**EXPERIMENT DESCRIPTION
AND HAZARDS EVALUATION FOR THE
PACIFIC NORTHWEST LABORATORY
MIXED OXIDE ($\text{UO}_2\text{-PuO}_2$)
IRRADIATIONS IN EBR-I I, TASK A,
SUBTASK 1 IRRADIATIONS**

J. E. HANSON

JULY 1968

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DEVELOPMENT REPORT**

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PACIFIC NORTHWEST LABORATORY
MIXED OXIDE (UO_2 - PuO_2) IRRADIATIONS
IN EBR-II TASK A, SUBTASK 1 IRRADIATIONS

By

J. E. Hanson
FFTF Fast Reactor Fuels Section
FFTF Project

July 1968

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AND
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MIXED OXIDE (UO_2 - PuO_2) IRRADIATIONS
IN EBR-II
TASK A, SUBTASK 1 IRRADIATIONS
J. E. Hanson

I. PURPOSE AND OBJECTIVES

A description of PNL's program for development of the FFTF-FTR driver fuel⁽¹⁾ was presented on February 1, 1967 to the EBR-II Evaluations Committee. As emphasized in that meeting, the proposed irradiations in EBR-II are directed solely toward this end.

Recent developments in mixed oxide fuel irradiation experience have permitted consideration of some reduction in the number of fuel design parameters in the FTR driver fuel development program. Accordingly, the Task A, Subtask 1 irradiations herein described differ to some extent from those previously described.⁽¹⁾ Generally, in addition to providing fast-flux irradiated fuel specimens for subsequent transient irradiation, the 19-capsule subassemblies will also include specimens for evaluating the effects of cladding thickness reduction and alternate fuel forms on fuel performance.

The purpose and objectives of the proposed 19-capsule subassembly irradiations are:

1. To obtain preliminary fast flux irradiation experience at intermediate and approximately FFTF goal exposures with mechanically-blended fuel in forms selected as principal candidates for the FTR driver fuel. These

forms (see Table I) include solid pellets of 93, 92, and 90% theoretical density, and annular pellets of 96% theoretical density.

2. To obtain fast flux irradiation performance data on fuel specimens containing a reference design axial fuel motion restrictor. This restrictor design, only very recently selected, was based on analysis of results obtained from a series of irradiations in the MTR VH-4 facility wherein restrictor geometry, material, and degree of freedom within the cladding were variables. Details of the reference axial fuel motion restrictor design are delineated in Appendix A Drawing H-3-27907.
3. To obtain fuel specimens irradiated in a fast flux for subsequent evaluation of the fuel and cladding transient performance through irradiation in TREAT. It will be noted from Table I that capsules 1 through 12 incorporate four groups of three specimens each. Following the completion of the steady-state irradiation in EBR-II, one capsule from each group of three will be destructively examined to characterize the condition of the fuel and cladding. The remaining two specimens in each group will be re-encapsulated for transient irradiation in TREAT.* In this fashion, it should be possible to separate steady-state and transient irradiation effects on the fuel and cladding. The purpose of the transient irradiations in TREAT is to establish failure thresholds and to determine failure mechanisms of the irradiated mixed oxide fuel. In addition, the extent of axial fuel redistribution as a result of transient melting will be evaluated as a function of the presence or absence of

* This results in a total of 16 transient irradiations; 8 at 10,000 MWd/tonne burnup and 8 at 50,000 MWd/tonne burnup.

axial fuel motion restrictors. Gross failure of the fuel specimens during the transient irradiations are not planned as part of this program.

4. To provide irradiation experience with fuel specimens whose cladding thickness has been reduced to simulate pessimistic end-of-life conditions. These tests may also provide data on the role of cladding strength in the restraint of fuel swelling.

II. DESCRIPTION OF THE FUEL AND CLADDING AND SUMMARY OF PRIOR IRRADIATION EXPERIENCE

Fuel to be used in these experiments is hypostoichiometric ($O/M = 1.97 - 1.98$) mechanically-mixed 25% PuO_2 -75% UO_2 sintered in a hydrogen atmosphere at 1550 °C for 4 hr to yield pellet densities of 90 to 96% TD. This sintering process yields a solid solution of PuO_2 and UO_2 virtually indistinguishable from solid solutions of PuO_2 and UO_2 prepared from the co-precipitation process. Cladding for the fuel pins will be type 304 stainless steel. Materials for the capsule components were obtained from Argonne National Laboratory.

The prior irradiation experience of mixed oxide (PuO_2 - UO_2) fuel has been recently reviewed in a number of papers.⁽²⁻¹⁰⁾ The latest paper⁽¹⁰⁾ presents preliminary post-irradiation examination data from encapsulated mixed-oxide fuel specimens irradiated in EBR-II to peak burnups of about 5 at.%. In general, it can be concluded that the performance of mixed oxide fuel has been reasonably well established up to burnups of 5 at.%. Tests currently in progress in EBR-II and Dounreay will shortly provide data on fuel performance at burnups greater than 5 at.% in a fast flux. Thermal flux tests of mixed oxide fuel specimens to burnups beyond 12 at.% have also been successfully completed.⁽¹¹⁾

III. FUEL ELEMENT AND CAPSULE DATA

Design data are presented in Table I and isotopic analyses of the uranium and plutonium feed materials in Table II. As-fabricated data will be provided in an addendum to this Experiment Description and Hazards Evaluation.

TABLE I. Fuel Specimen Data (Design values)

Specimens in Subassembly PNL-2 Are Identical to Those in PNL-1

PNL 1	Capsule Designation	Fuel Column Length, in.	Fuel Pellet OD, in.	Fuel Pellet ID, in.	Fuel Pellet Density, % TD**	Fuel Smeared Density, % TD**	Fuel Weight, g	Fuel-Cladding Diameter Gap, in.	Depleted UO ₂ Pellet Diameter, in.	Depleted UO ₂ Length, in.	Depleted UO ₂ Pellet Weight, in. (total)	Cladding OD,* in.	Cladding Thickness, in.	Axial Motion Restrictor
1	13.5	0.212	-	93	88	80.62	0.006	0.214	1.0	5.8	0.250	0.016	no	
2						80.62							no	
3						80.62							no	
4						79.12							yes	
5						79.12							yes	
6						79.12							yes	
7			0.054	96	85	77.87							no	
8						77.87							no	
9						77.87							no	
10						76.43							yes	
11						76.43							yes	
12						76.43							yes	
13				93	88	80.62					0.238	0.010	no	
14				93	88	80.62								
15			0.054	96	85	77.87								
16			0.054	96	85	77.87								
17			-	89.9	85	77.87								
18			-	89.9	85	77.87								
19		0.210	-	91.6	85	77.87	0.008							

* Cladding material is type 304 stainless steel

** Denotes theoretical density

Theoretical density of 25% PuO₂-75% UO₂ is 11.0925 g/cc

TABLE II. *Isotopic Analyses of Uranium and Plutonium
Feed Materials for Fuel Specimens*

<u>Isotope</u>	<u>Ratio, %</u>
^{234}U	$^{234}\text{U}/\text{U} = 0.98$
^{235}U	$^{235}\text{U}/\text{U} = 93.17$
^{236}U	$^{236}\text{U}/\text{U} = 0.26$
^{238}U	$^{238}\text{U}/\text{U} = 5.59$
^{239}Pu	$^{239}\text{Pu}/\text{Pu} = 88.3$
^{240}Pu	$^{240}\text{Pu}/\text{Pu} = 10.3$
^{241}Pu	$^{241}\text{Pu}/\text{Pu} = 1.3$
^{242}Pu	$^{242}\text{Pu}/\text{Pu} = 0.1$

Also common to all the specimens and capsules, but not provided in Table I, are the following design requirements:

1. The fuel specimens will be all helium gas-bonded to the cladding.
2. All the fuel specimens will initially contain helium at one atmosphere pressure. The volume of helium at STP is calculated to be 7.1 cc.
3. The design value for the capsule sodium level, as shown in Drawings H-3-27687 and H-3-27802 (see Appendix A), is 5-11/64 in. $\pm$ 1 in. above the top of the fuel. The sodium level as a function of temperature is given in Figure 1. The capsules will be back-filled with helium at one atmosphere pressure after being sodium filled. The calculated helium volume is 12.6 cc at STP. Volume and pressure as a function of temperature are given in Figures 2 and 3, respectively.
4. Capsule identification will be by numbers and letters stamped in the locations shown in Drawings H-3-27687 and H-3-27802. These symbols correspond to those given in Table I for each fuel specimen.

FIGURE 1. Sodium Level Versus Sodium Temperature
in Hottest Specimen in Subassembly.
Position 7E4/1, 45 MW Reactor Power,
1.685 gal/min Flow

FIGURE 2. Internal Void Volume as a Function of Capsule Temperature. 7E4/1, 45 MW Reactor Power, 1.685 gal/min Flow

FIGURE 3. Internal Capsule Pressure as a Function of Capsule Temperature. Position 7E4/1, 45 MW Reactor Power, 1.685 gal/min Flow

IV. OTHER MATERIALS DATA

As shown in Table I (capsule designation), specimens PNL 1-4, 1-5, 1-6, 1-10, 1-11, 1-12, 2-4, 2-5, 2-6, 2-10, 2-11, and 2-12 contain fuel axial motion restrictors. Selection of the restrictor material and design was accomplished through a series of screening tests conducted in the VH-4 facility in the MTR.⁽¹²⁾ The basis for selection was the effectiveness of the design with regard to controlling axial redistribution of molten fuel.

Candidate restrictor materials tested were:

- 1) Tungsten
- 2) 304 Stainless Steel
- 3) Molybdenum
- 4) Nickel
- 5) Depleted UO_2

Candidate restrictor designs were floating (unattached to cladding) and fixed (crimped to cladding). The test capsule design and restrictor designs are illustrated in Figure 4. The resultant reference axial fuel motion restrictor design selected is delineated in Drawing H-3-27907.

