}	55.6	55.6	55.4	54.9	48.4	41.8	29.9
¹²⁹ A _{Te}	114	110	24.7	4.1			
¹²⁹ B _{Te}	55.7	55.7	55.2	54.6	48.2	41.6	29.7
¹³¹ M _{Te}	112	110	84.5	63.3	2.0		
¹³¹ A _{Te}	560	271					
¹³¹ B _{Te}	22.3	22.0	16.9	12.7			

APPENDIX (continued)

Isotope	Curies						
	0	1 h	12 h	1 d	7 d	14 d	30 d
¹³² Te	1100	1100	993	893	248	55.7	1.8
^{133M} Te	1240	594					
^{133A} Te	436						
^{133B} Te	162	76.9					
¹³⁴ Te	1750	653					
¹³⁵ Te	682						
^{131A} I	180	179	175	171	107	58.9	15.0
^{131B} I	521	520	501	480	287	158	40.1
¹³² I	1100	1100	1020	919	256	57.4	18.9
^{133A} I	1080	1070	756	507	4.2		
^{133B} I	598	579	402	269	2.2		
¹³⁴ I	1980	1470					
¹³⁵ I	1550	1400	448	130			
¹³⁶ I	786						
¹³⁷ I	674						
¹³⁸ I	370						
¹³⁹ I	659						
^{133M} Xe	40.3	40.3	39.2	36.9	7.7		
¹³³ Xe	1600	1600	1590	1550	773	311	38.5
^{135M} Xe	464	434	140	40.5			
¹³⁵ Xe	1600	1590	1120	589			
¹³⁷ Xe	1520						
¹³⁸ Xe	1390	121					
¹³⁹ Xe	1370						
¹⁴⁰ Xe	963						
¹³⁶ Cs	76.2	76.0	74.2	72.2	52.5	36.1	15.4
¹³⁷ Cs	24.3	24.3	24.3	24.3	24.3	24.3	24.2
¹³⁸ Cs	1450	690					
¹³⁹ Cs	1640	21.6					
¹⁴⁰ Cs	1520						
¹⁴² Cs	811						
^{137M} Ba	23.3	23.3	23.3	23.3	23.3	23.3	23.3
¹³⁹ Ba	1660	1140	5.2				
¹⁴⁰ Ba	1270	1270	1240	1210	877	604	257
¹⁴¹ Ba	1600	161					
¹⁴² Ba	1370	33.1					
¹⁴³ Ba	1130						
¹⁴⁰ La	1220	1220	1230	1230	993	692	295
¹⁴¹ La	1620	1450	198	22.2			
¹⁴² La	1500	1040	4.8				
¹⁴³ La	1520	78.9					
¹⁴¹ Ce	1020	1020	1010	1000	884	763	545

APPENDIX (continued)

Isotope	Curies						
	0	1 h	12 h	1 d	7 d	14 d	30 d
¹⁴³ Ce	1520	1500	1190	925	45.1	1.3	
¹⁴⁴ Ce	518	518	517	516	509	500	481
¹⁴⁵ Ce	1010						
¹⁴⁶ Ce	778	39.9					
¹⁴³ Pr	1150	1150	1160	1150	924	654	293
¹⁴⁴ Pr	518	518	517	517	509	500	481
¹⁴⁵ Pr	1100	906	254	63.5			
¹⁴⁶ Pr	778	284					
¹⁴⁷ Nd	495	494	480	465	319	205	75.0
¹⁴⁹ Nd	286	195	2.8				
¹⁵¹ Nd	112	3.5					
¹⁴⁷ Pm	87.6	87.6	87.7	87.8	89.2	90.0	90.1
¹⁴⁹ Pm	286	286	254	218	34.2	4.0	
¹⁵¹ Pm	112	110	83.0	61.3	1.6		
¹⁵³ Sm	38.0	37.5	31.8	26.7	3.2		
¹⁵⁵ Sm	8.4	1.5					
¹⁵⁶ Sm	3.3	3.1	1.4				
¹⁵⁵ Eu	1.8	1.8	1.8	1.8	1.8	1.8	1.8
¹⁵⁶ Eu	2.7	2.7	2.7	2.6	2.0	1.5	
¹⁵⁷ Eu	2.0	1.9	1.2				
Totals	108560	48730	27320	21880	10850	7920	5300