As illustrated in Drawings H-3-27688 and H-3-27801 (Appendix A), the spiral wire wrapping was replaced by 304 stainless steel tubing containing cobalt or iron flux monitor wire. The tubing was sealed to prevent the flux monitor wire from contacting the bonding sodium. Flux monitor wire studies are conducted in cooperation with the PNL FFTF-oriented dosimetry studies program.

FIGURE 4. VH-4 Restrictor Screening Capsules

V. DRAWINGS OF FUEL SPECIMENS, CAPSULES, AND OTHER DEVICES

The fuel specimens and capsule assembly details for the fuel motion restrictor tests are delineated in PNL Drawings H-3-27687, H-3-23688, H-3-15274, and H-3-27907 (Appendix A).

The fuel specimens and capsule assembly details for the reduced cladding thickness tests are delineated on PNL Drawings H-3-27801, H-3-27802, and H-3-15274.

The purpose of the extensometer (Drawing H-3-15274) in each fuel specimen is to indicate the maximum length of the fuel column and to prevent separation of the pellets during handling by axially restraining the column.

VI. FABRICATION AND TESTING OF CAPSULES PRIOR TO IRRADIATION

A. Fuel Specimens

Quality assurance for the fuel specimens will be attained through:

1. Ultrasonic and eddy-current inspection of cladding material, and rejecting defects greater than 1 mil deep, 4 mils wide, and 30 mils long.
2. Air gage and profilometer measurements for the cladding ID and OD, respectively.
3. Selective fitting of fuel pellets to cladding to obtain specified test parameters (gap and smeared density).
4. Mass spectrometer helium leak tests of closure welds. Limit of leak detector sensitivity is 1.0×10^{-10} cc/sec.
5. X-radiography of completed specimens to ascertain proper fuel and component positions, dimensions, and weld quality.

B. Capsules

1. Capsule tubes will be subjected to ultrasonic and eddy current inspection to verify soundness. Defects greater than 10% of the wall thickness will be cause for rejection.
2. Closure welds will be X-radiographed and helium-leak tested. Leak detector sensitivity will be 1.0×10^{-10} cc/sec.
3. Sodium filling will be accomplished in accordance with the procedure given in Appendix B.
4. Sodium bonding will be eddy-current tested to detect voids larger than 1/32 in. in diameter.
5. X-radiographs of the capsules will be taken to verify fuel location and component positions, and to further evaluate the sodium bond integrity.

VII. SUBASSEMBLY DATA

A. Capsule Materials and Weights

1. <u>Fuel</u> (from Table II)	<u>Weight, g</u>
5 specimens @ 80.62 g each	403.10
3 specimens @ 79.12 g each	237.36
8 specimens @ 77.87 g each	621.96
3 specimens @ 76.43 g each	<u>229.29</u>
Total weight	1491.71
2. <u>Stainless Steel</u>	
a) <u>Fuel pin</u>	
<u>Cladding</u>	
11 specimens, 0.250 in. OD x 0.016 in. wall @ 47.25 g each	519.75
9 specimens, 0.238 in. OD x 0.010 in. wall @ 287 g each	258.3

	<u>Weight, g</u>
<u>Pedestal</u>	
19 specimens @ 5.12 g each	97.28
<u>Spacer Tube</u>	
19 specimens @ 4.35 g each	82.65
<u>Spiral Wrap Tube</u>	
19 specimens @ 4.45 g each	84.55
<u>Extensometer</u>	
19 specimens @ 6.33 g each	120.27
<u>End Cap</u>	
19 specimens @ 2.98 g each	<u>56.62</u>
Total weight	1219.42
b) <u>Capsule</u>	
<u>Top Cap</u>	
19 specimens @ 10.45 g each	198.55
<u>End Plug</u>	
19 specimens, 2 per specimen @ 3.37 g each	128.06
<u>Bottom Cap</u>	
19 specimens @ 10.50 g each	199.5
<u>Tube</u>	
19 specimens @ 106.20 g each	2017.80
<u>Fill Tube</u>	
19 specimens @ 3.0 g each	<u>57.0</u>
Total weight	2600.91
Total stainless steel	3820.33

	<u>Weight, g</u>
3. <u>Nickel</u>	
19 specimens @ 11.5 g	218.5
4. <u>Sodium</u>	
12 specimens @ 12.75 g	153.0
7 specimens @ 14.28 g	<u>100.0</u>
	253.0
5. <u>Cobalt</u>	
10 specimens @ 1.35 g	13.5
6. <u>Iron</u>	
9 specimens @ 1.2 g	10.8

B. Desired Reactor Locations

1. Subassembly PNL-1

The desired reactor position for subassembly PNL-1 is row 7, sector E, position 4. The location of each fuel specimen within the subassembly for this location is given in Figure 5. Alternate positions such as 7A4, 7B4, 7C4, 7D4, and 7F4 are also suitable with rearrangement of the specimen locations within the subassembly to provide the same specimen heat generation rates as in the 7E4 location. As an alternate, a row 6, sector E, position 3 (or equivalent position in other sectors) could be used. Performance calculations for a row 6 position are given in Appendix D.

The desired average fuel exposure for subassembly PNL-1 is 10,000 MWd/tonne (U + Pu) which corresponds to about 75 EFPD (@ 45 MW power) of reactor operation.

<u>Cap. Pos. No.</u>	<u>Spec. No.*</u>	<u>Cap. Pos. No.</u>	<u>Spec. No.*</u>
1	19	11	1
2	7	12	2
3	8	13	18
4	9	14	4
5	10	15	5
6	15	16	13
7	16	17	3
8	11	18	6
9	12	19	14
10	17		

**See Table II for description of fuel specimens.*

FIGURE 5. *Required Specimen Orientation Within Sub-assemblies PNL-1 and PNL-2. Based on Reactor Grid Position 7E4.*

2. Subassembly PNL-2

Subassembly PNL-2, to be an exact duplicate of subassembly PNL-1, also requires a row 7, position 4 location. The specimen locations within the subassembly should be the same as those for subassembly PNL-1.

The desired average fuel exposure for subassembly PNL-2 is 50,000 MWd/tonne (U + Pu) which corresponds to about 375 EFPD (@ 45 MW power) of reactor operation.

3. Mutual Concurrence Required

Mutual concurrence between ANL and PNL on subassembly positions within the reactor grid, and corresponding specimen locations within the subassemblies must be established before the specimens are assembled into a subassembly. A PNL representative will be present during assembly of the capsules into subassemblies.

VIII. NUCLEAR EFFECTS OF THE SUBASSEMBLY IN EBR-II

- A. The calculated reactivity effect, ΔK , for these capsule subassemblies is -0.0046 based on two-dimensional diffusion theory calculations by EXTERMINATOR code. The ΔK is relative to a standard EBR-II driver fuel element in the center of the reactor.
- B. The flux perturbation in adjacent EBR-II driver fuel subassemblies due to the insertion of this subassembly is 0.988. This perturbation is based on a value of 1.000 for an EBR-II driver element in the center hole.
- C. Test subassembly effects on the worth of the control rods would be expected to approximate those for the driver fuel subassemblies.

IX. PERFORMANCE OF THE SUBASSEMBLY DURING IRRADIATION

The performance predictions for each fuel specimen were made with the aid of the EBR-II Hazards Analysis code prepared by the late C. R. Reinke of Argonne National Laboratory to predict fuel specimen performance for a number of EBR-II irradiations, including mixed oxide specimens SOV-5 and SOV-6.⁽¹³⁾ The equations used in the code are those given in the Guide.⁽¹⁴⁾ Fissioning rates, Figure 6, are based on the 91-subassembly core.⁽¹⁵⁾ Gamma heating values are based on the revised values for uranium, sodium, and iron given in Figure 17 (revision 2, corrected by MAB) of the Guide. It was necessary to extrapolate the curves to radii corresponding to the row 7 positions. Although these extrapolations probably do not yield highly precise values, errors because of the low gamma heating effect relative to the fission contribution are considered insignificant.

A. Specimen Heat Generation Rates and Temperatures

Table III presents the calculated heat generation rates, temperatures, capsule surface heat fluxes, and coolant ΔT 's for each position (and specimen) in the subassembly. Curves of these parameters as a function of flow rate will be found in Appendix C.

- B. Total heat generation of the subassembly is calculated to be 183.7 kW based on a peak-to-average flux ratio of 1.14:

$$\left(\frac{186.1}{1.14} \right) \left(\frac{13.5}{12} \right) = 183.7$$

- C. Total coolant flow through the subassembly required to remove 183.7 kW of heat and:

FIGURE 6. Fission Rate Versus Distance. 62.5 MW Reactor Power, 91 Subassembly Core

TABLE III. Specimen Heat Generation Rates and Temperatures*

Position Number*	Specimen Number**	Maximum Heat Generation Rate, kW/ft	Maximum [†] Mean Capsule Wall Temp, °C	Maximum Capsule Surface Heat Flux, W/cm ²	Coolant ^{††} ΔT , C°
1	PNL 1-19	10.2	481 ^{††}	111.9	87.4
2	- 7	10.1	479	110.6	86.5
3	- 8	9.9	478	109.3	85.0
4	- 9	10.1	479	107.7	86.5
5	-10	10.0	478	106.3	85.3
6	-15	9.8	477	109.1	84.0
7	-16	9.7	475	107.7	83.0
8	-11	9.9	478	106.3	85.0
9	-12	9.8	477	108.2	84.0
10	-17	9.7	475	106.3	83.0
11	- 1	9.9	478	106.3	85.0
12	- 2	9.7	475	108.2	83.0
13	-18	9.7	475	106.3	83.0
14	- 4	9.9	478	104.5	85.0
15	- 5	9.7	475	106.3	83.0
16	-13	9.5	473	104.5	81.5
17	- 3	9.7	475	106.3	83.0
18	- 6	9.5	473	104.2	81.5
19	-14	9.3	471	102.5	79.5
		$\Sigma = 186.1$			
					$\Sigma = 1594.0$
					$\Delta T_{ave} = 84 \text{ C}^\circ$

* Based on core position 7E4

** Specimens and their arrangement within the subassembly are identical in subassembly PNL-2.

† Based on coolant flow rate of 1.79 gpm per specimen (32 gpm total).

†† Uncertainty factor 15 C° (see Appendix C) increases this to 496 °C.

- 1) not exceed the coolant mixed mean maximum ΔT of 260 F° (144 C°), or minimum ΔT of 60 F° (33 C°) at 19.6 gpm and 73 gpm, respectively, based on the hottest specimen in the subassembly.*
- 2) not exceed the maximum mean capsule wall temperature of 1105 °F (including uncertainty factors) at 16.9 gpm.
- 3) not exceed the flow rate through adjacent driver fuel assemblies (row 6) at 73 gpm.

The desired coolant flow rate is the maximum possible attainable in an unorificed MK A-19 subassembly with the strainer removed.

- D. An unorificed Mark A capsule subassembly will provide a flow rate of approximately 32 gpm.⁽¹⁶⁾
- E. The mixed mean coolant temperature at the subassembly exit is calculated to be 851 °F. (371 + 84 = 455 °C, or 851 °F.)
- F. Thermal conductivities for the materials in the capsules are presented in Table IV.

TABLE IV. Thermal Conductivity Values for Capsule Materials

<u>Material</u>	<u>Thermal Conductivity, Btu/(hr-ft²) - (F°/ft)</u>
304 Stainless Steel	9.9
Sodium	36.2
Fuel	Variable thermal conductivity (see Figure 8 for $\int k dt$ vs T curve)
Fuel-Cladding Gap Conductance	1000 Btu/(hr-ft ²) - F° (see Figure 7)

* Row 6 @ 45 MW power is 160 F°. Mixed mean coolant outlet temperature must be within ± 100 F°. $\Delta T_{max} = 260$ F°, $\Delta T_{min} = 60$ F°.

- G. Figure 7 presents experimental data for gap conductivity in 1/4 in. diam mixed oxide pins.
- H. The fuel center temperature was estimated by means of the $\int_0^{T_c} k dt$ versus temperature curve, Figure 8, to account for the variation of fuel thermal conductivity with temperature. A typical calculation is given in Appendix C. Axial temperatures with and without uncertainty factors for the hottest fuel specimen (number 19, position number 1) are given in Figures 9 and 10.

I. Radial and Axial Expansion and Growth of Fuel

The selection of the principal fuel parameters for these tests considered the accommodation of radial and axial thermal expansion of the fuel as well as fuel swelling and growth due to the accumulation of gaseous and solid fission products. Verification of the assumptions made in establishing the fuel dimensions is a principal objective of this test. Based on currently available mixed oxide fuel performance data, gross fuel-cladding interaction resulting in significant expansion of the cladding is not anticipated in these tests.

J. Fission Gas Pressure and Resultant Clad Stress

The volume of fission gases formed in the hottest specimen in the subassembly after 50,000 MWd/tonne will be 101.20 cm³. Assuming 100% gas release and a plenum temperature of 460 °C, the pressure will be 925 psig.

K. Cladding Thermal and Hoop Stresses

The thermal and hoop stresses in the cladding of the hottest specimen, PNL-2-19, were calculated to be 7,154 and 10,086 psi, respectively.

FIGURE 7. Conductivity Versus Gap Size, Argon atm

FIGURE 8. Mixed Oxide $\int_0^{T_c} k dt$ Design Curve (95% Dense
Pellet O/M = 1.98, 20% PuO_2 -80% UO_2)

FIGURE 9. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 7E4/1, 45 MW Reactor Power, 10.2 kW/ft (Peak) Without Uncertainty Factors*

FIGURE 10. *Calculated Axial Temperature Profile for Specimen #19 (Position #1), 10.2 kW/ft Maximum Linear Heat Generation Rate, 1.69 gal/min Flow (with uncertainty factors)*

L. Irradiation Effect on Clad and Capsule Material Strength

Based on currently available data, an increase in cladding yield strength and a decrease in ductility can be anticipated as a consequence of fast flux irradiation.

- M. Capsule wall thermal stress for the hottest capsule in subassembly PNL-2 was calculated to be 9,346 psi. Calculated hoop stress is 355 psi under normal conditions, and 2,399 psi with a ruptured fuel pin (at a burnup of 50,000 MWd/MT).

X. POSTIRRADIATION CHARACTERISTICS OF SUBASSEMBLY AND CAPSULES

- A. The decay heat production of the subassemblies as a function of time is shown in Figure 11. The decay heat production of the maximum pin, PNL-2-19, as a function of time is shown in Figure 12. These calculations were made on the basis of a continuous irradiation for a period of 365.2 days at design power using the RIBD Computer Code.⁽¹⁷⁾ Assuming a 65% plant factor for the EBR-II rather than continuous operation, the decay heat production of the subassembly after 10 days of cooling was determined to be less than 1 kW, and easily within the cooling capability of the IBC.
- B. The minimum cooling period required in the EBR-II storage basket is 10 days.
- C. The estimated gamma radiation as a function of time for the subassembly is shown in Figure 13. The estimated gamma radiation as a function of time for the hottest pin in the subassembly is shown in Figure 14. Gamma radiation by energy level after 10 days of cooling is given in Table V.

FIGURE 11. Subassembly Decay Heat as Function of Time After Irradiation

FIGURE 12. Decay Heat from Single Pin as Function of Time After Irradiation

FIGURE 13. Point Source Dose Rate from Irradiated Subassembly (Unshielded) as Function of Cooling Time

FIGURE 14. Point Source Dose Rate from Hottest Fuel Pin in Subassembly (Unshielded) as Function of Cooling Time

TABLE V. *Subassembly Radiation Level After 10 Days Cooling*

<u>Group Average</u>	<u>Average Energy, MeV</u>	<u>Dose Rate At Dose Point R/hr x 10⁻¹⁰</u>
1	0.1500	0.2783
2	0.2500	0.02509
3	0.3500	0.1288
4	0.4750	0.3936
5	0.6500	0.3974
6	0.8250	0.8992
7	1.000	0.008528
8	1.225	0.02638
9	1.475	0.03849
10	1.700	0.5304
11	1.900	0.02716
12	2.100	0.01193
13	2.300	0.002651
14	2.500	0.04528
15	2.700	0.02824
16	3.000	0.02337
Total		2.865

D. Estimated gamma radiation level of stainless steel hardware is 6.70×10^8 R/hr after a 10 day cooling period.

E. The curie content of each capsule is listed in Table VI.

XI. HAZARDS EVALUATION

A number of hazards evaluations for the irradiation of encapsulated mixed oxide fuel specimens have been prepared by others. (13,18-20) These analyses show that the irradiation of encapsulated mixed oxide fuel specimens present no significant hazards to the operating safety of EBR-II. Moreover, as was recently demonstrated, the presence of failed fuel specimen cladding (specimen F-2-F, subassembly XG06) can be successfully tolerated by the capsule.

*TABLE VI. Isotopic Curie Content of Each Capsule
After 10 Days Cooling*

<u>Isotope</u>	<u>Curies/Rod</u>	<u>Isotope</u>	<u>Curies/Rod</u>	<u>Isotope</u>	<u>Curies/Rod</u>
Kr ⁸⁵	288	Sb ¹²⁵	2.3×10^2	Pm ¹⁴⁸	3.2×10^3
Rb ⁸⁶	29.3	Te ¹²⁵	46	Pm ¹⁴⁹	1.66×10^3
SR ⁸⁹	3.99×10^3	Sb ¹²⁶	2.3×10^2	Pm ¹⁵¹	44.2
Y ⁸⁹	8	Sb ¹²⁷	1.3×10^2	Sm ¹⁵¹	4.4
Sr ⁹⁰	1.8×10^3	Te ¹²⁷	3.7×10^3	Sm ¹⁵³	5.9×10^2
Y ⁹⁰	1.8×10^3	Te ¹²⁹	1.16×10^4	Eu ¹⁵⁴	59
Y ⁹¹	5.6×10^4	Te ¹³¹	44.1	Eu ¹⁵⁵	92
Zr ⁹⁵	1.04×10^5	I ¹³¹	3.2×10^4	Eu ¹⁵⁶	7.0×10^3
Nb ⁹⁵	1.15×10^5	Xe ¹³¹	5.3×10^2	Tb ¹⁶⁰	16.5
Mo ⁹⁹	9.8×10^3	Te ¹³²	1.23×10^4	Tb ¹⁶¹	35.3
Tc ⁹⁹	9.8×10^3	I ¹³²	1.26×10^1	Gd ¹⁶²	11.2
Ru ¹⁰³	8.8×10^4	I ¹³³	39.9	Zr ⁹⁷	6.4
Rh ¹⁰³	8.8×10^4	Xe ¹³³	4.3×10^4	Nb ⁹⁷	12.7
Rh ¹⁰⁵	600	Cs ¹³⁴	1.54×10^3		
Ru ¹⁰⁶	3.0×10^4	Cs ¹³⁶	1.28×10^3		
Rh ¹⁰⁶	3.0×10^4	Cs ¹³⁷	2.95×10^3		
Ag ¹¹⁰	40	Ba ¹³⁷	2.7×10^3		
Ag ¹¹¹	1.63×10^3	Ba ¹⁴⁰	6.4×10^4		
Cp ¹¹⁵	10.5	La ¹⁴⁰	7.3×10^4		
IN ¹¹⁵	4.9	Ce ¹⁴¹	9.2×10^4		
Sn ¹¹⁹	7.4	Ce ¹⁴³	7.1×10^2		
Sm ¹²¹	3.2	Pr ¹⁴³	7.2×10^4		
Sn ¹²³	8.7	Nd ¹⁴⁷	2.4×10^4		
Sn ¹²⁵	6.1×10^2	Pm ¹⁴⁷	7.7×10^3		

A. Reactivity and Flux Effects

The overall reactivity effect of the subassembly is $-0.0046 \Delta K$. The reactivity effect associated with axial fuel motion has not been investigated. However, it is known that no significant reactivity effects can occur except in the presence of fuel melting.⁽⁶⁾ Since the specimens will normally operate at less than 12 kW/ft, and since melting of mixed oxide fuel has not been observed in this type of fuel pin at less than 22 kW/ft,⁽²⁾ fuel melting can be assumed only by postulating a gross overpower condition or severe transient. Further, the performance of the fuel specimens at a 25% overpower condition (equivalent to a reactor power level of 56.25 MW) has been analyzed. These analyses (Appendices C and D) show fuel specimen and capsule temperatures as calculated to be well within the limits of the EBR-II irradiation test criteria.

B. Flux Perturbation

The flux perturbation in a driver fuel subassembly adjacent to the experimental subassembly was calculated to be 0.988.

C. Misloading of Subassemblies

The desired position of the subassembly (7E4) coincides with the highest flux position for row 7. The desired alternate position (6E3) also corresponds with the highest flux position for row 6.

D. Sodium Bond Defects

Sodium bond discontinuities are not suspected in these test capsules. Previous calculations,⁽¹⁹⁾ however, show that a 3/8 in. x 1/16 in. void in the

annulus of a specimen operating at more than 20 kW/ft would not result in failure of the capsule.

E. Immersion in Water

Flooding of these test subassemblies with water would not result in a critical nuclear condition.⁽²³⁾

F. Consequences of Fuel Specimen Clad Failure

Failure of fuel specimen cladding with attendant release of fission gases to the capsule would result in a hoop stress of 2396 psi in the hottest high burnup specimen (PNL-2-19, 10.2 kW/ft peak, 50,000 MWd/tonne burnup). The thermal stress of 9346 psi, combined with this hoop stress, is still well below the failure point of the capsule wall material on the basis of an unirradiated yield strength of 16,300 psi for 304L stainless steel at 860 °F (460 °C).

G. Fuel Contact with Capsule Wall

Rupture of the fuel specimen cladding might result in the ejection of fuel toward the capsule wall. The sodium present between the fuel and the capsule, by providing a better heat transfer bond, would serve to reduce the fuel temperature. Only a slight change in heat flux at the capsule-coolant interface could be expected unless axial fuel was redistributed.

H. Capsule Leak

Development of a leak at the bottom of the capsule without fuel specimen failure would cause sodium level in the capsule to vary with reactor coolant pressure, as shown in Figure 15.

FIGURE 15. Sodium Level Versus Reactor Coolant Pressure in Capsule with Leak at the Lower End for Hottest Specimen in Subassembly. Position 7E4/1, 45 MW Reactor Power, 1.685 gal/min Flow

I. Nonstandard Materials

No materials other than standard subassembly and capsule structural materials are exposed to the reactor coolant.

J. Consequences of Capsule Failure

In the unlikely event of gross fuel pin and capsule failure:

- 1) The fission product inventory (Table VI) would be released to the reactor coolant.
- 2) Soluble fission products would be leached from the fuel and fuel particles entrained by the sodium in event of a large rupture in both the cladding and capsule walls.
- 3) As previously determined^(13,18-20) from similar experiments, the capsules are isolated sufficiently to virtually exclude the probability of a failure propagating from one capsule to another.
- 4) The capsules are isolated from the reactor by the subassembly hardware as well as from one another. Thus, the probability of a failure propagating to an adjacent subassembly appears extremely unlikely.

K. Consequences of Cleaning Failed Capsule

Consequences of subjecting a radioactive capsule with defective fuel cladding to the cleaning process in the IBC would depend on the rupture. Much or none of the activity, depending on the geometry at the rupture, would be released. A double rupture might possibly escape detection at this point, since the rupture could be closed during the cleaning process after allowing fission gases from the defective fuel

to be released in the reactor. Analysis of the behavior of subassembly X011 should provide further guidance.

L. Behavior During Hypothetical Accident

In accordance with discussions with R. Neidner,⁽²¹⁾ the power transient described as case 4^(23, p. 206) was not used as a basis for evaluating the performance of the test specimens under abnormal conditions. Instead, the steady-state temperatures at a 25% over-power condition (56.25 MW reactor power) were determined. These analyses show that, with uninterrupted coolant flow, the hottest capsule in the subassembly (specimen number 19, capsule position number 1) attains a maximum mean capsule wall temperature of 527 °C (980 °F) with uncertainty factors included (see Appendix C). This maximum is substantially less than the 1105 °F specified in criterion V-A-4 in the Guide.⁽⁴⁾

Calculations by others^(13, 18-20) on the temperature history of mixed oxide fuel specimens, initially operating at substantially higher rod powers than the present specimens under the prescribed power transient, have shown hazards to the reactor from mixed oxide specimens to be no greater than that from the EBR-II driver fuel itself.

M. Behavior During Loss of Flow Accident

The ANL EBR-II Hazards Analysis Code was also used to evaluate fuel specimen behavior during the prescribed loss-of-flow accident.^(para 13, p. 50, 14) Results of these calculations for the hottest specimen in the subassembly are presented in Figure 16.

FIGURE 16. Capsule and Specimen Temperatures During Loss of Flow Accident for Hottest Specimen in Subassembly. Position 7E4/1, 45 MW Initial Reactor Power, 1.685 gal/min Initial Flow Rate

N. Additional Loss-of-Flow Calculations

The temperatures thus calculated indicate no probability of capsule failure and, consequently, no need for further calculations.

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APPENDIX A

REFERENCE DRAWINGS

GENERAL NOTES:

(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: DECIMAL $\pm .005$
ANGULAR $\pm 0^{\circ}30'$
2. ALL MACHINED SURFACES $\sqrt[63]{}$ (ASA B46.1 1962)
3. ALL MATL TO BE AS SPECIFIED

QTY.	PT. NO.	DESCRIPTION	MATERIAL OR DWG
1	1	ASSEMBLY	
1	2	FUEL PIN	H-3-27688
1	3	TOP CAP	
2	4	END PLUG	
1	5	BOTTOM CAP	
1	6	TUBE .375 OD x .235 ID x 36" LG	
2	7	TUBE 1/8 OD x .020 WALL x 7" LG	

GENERAL NOTES

(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: FRACTIONAL $\pm \frac{1}{64}$, DECIMAL $\pm .005$, ANGULAR $\pm 0^{\circ}.30'$
2. ALL MACHINED SURFACES $\sqrt{63}$ (ASA-B46.1-1962)
3. BREAK ALL SHARP EDGES & REMOVE ALL BURRS
4. DO NOT BREAK SHARP EDGES ON WELD LIPS & TUBE ENDS.
5. VOID VOLUME TO BE FILLED WITH HELIUM AT ONE ATMOSPHERE PRESSURE.
6. WELDING TO BE TIG W/NO FILLER ROD USED.
7. ALL MATERIAL TO BE AS SPECIFIED.
8. REFERENCE DIMENSIONS TO BE CONFIRMED BY RADIOGRAPHY.

PNL - EBR-II Capsule Assembly and Details

QTY.	PT. NO.	DESCRIPTION	MATERIAL OR DWG.
	1	ASSEMBLY	
1	2	FUEL PIN	H-3-27801
1	3	TOP CAP	
1	4	END PLUG	
1	5	BOTTOM CAP	
1	6	TUBE $\frac{316}{315}$ OD x $\frac{336}{334}$ ID x 36 LG	
2	7	TUBE $\frac{1}{8}$ OD x .020 WALL x 7 LG	

GENERAL NOTES

(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: FRACTIONAL $\pm \frac{1}{64}$, DECIMAL $\pm .005$
ANGULAR $\pm 0^{\circ}-30'$
2. ALL MACHINED SURFACES $\sqrt{32}$ (ASA-B46-(1-1962)
3. BREAK ALL SHARP EDGES & REMOVE ALL BURRS
4. DO NOT BREAK SHARPEDES ON WELD LIPS & TUBE ENDS.
5. VOID VOLUME TO BE FILLED WITH HELIUM AT ONE ATMOSPHERE PRESSURE
6. WELDING TO BE TIG W/ NO FILLER ROD USED.
7. ALL MATERIAL TO BE AS SPECIFIED
8. REFERENCE DIMENSIONS TO BE CONFIRMED BY RADIOGRAPHY.

POSITIVELY!! DO NOT MAKE ANY PRINTS
FROM THIS TRACING (DWG.# H-5-2780 C)
WITHOUT PRIOR APPROVAL FROM

NAME: ER HAYWARD or JE HANSON or CA BURGESS

PHONE: 3056 3170 3767

H-3-27802

*Reduced Cladding Thickness Specimens
EBR-II Capsule Assembly and Details*

1 2 3 ASSEMBLY
FUEL PIN WITHOUT AXIAL
MOTION RESTRICTOR

4 5 6 ASSEMBLY
FUEL PIN WITH AXIAL
MOTION RESTRICTOR

17 304 SST

7 304 SST

DETAIL 'Z'
(TYP) BOTH ENDS OF
PART #10

16 304 SST

QTY.	PI NO.	DESCRIPTION
1	1	ASSEMBLY
1	2	
1	3	
1	4	
1	5	
1	6	
1	7	PEDESTAL
1	8	TUBE .250 DIA x .016 WALL x 31 3/16 LG 304 SST
1	9	SOLDER TUBE .188 OD x .148 ID x 10 7/16 LG 304 SST
1	10	TUBE .040 OD x .034 ID x 32 27/32 LG 304 SST
1	11	EXTENSOMETER H-3-15274
2	12	INSULATOR PELLET .214 DIA x 1/2 LG DEPLETED UO ₂
53	13	FUEL, SOLID PELLET, 25% PUO ₂ , 75% UO ₂ , 1/4 LG (FULLY ENRICHED)
54	14	FUEL, ANNULAR PELLET, 25% PUO ₂ , 75% UO ₂ , 1/4 LG (FULLY ENRICHED)
1	15	FUEL MOTION RESTRICTOR
1	16	END CAP
1	17	ROD .212 ± .001 DIA x 27 7/8 LG - NICKEL
1	18	FLUX MONITOR WIRE .020 DIA x 28 3/8 LG, PURE COBALT

GENERAL NOTES

UNLESS OTHERWISE SPECIFIED

1. TOL FRAC $\frac{1}{64}$ DEC $\pm .005$ ANG $\pm 0^{\circ} 30'$
2. ALL MACHINED SURFACES $63 \sqrt{1}$ (ASA B46.1 1962)
3. REMOVE ALL BURRS & BREAK ALL SHARP EDGES EXCEPT ON WELD LIPS AND TUBE ENDS
4. ALL MATL TO BE AS SPECIFIED
5. WELDING TO BE TIG WITH NO FILLER ROD USED
6. REFERENCE DIMENSIONS TO BE CONFIRMED BY RADIOGRAPHY

QTY	QTY	PT	NO	
			1	ASSEMBLY
			2	ASSEMBLY
1	1		3	END CAP
1	1		4	PEDESTAL
1	1		5	TUBE .238 DIA x .010 WALL x 31 ¹⁷ / ₃₂ LG SST 304
1	1		6	SPACER TUBE .188 OD x .148 ID x 10 ⁷ / ₃₂ LG SST 304
1	1		7	TUBE .046 OD x .032 ID x 32 ²⁷ / ₆₄ LG SST 304
1	1		8	EXTENSOMETER H-3-15274
2	2		9	INSULATOR PELLET .214 DIA x ¹ / ₂ LG DEPLETED UO ₂
	54		10	FUEL, SOLID PELLET, 25% PUO ₂ , 75% UO ₂ , ¹ / ₄ LG FULLY ENRICHED
54			11	FUEL, ANNULAR PELLET, 25% PUO ₂ , 75% UO ₂ , ¹ / ₄ LG FULLY ENRICHED
1	1		12	ROD ²¹³ / ₂₁₁ DIA x 2 ⁷ / ₃₂ LG NICKEL
1	1		13	FLUX MONITOR WIRE .020 DIA x 29 ¹³ / ₃₂ LG PURE COBALT

GENERAL NOTES

UNLESS OTHERWISE SPECIFIED

1. TOL FRAC $\pm \frac{1}{64}$ DEC $\pm .005$ ANG $\pm 0^\circ 30'$
2. ALL MACHINED SURFACES $63\sqrt{}$ (ASA B46.1-1962)
3. REMOVE ALL BURRS & SHARP EDGES EXCEPT ON WELD LIPS & TUBE ENDS
4. ALL MATL TO BE AS SPECIFIED
5. WELDING TO BE TIG WITH NO FILLER ROD USED
6. REFERENCE DIMS TO BE CONFIRMED BY RADIOGRAPHY

POSITIVELY!! DO NOT MAKE ANY PRINTS
FROM THIS TRACING (DWG.# H-3-27801)
WITHOUT PRIOR APPROVAL FROM

NAME: ER Hayward or JE Hanson or CA Burgess
PHONE: 3056 3170 3767

H-3-27801

Reduced Cladding Thickness Specimens
EBR-II Capsule Fuel Pin Assembly and Details

QTY	PT NO	DESCRIPTION	MATL OR CAT NO
1	1	ASSEMBLY	
1	2	PIN	
1	3	SPRING	
1	4	SLEEVE	
1	5	PIN, $\frac{1}{8}$ DIA $\times$ $\frac{3}{16}$ LG "SPIROL" LIGHT DUTY 302 SST

GENERAL NOTES:

(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: FRACTIONAL $\pm \frac{1}{64}$, DECIMAL $\pm .005$, ANGULAR $\pm 0^{\circ}30'$
2. ALL MACHINED SURFACES $\sqrt{63}$ (ASA B46.1 1962)
3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS.
4. ALL MATL TO BE AS SPECIFIED OR APPROVED EQUAL QUALITY.
5. END OF PART #2 MUST BE IN CONTACT WITH PART #5 AT ASSY.

QTY.	PT. NO.	DESCRIPTION	MATL	REMARKS
X	1	ASSEMBLY		
1	2	SHELL		SH-2
1	3	CONTAINER		SH-3
1	4	CAP		
4	5	TRUNNION		SH-2
1	6	LIFTING BAIL COVER		SH-3
1	7	DISHED HEAD		SH-2
1	8	VENT TUBE		SH-3
1	9	DRAIN TUBE		
1	10	COUPLING		
1	11	COUPLING		
1	12	OUTER SHELL		SH-2
2	13	SPACER		SH-1
2	14	PLUG, PIPE		SH-3
1	15	PLUG 5/8" DIA X 3/8" THK	SST	
1	16	SHIELD (APPROX 25,000 LB)	LEAD	
2	17	CLIP		SH-3
1	18	O-RING (U-21249-11750-5E)	SST	UNITED ARROW BRAND
2	19	O-RING (U-2310 - 01500-5E)	SST	
2	20	RD HD MACH SCR 1/2" X 40UNC-2A	SST	1 LG
4	21	NUT 1/2" X 20UNC-2B HEX	SST	
6	22	NUT 1" X 12UNC-2B HEX	SST	
6	23	WASHER 1" PLAIN	SST	

GENERAL NOTES

ALL PARTS, UNLESS OTHERWISE SPECIFIED, SHALL
CONFORM TO THE FOLLOWING

1. TOLERANCES FRACTIONAL $\pm \frac{1}{32}$
DECIMAL $\pm .005$
ANGULAR $\pm 1^\circ$

2. PARTS TO SHOW GOOD WORKMANSHIP, FREE
FROM BURRS, UNMARRED AND CLEAN.

3. BREAK ALL CORNERS.

4. ALL MACHINED SURFACES $63 \sqrt{\text{in}}$

5. ALL MATERIAL TO BE AS SPECIFIED OR APPROVED
EQUAL QUALITY.

6. CASK TO BE WELDED WATERTIGHT.

7. ALL EXPOSED SURFACES INCLUDING WELDS TO
BE POLISHED.

8. LEAD FILL SHALL BE MADE IN ONE CONTINUOUS
POUR TO ELIMINATE VOIDS. PRE HEAT CASK
BEFORE STARTING LEAD POUR TO ASSURE ALL
THE LEAD IN THE CASK IS MOLTEN; AS THE
LEAD SOLIDIFIES THE PIPING SHALL BE
KEPT FILLED WITH MOLTEN LEAD. WELD IN
SST PLUG (PART #15) AFTER LEAD HAS COOLED.

9. SEQUENCE OF ASSY MACHINING & WELDING
TO BE VENDORS RESPONSIBILITY.

10. ALL MATL TO BE 300 SERIES SST UNLESS
OTHERWISE SPECIFIED.

BNWL-650

APPENDIX B

EBR-II SODIUM BONDING PROCEDURE

APPENDIX B

EBR-II SODIUM BONDING PROCEDURE1.0 Scope

This procedure covers the sodium bonding of clad pins to EBR-II irradiation capsules.

2.0 Procedures

- 2.1 Make up an assembly of three capsules (see Figure B-1) and helium leak check each one.
- 2.2 Purge capsules with helium to exclude air.
- 2.3 Mount the capsules in the bonding furnace so the desired sodium level corresponds with the eddy current coil in the level detection line. Connect the capsules to the sodium fill lines.
- 2.4 Turn on all heaters; set controllers for the sodium supply vessel, and heat tape on piping and sump at 350 °F. Maintain continuous vacuum on sump.
- 2.5 Blow down through the level detection line to the sump with helium to clear the line of sodium.
- 2.6 Blow down through the capsules to the sump with helium to assure the line is clear.
- 2.7 Open all valves required to evacuate the system with the exception of the sodium supply vessel.
- 2.8 Check and adjust the helium pressure in the sodium supply vessel. Pressure should read ~6 psi at operating temperature.
- 2.9 When vacuum reaches 50 microns, switch on bonding furnaces and heat capsules to 1000 °F.

FIGURE B-1. Enclosed Sodium Bonding Loop

- 2.10 Fill capsules with sodium until the level indicator shows the sodium above the desired level.
- 2.11 Maintain for 2 hr the capsule temperature at 1000 °F, the sodium above the desired level, and a vacuum above the sodium.
- 2.12 Drain the sodium from the capsule into the sump and refill the capsule to the desired level with clean sodium from the sodium supply vessel.
- 2.13 After the filling has been completed, pressurize the space above the sodium in the capsule to 15 psi with helium.
- 2.14 Recheck sodium level and adjust if necessary.
- 2.15 Soak capsules for 2 hr at 1000 °F with 15 psi helium above the sodium.
- 2.16 Recheck sodium level periodically during the soak period and adjust as necessary.
- 2.17 Close all valves in the sodium system, and turn off both bonding furnaces and line heaters.
- 2.18 Open the door of the bottom bonding furnace and allow the capsules to cool to room temperature.
- 2.19 Check the sodium level in each capsule with the portable level detector before removing the capsules from the bonding loop.
- 2.20 Release the helium pressure above the sodium in the capsules to atmospheric pressure and remove the capsule from the bonding loop. Reheat the capsules to 1000 °F and directionally cool from the bottom with flowing air.
- 2.21 Eddy current test capsules for voids in the sodium annulus.
- 2.22 Complete the procedure by helium leak testing the capsule.

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APPENDIX C

TYPICAL CALCULATIONS

APPENDIX C - PART I

PREDICTED PERFORMANCE AT STANDARD OPERATING CONDITIONS
(45 MW REACTOR POWER)PERFORMANCE PREDICTIONS FOR SPECIMENS IN CORE POSITION 7E4

Performance predictions for each fuel specimen in the subassembly were made with the aid of the EBR-II Hazards Analysis Code prepared by the late C. R. Reinke, Argonne National Laboratory. A description of the input required for the various parts of the computer program is given in Appendix F as the code was not fully documented prior to Mr. Reinke's death.

Appropriate fissioning rates were determined for each capsule location in the subassembly (Figure 6) and the distance of each capsule from the center of the core. A tabulation of these distances is given in Appendix E.

Part one of the computer program was then employed to determine the heat generation of each type of fuel specimen in each position of the subassembly. It was possible in this manner to arrange the specimen locations within the subassembly to minimize the variations in linear heat generation rate. As illustrated by Figures C-1 and C-2, peak linear heat generation rates range from a maximum of 10.2 kW/ft to a minimum of 9.3 kW/ft.

An excerpt from the Part I computer input/output for the hottest specimen in the subassembly (specimen number 19, capsule position one) is included in this appendix.

Uncertainty factor calculations were made only for the hottest specimen in the subassembly. These data are presented in Figures C-3, C-4, and C-5. Tabulation of the calculations for various flow rates are also included in this appendix.

Stress calculations were made only for the hottest specimen in the subassembly for an average burnup of 50,000 MWd/tonne. Part II of the program (stress calculations) also includes calculations for the variation of sodium level, capsule pressure, and void volume with sodium temperature and the variation

FIGURE C-1. Coolant ΔT Versus Flow for Highest and Lowest Specimen Linear Heat Generation in Subassembly. Position 7E4, 45 MW Reactor Power

FIGURE C-2. Maximum Mean Capsule Wall Temperature Versus Coolant Flow Rate for Highest and Lowest Specimen Linear Heat Generation in Sub-assembly. Position 7E4/1, 45 MW Reactor Power

FIGURE C-3. Maximum Capsule Mean Wall Temperature (with Uncertainty Factors) Versus Flow. Specimen #19, Position 7E4/1, 10.2 kW/ft Maximum at 45 MW Reactor Power

FIGURE C-4. Coolant-Capsule Film ΔT Versus Flow Rate for Specimen 19. Position 7E4/1, 10.2 kW/ft Maximum Linear Heat Generation Rate at 45 MW Reactor Power

FIGURE C-5. *Uncertainty Factor Versus Flow. Specimen #19, Position 7E4/1, 10.2 kW/ft Maximum Linear Heat Generation Rate*

of sodium level with reactor coolant pressure in a capsule with a leak at its lower end. These data are presented in Text Figures 1, 2, 3, and 15, respectively. The computer input/output for the calculations is included in this appendix.

The axial temperature profiles presented in Text Figures 9 and 10 were determined from the Part I calculations, the uncertainty factor calculations, and hand calculations for the fuel center temperature. Hand calculations for fuel center temperatures were necessary since the code does not contain a routine for treating the variation of fuel thermal conductivity with temperature.

The fuel center temperatures were calculated from the expression

$$\int_0^{T_c} k dt = \int_0^{T_s} k dt + \int_{T_s}^{T_c} k dt$$

where

T_s is the fuel surface temperature as determined from the computer calculations, and

$$\int_{T_s}^{T_c} k dt = \frac{Q'}{4\pi}$$

Preceding Figure 8 presents the $\int_0^T k dt$ curve for mixed oxide fuel used for these calculations.

Fuel center temperature reduction attendant with center void formation in the fuel was not considered in these calculations. Thus, the estimated temperatures are only representative of start-of-irradiation conditions.

The temperature histories of the hottest specimen in the sub-assembly under loss-of-reactor coolant condition were calculated under Part IV of the program using power and flow decay input specified in the Guide.⁽¹⁴⁾ These data are presented in Text Figure 16 and Figure C-14.

In lieu of the reactor power transient accident specified in the Guide and in accordance with instructions from R. Neidner,⁽²¹⁾ the steady-state performance of the subassembly at a 25% overpower condition (56.25 MW reactor power) was evaluated. The calculations show that no additional hazards to the reactor are presented by the subassembly under these conditions. The data and sample calculations are given in Part II of this Appendix.

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 1.0 gpm		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)		
		Coolant $\Delta T=147.2$ f d	Film $\Delta T=18.5$ f d	Cap. Wall $1/2\Delta T=17.7$ f d		
45 MW					Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10 14.7	0.10 1.9	0.10 1.8	18.4	338.6
Flow Rate		0.05 7.4			7.4	54.8
Flow Profile		0.05 7.4			7.4	54.8
Mfg. Deviation	Cladding Dimension			0.05 0.9	0.9	0.8
	U, Pu Dimension		0.02 0.4	0.02 0.4	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01 1.5	0.01 0.2	0.01 0.2	1.9	3.6
Capsule Wall Thermal Conductivity				0.10 1.8	1.8	3.2
Film Coefficient			0.02 0.4		0.4	0.2
Power Level		0.02 2.9	0.02 0.4	0.02 0.4	3.7	13.7
Transient Overload		0.05 7.4	0.05 1.0	0.05 0.9	9.3	86.5
					556.8	

$$\text{Uncertainty factor} = \sqrt{556.8} = 23.6$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 342.0 \\ &+ \frac{23.6}{365.6} \text{ } ^\circ\text{C} \\ &= 689.9 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 1.5 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
45 MW		Coolant $\Delta T=98.1$ f d		Film $\Delta T=16.5$ f d		Cap. Wall $1/2 \Delta T=17.7$ f d		Σd $(\Sigma d)^2$	
Neutron + Gamma Flux		0.10	9.8	0.10	1.7	0.10	1.8	13.3	176.9
Flow Rate		0.05	4.9					4.9	24.0
Flow Profile		0.05	4.9					4.9	24.0
Mfg. Deviation	Cladding Dimension					0.05	0.9	0.9	0.8
	U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01	1.0	0.01	0.2	0.01	0.2	1.4	2.0
Capsule Wall Thermal Conductivity						0.10	1.8	1.8	3.2
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	2.0	0.02	0.4	0.02	0.4	2.8	7.8
Transient Overload		0.05	4.9	0.05	0.9	0.05	0.9	6.7	44.9

284.4

$$\text{Uncertainty factor} = \sqrt{284.4} = 16.4$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 491.7 \\ &\quad +16.4 \\ &= 508.1 \text{ } ^\circ\text{C} \\ &= 946.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 2.0 gpm		T_1 Out	T_1 In	T_2 (max)	T_1	$\frac{T_3 - T_2}{2}$ (max value)		
		Coolant $\Delta T = 73.6$		Film $\Delta T = 15.1$		Cap. Wall $1/2 \Delta T = 17.7$		
45 MW		f	d	f	d	f	d	Σd $(\Sigma d)^2$
Neutron + Gamma Flux		0.10	7.4	0.10	1.5	0.10	1.8	10.7 114.5
Flow Rate		0.05	3.7					3.7 13.7
Flow Profile		0.05	3.7					3.7 13.7
Mfg. Deviation	Cladding Dimension					0.05	0.9	0.9 0.8
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7 0.5
	^{235}U ^{239}Pu Dimension	0.01	0.7	0.01	0.2	0.01	0.2	1.1 1.2
Capsule Wall Thermal Conductivity						0.10	1.8	1.8 3.2
Film Coefficient				0.02	0.3			0.3 0.1
Power Level		0.02	1.4	0.02	0.3	0.02	0.4	2.1 4.4
Transient Overload		0.05	3.7	0.05	0.8	0.05	0.9	5.4 29.2

181.3

$$\text{Uncertainty factor} = \sqrt{181.3} = 13.5$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 466.3 \\ &+ 13.5 \\ &= 479.8 \text{ } ^\circ\text{C} \\ &= 895.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 2.5 gpm		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)				
45 MW		Coolant $\Delta T = 58.9$ f d	Film $\Delta T = 14.1$ f d	Cap. Wall $1/2 \Delta T = 17.7$ f d				
						Σd	$(\Sigma d)^2$	
Neutron + Gamma Flux		0.10	5.9	0.10	1.4	0.10	1.8	9.1 82.8
Flow Rate		0.05	2.9					2.9 8.4
Flow Profile		0.05	2.9					2.9 8.4
Mfg. Deviation	Cladding Dimension					0.05	0.9	0.9 0.8
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7 0.5
	^{235}U ^{239}Pu Dimension	0.01	0.6	0.01	0.1	0.01	0.2	0.9 0.8
Capsule Wall Thermal Conductivity						0.10	1.8	1.8 3.2
Film Coefficient				0.02	0.3			0.3 0.1
Power Level		0.02	1.2	0.02	0.3	0.02	0.4	1.9 3.6
Transient Overload		0.05	2.9	0.05	0.7	0.05	0.9	4.5 20.3
		128.9						

$$\text{Uncertainty factor} = \sqrt{128.9} = 11.4$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 450.9 \\ &+ 11.4 \\ &= 462.3 \text{ } ^\circ\text{C} \\ &= 864.1 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 3.0 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
45 MW		Coolant $\Delta T = 49.1$ f d		Film $\Delta T = 13.3$ f d		Cap. Wall $1/2 \Delta T = 17.7$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	4.9	0.10	1.3	0.10	1.8	8.0	64.0
Flow Rate		0.05	2.5					2.5	6.3
Flow Profile		0.05	2.5					2.5	6.3
Mfg. Deviation	Cladding Dimension					0.05	0.9	0.9	0.8
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7	0.5
	^{235}U ^{239}Pu Dimension	0.01	0.5	0.01	0.1	0.01	0.2	0.8	0.6
Capsule Wall Thermal Conductivity						0.10	1.8	1.8	3.2
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	1.0	0.02	0.3	0.02	0.4	1.7	2.9
Transient Overload		0.05	2.5	0.05	0.7	0.05	0.9	4.1	16.8

101.5

$$\text{Uncertainty factor} = \sqrt{101.5} = 10.1$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 440.5 \\ &+ 10.1 \\ &= 450.6 \text{ } ^\circ\text{C} \\ &= 843.1 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 3.5 gpm		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)				
45 MW		Coolant $\Delta T = 42.1$ f d	Film $\Delta T = 12.6$ f d	Cap. Wall $1/2 \Delta T = 17.7$ f d	$\Sigma d \quad (\Sigma d)^2$			
Neutron + Gamma Flux		0.10 4.2	0.10 1.3	0.10 1.8	7.3	53.3		
Flow Rate		0.05 2.1			2.1	4.4		
Flow Profile		0.05 2.1			2.1	4.4		
Mfg. Deviation	Cladding Dimension			0.05 0.9	0.9	0.8		
	U, Pu Dimension		0.02 0.3	0.02 0.4	0.7	0.5		
	^{235}U , ^{239}Pu Dimension	0.01 0.4	0.01 0.1	0.01 0.2	0.7	0.5		
Capsule Wall Thermal Conductivity				0.10 1.8	1.8	3.2		
Film Coefficient			0.02 0.3		0.3	0.1		
Power Level		0.02 0.8	0.02 0.3	0.02 0.4	1.5	2.3		
Transient Overload		0.05 2.1	0.05 0.7	0.05 0.9	3.7	13.7		

83.2

$$\text{Uncertainty factor} = \sqrt{83.2} = 9.1$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 433.1 \\ &+ \frac{9.1}{} \\ &= 442.2 \text{ } ^\circ\text{C} \\ &= 828.0 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 4.0 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
		Coolant $\Delta T = 36.8$		Film $\Delta T = 12.1$		Cap. Wall $1/2 \Delta T = 17.7$			
45 MW		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	3.7	0.10	1.2	0.10	1.8	6.7	44.9
Flow Rate		0.05	1.8					1.8	3.2
Flow Profile		0.05	1.8					1.8	3.2
Mfg. Deviation	Cladding Dimension					0.05	0.9	0.9	0.8
	U, Pu Dimension			0.02	0.2	0.02	0.4	0.6	0.4
	^{235}U ^{239}Pu Dimension	0.01	0.4	0.01	0.1	0.01	0.2	0.7	0.5
Capsule Wall Thermal Conductivity						0.10	1.8	1.8	3.2
Film Coefficient				0.02	0.2			0.2	0.04
Power Level		0.02	0.7	0.02	0.2	0.02	0.4	1.3	1.7
Transient Overload		0.05	1.8	0.05	0.6	0.05	0.9	3.3	10.9

68.84

$$\text{Uncertainty factor} = \sqrt{68.84} = 8.3$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 427.4 \\ &+ 8.3 \\ &= 435.7 \text{ } ^\circ\text{C} \\ &= 816.3 \text{ } ^\circ\text{F} \end{aligned}$$

74E01	.8260+13	.4120+12	.1000+14	Fission Rates (Card 1)												
74E02	.8180+13	.3760+12	.9900+13													
74E03	.8090+13	.3380+12	.9780+13													
74E04	.8180+13	.3760+12	.9900+13													
74E05	.8100+13	.3430+12	.9800+13													
74E06	.8000+13	.3080+12	.9680+13													
74E07	.7900+13	.2710+12	.9550+13													
74E08	.8090+13	.3380+12	.9780+13													
74E09	.8000+13	.3080+12	.9680+13													
74E10	.7910+13	.2740+12	.9570+13													
74E11	.7800+13	.2450+12	.9430+13													
74E12	.7660+13	.2180+12	.9270+13													
74E13	.7900+13	.2710+12	.9550+13													
74E14	.7800+13	.2450+12	.9430+13													
74E15	.7580+13	.2200+12	.9290+13													
74E16	.7540+13	.1950+12	.9120+13													
74E17	.7660+13	.2180+12	.9270+13													
74E18	.7540+13	.1950+12	.9120+13													
74E19	.7590+13	.1750+12	.8940+13													
74E01	3.12	1.13	Gamma Heating (Card 2)													
74E02	2.94	1.00														
74E03	2.76	.86	PART I. Computer Input & Output for Calculation													
74E04	2.94	1.00														
74E05	2.79	.87	of Temperatures in Highest Heat Genera-													
74E06	2.61	.73														
74E07	2.41	.57	tion Specimen in Subassembly.													
74E08	2.76	.86														
74E09	2.61	.73	Position 7E4/1, 45 MW Reactor Power,													
74E10	2.43	.60														
74E11	2.25	.45	1.685 GPM Flow.													
74E12	2.06	.30														
74E13	2.41	.57														
74E14	2.25	.45														
74E15	2.08	.31														
74E16	1.89	.17														
74E17	2.06	.30														
74E18	1.69	.17														
74E19	1.71	.06														
1.00	.93	.81	.70	.63	.55	.50	.44	.37	.30	.18	.07	.00	(Card 3)			
.66	.76	.87	.93	.97	.99	1.00	.99	.98	.95	.88	.77	.68	(Card 4)			
45.0	3/5.	1.685	(Card 5)													

		34.29	34.29	34.29	34.29	34.29	84.06
		.277	.277	.302	.425	.477	
		.554	.327	.554	.604	.851	.954
		.041	13.600	.150	.041	.136	.500-02
15-19		47.96	5.28	15.40	90.39	48.88	
15-19	74E01	T1= 458.4	T2= 468.8	T3= 492.1	T6= 514.7	T7= 532.6	T8= 754.8
15-19	74E01	T1= 451.9	T2= 464.0	T3= 490.9	T6= 516.9	T7= 537.9	T8= 793.9
15-19	74E01	T1= 442.2	T2= 456.0	T3= 486.8	T6= 516.6	T7= 540.2	T8= 833.8
15-19	74E01	T1= 432.5	T2= 447.4	T3= 480.5	T6= 512.5	T7= 537.9	T8= 853.3
15-19	74E01	T1= 426.0	T2= 441.5	T3= 475.7	T6= 508.8	T7= 535.2	T8= 861.8
15-19	74E01	T1= 419.6	T2= 435.4	T3= 470.6	T6= 504.7	T7= 531.7	T8= 867.5
15-19	74E01	T1= 414.7	T2= 430.6	T3= 466.0	T6= 500.3	T7= 527.1	T8= 865.3
15-19	74E01	T1= 409.8	T2= 425.6	T3= 460.9	T6= 495.0	T7= 522.1	T8= 858.1
15-19	74E01	T1= 403.3	T2= 418.9	T3= 453.5	T6= 487.0	T7= 513.6	T8= 843.5
15-19	74E01	T1= 396.9	T2= 412.0	T3= 445.5	T6= 477.9	T7= 503.7	T8= 823.6
15-19	74E01	T1= 387.2	T2= 401.1	T3= 432.1	T6= 462.2	T7= 486.0	T8= 781.9
15-19	74E01	T1= 377.5	T2= 389.8	T3= 417.1	T6= 443.6	T7= 464.6	T8= 725.3
15-19	74E01	T1= 371.0	T2= 381.8	T3= 405.7	T6= 428.9	T7= 447.3	T8= 675.6
15-19	74E01	MAX WATTS/CM= 334.6		MAX KW/FT= 10.2		MAX CAL/SEC-CC= 331.8	
		FILM COEFFICIENT, CAL/SEC-CM2-C= 1.67		MAX CAPSULE HEAT FLUX, WATTS/CM2= 111.9			
		FUEL VOLUME, CM3= 8.26					

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74E01	.8280+13	.4140+12	.1002+14			
=19	47.96	5.28	15.40	90.39	48.88	
.554	.330	.554	.604	.851	.954	.590
36.116	5.080	.850	.064	14.576		
534.4	27.1	18.0	35.4	23.3	733.	
25.00	34.29	93.98	84.06			
.192+08	.192+08	.187-04	.187-04	.32	.32	
45.0	340.	1.685				
.2724	.2620	.2318	1.00			
25. 100. 200. 300. 400. 460. 500. 550.						
.0 .1 .2 .3 .4 .5 .6 .7 .8 .9 1.0						
7E4/1, 45 MW Reactor Power						
1.685 gpm Flow						
21 in. Na Level						
TOTAL FISSIONS= .1027+23 AVERAGE BURNUP,A/O= 5.9 PEAK BURNUP,A/O= 6.7						
TOTAL FISSION GAS GENERATED,CC= 101.20 TOTAL FISSION GAS RELEASED,CC= 101.20						
INITIAL VOID OF PLENUM,CC= 5.05 TOTAL INITIAL VOID,CC= 6.29						
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI= 925.4						
CLAD HOOP STRESS,PSI= 10086.0						
CLAD THERMAL STRESS AT MIDPLANE,PSI= 7154.4 CLAD THERMAL STRESS AT TOP,PSI= 4752.0						
CAPSULE THERMAL STRESS AT MIDPLANE,PSI= 9345.6 CAPSULE THERMAL STRESS AT TOP,PSI= 6151.2						
TEMP,C= 25.	NA VOL,CC= 14.58	NA LEVEL,CM= 53.34	VOID VOL,CC= 13.83	CAP PRESS,PSI= 14.7		
TEMP,C= 100.	NA VOL,CC= 15.23	NA LEVEL,CM= 55.72	VOID VOL,CC= 13.18	CAP PRESS,PSI= 19.3		
TEMP,C= 200.	NA VOL,CC= 15.61	NA LEVEL,CM= 57.13	VOID VOL,CC= 12.80	CAP PRESS,PSI= 25.2		
TEMP,C= 300.	NA VOL,CC= 16.02	NA LEVEL,CM= 58.63	VOID VOL,CC= 12.39	CAP PRESS,PSI= 31.6		
TEMP,C= 400.	NA VOL,CC= 16.46	NA LEVEL,CM= 60.24	VOID VOL,CC= 11.95	CAP PRESS,PSI= 38.4		
TEMP,C= 460.	NA VOL,CC= 16.74	NA LEVEL,CM= 61.26	VOID VOL,CC= 11.67	CAP PRESS,PSI= 42.9		
TEMP,C= 500.	NA VOL,CC= 16.94	NA LEVEL,CM= 61.97	VOID VOL,CC= 11.47	CAP PRESS,PSI= 46.0		
TEMP,C= 550.	NA VOL,CC= 17.18	NA LEVEL,CM= 62.88	VOID VOL,CC= 11.22	CAP PRESS,PSI= 50.0		
CAPSULE PRESSURE AT 4600C,PSI= 42.9 CORRESPONDING CAPSULE HOOP STRESS,PSI= 354.5						
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI= 289.8 CORRESPONDING CAPSULE HOOP STRESS,PSI= 2396.3						
SODIUM LEVEL WITH LEAK IN CAPSULE AT 4600C						
HEIGHT OF SODIUM,CM= .000	CAPSULE PRESSURE,PSI= 17.6					
HEIGHT OF SODIUM,CM= 8.406	CAPSULE PRESSURE,PSI= 19.2					
HEIGHT OF SODIUM,CM= 16.812	CAPSULE PRESSURE,PSI= 21.1					
HEIGHT OF SODIUM,CM= 25.218	CAPSULE PRESSURE,PSI= 23.5					
HEIGHT OF SODIUM,CM= 33.624	CAPSULE PRESSURE,PSI= 26.4					
HEIGHT OF SODIUM,CM= 42.030	CAPSULE PRESSURE,PSI= 30.2					

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	35.2
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	42.2
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	52.8
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	70.3
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	105.4

74E01	.8280+13	.4140+12	.1002+14		
=19	47.96	5.28	15.40	90.39	48.88
.554	.330	.554	.604	.851	.954
.590					
36.116	5.080	.850	.064	15.966	
334.4	27.1	18.0	35.4	23.3	733.
25.00	34.29	93.98	84.06		
.192+08	.192+08	.187-04	.187-04	.32	.32
45.0	340.	1.685			
.2724	.2620	.2318	1.00		
25. 100. 200. 300. 400. 460. 500. 550.					
.0 .1 .2 .3 .4 .5 .6 .7 .8 .9 1.0					

7E4/1, 45 MW Reactor Power
1.685 gpm Flow
23 in. Na Level

TOTAL FISSIONS=	.1027+23	AVERAGE BURNUP,A/O=	5.9	PEAK BURNUP,A/O=	6.7
TOTAL FISSION GAS GENERATED,CC=	101.20	TOTAL FISSION GAS RELEASED,CC=	101.20		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	925.4				
CLAD HOOP STRESS,PSI=	10086.0				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	7154.4	CLAD THERMAL STRESS AT TOP,PSI=	4752.0		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	9345.6	CAPSULE THERMAL STRESS AT TOP,PSI=	6151.2		
TEMP,C= 25.	NA VOL,CC= 15.97	NA LEVEL,CM= 58.42	VOID VOL,CC= 12.44	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 16.68	NA LEVEL,CM= 61.03	VOID VOL,CC= 11.73	CAP PRESS,PSI= 19.5	
TEMP,C= 200.	NA VOL,CC= 17.10	NA LEVEL,CM= 62.58	VOID VOL,CC= 11.31	CAP PRESS,PSI= 25.7	
TEMP,C= 300.	NA VOL,CC= 17.55	NA LEVEL,CM= 64.22	VOID VOL,CC= 10.86	CAP PRESS,PSI= 32.4	
TEMP,C= 400.	NA VOL,CC= 18.03	NA LEVEL,CM= 65.99	VOID VOL,CC= 10.38	CAP PRESS,PSI= 39.8	
TEMP,C= 460.	NA VOL,CC= 18.34	NA LEVEL,CM= 67.10	VOID VOL,CC= 10.07	CAP PRESS,PSI= 44.7	
TEMP,C= 500.	NA VOL,CC= 18.55	NA LEVEL,CM= 67.88	VOID VOL,CC= 9.86	CAP PRESS,PSI= 48.1	
TEMP,C= 550.	NA VOL,CC= 18.82	NA LEVEL,CM= 68.87	VOID VOL,CC= 9.59	CAP PRESS,PSI= 52.7	
CAPSULE PRESSURE AT 4600C,PSI=	44.7	CORRESPONDING CAPSULE HOOP STRESS,PSI=	369.5		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	317.5	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2625.4		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 4600C					
HEIGHT OF SODIUM,CM= .000	CAPSULE PRESSURE,PSI=	15.8			
HEIGHT OF SODIUM,CM= 8.406	CAPSULE PRESSURE,PSI=	17.3			
HEIGHT OF SODIUM,CM= 16.812	CAPSULE PRESSURE,PSI=	19.0			
HEIGHT OF SODIUM,CM= 25.218	CAPSULE PRESSURE,PSI=	21.1			
HEIGHT OF SODIUM,CM= 33.624	CAPSULE PRESSURE,PSI=	23.8			
HEIGHT OF SODIUM,CM= 42.030	CAPSULE PRESSURE,PSI=	27.1			

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	31.7
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	38.0
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	47.5
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	63.3
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	94.8

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APPENDIX C - PART II

PREDICTED PERFORMANCE AT 25% OVERPOWER CONDITIONS
(56.25 MW REACTOR POWER)

TABLE C-I. *Specimen Heat Generation Rates and Temperature at 56.25 MW Reactor Power (25% overpower from normal 45 MW reactor power)**

Position Number*	Specimen Number	Maximum Heat Generation Rate, kW/ft	Maximum** Mean Capsule Wall Temp, °C	Maximum Capsule Surface Heat Flux W/cm ²	Coolant** ΔT, C°
1	PNL 19	12.7	508 [†]	139.8	109
2	7	12.6	507	138.2	109
3	8	12.4	505	136.4	107
4	9	12.6	507	138.2	109
5	10	12.5	505	136.6	107
6	15	12.3	503	134.6	105
7	16	12.1	501	132.6	104
8	11	12.4	505	136.4	107
9	12	12.3	503	134.6	105
10	17	12.1	501	132.8	104
11	1	12.3	503	135.3	105
12	2	12.1	501	132.6	104
13	18	12.1	501	132.6	104
14	4	12.3	503	135.3	105
15	5	12.1	501	132.9	104
16	13	11.9	499	130.2	103
17	3	12.1	501	132.6	104
18	6	11.9	499	130.2	103
19	14	11.6	496	127.4	99
		Σ = 232.4		Σ = 1997	

$$\text{Total heat generation} = \frac{232.4}{1.14} \cdot \frac{13.5}{13} = 229.3 \text{ kW};$$

$$\text{Mean coolant } \Delta T = \frac{1997}{19} = 105.1 \text{ C}^\circ$$

$$\text{Mean coolant exit temp} = 37 + 105 = 476 \text{ }^\circ\text{C} \text{ (889 }^\circ\text{F)}$$

* Core position 7E4

** Based on coolant flow rate of 1.69 gpm (32 gpm total)

† Uncertainty factor of 19.1 C° increases this to 527 °C

FIGURE C-6. Coolant ΔT Versus Flow for Highest and Lowest Specimen Linear Heat Generation in Subassembly. Position 7E4, 56.25 MW Reactor Power

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FIGURE C-7. Maximum Mean Capsule Wall Temperature Versus Flow for Highest and Lowest Specimen Heat Generation in Subassembly. Position 7E4, 56.25 MW Reactor Power

FIGURE C-8. Coolant-Capsule Film ΔT Versus Flow for Highest Heat Generation Specimen in Subassembly. Position 7E4/1, 56.25 MW Reactor Power (12.7 kW/ft peak)

FIGURE C-9. *Uncertainty Factor Versus Flow for Highest Heat Generation Specimen in Subassembly. Position 7E4/1, 56.25 MW Reactor Power (12.7 kW/ft peak)*

FIGURE C-10. Maximum Mean Capsule Wall Temperature with Uncertainty Factors Versus Flow for Highest Heat Generation Specimen in Subassembly. Position 7E4/1, 56.25 MW Reactor Power (12.7 kW/ft peak)

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UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 1.0 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
56.25 MW		Coolant $\Delta T = 184.0$ f d		Film $\Delta T = 23.1$ f d		Cap. Wall $1/2 \Delta T = 22.1$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	18.4	0.10	2.3	0.10	2.2	22.9	524.4
Flow Rate		0.05	9.2					9.2	84.6
Flow Profile		0.05	9.2					9.2	84.6
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.5	0.02	0.4	0.9	0.8
	^{235}U ^{239}Pu Dimension	0.01	1.8	0.01	0.2	0.01	0.2	2.2	4.8
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.5			0.5	0.3
Power Level		0.02	3.6	0.02	0.5	0.02	0.4	4.5	20.2
Transient Overload		0.05	9.2	0.05	1.2	0.05	1.1	11.5	132.3

858.0

$$\text{Uncertainty factor} = \sqrt{858.0} = 29.3$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 584.8 \\ &+ 29.3 \\ &= 614.1^\circ \text{C} \\ &= 1137.4^\circ \text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 1.5 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
56.25 MW		Coolant $\Delta T = 122.7$ f d		Film $\Delta T = 20.7$ f d		Cap. Wall $1/2 \Delta T = 22.1$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	12.3	0.10	2.1	0.10	2.1	16.6	275.6
Flow Rate		0.05	6.2					6.2	38.4
Flow Profile		0.05	6.2					6.2	38.4
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U , ^{239}Pu Dimension	0.01	1.2	0.01	0.2	0.01	0.2	1.6	2.6
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	2.4	0.02	0.3	0.02	0.4	3.2	10.2
Transient Overload		0.05	6.2	0.05	1.0	0.05	1.1	8.3	68.9

440.9

$$\text{Uncertainty factor} = \sqrt{440.9} = 21.0$$

$$\begin{aligned} \text{Maximum Capsule mean wall temperature} &= 521.8 \\ &+ 21.0 \\ &= 542.8^\circ \text{C} \\ &= 1009.0^\circ \text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 2.0 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
56.25 MW		Coolant $\Delta T = 92.0$ f d		Film $\Delta T = 18.9$ f d		Cap. Wall $1/2 \Delta T = 22.1$ f d		Σd $(\Sigma d)^2$	
Neutron + Gamma Flux		0.10	9.2	0.10	1.9	0.10	2.2	13.3	176.9
Flow Rate		0.05	4.6					4.6	21.2
Flow Profile		0.05	4.6					4.6	21.2
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01	0.9	0.01	0.2	0.01	0.2	1.3	1.7
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	1.8	0.02	0.4	0.02	0.4	2.6	6.8
Transient Overload		0.05	4.6	0.05	0.9	0.05	1.1	6.6	43.6

278.2

$$\text{Uncertainty factor} = \sqrt{278.2} = 16.7$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 490.1 \\ &+ 16.7 \\ &= 506.8 \text{ } ^\circ\text{C} \\ &= 944.2 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 2.5 gpm		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
56.25 MW		Coolant $\Delta T = 73.6$ f d		Film $\Delta T = 17.6$ f d		Cap. Wall $1/2 \Delta T = 22.1$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	7.4	0.10	1.8	0.10	2.2	11.4	130.0
Flow Rate		0.05	3.7					3.7	13.7
Flow Profile		0.05	3.7					3.7	13.7
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U , ^{239}Pu Dimension	0.01	0.7	0.01	0.2	0.01	0.2	1.1	1.2
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.4		0.4	0.8	0.6
Power Level		0.02	1.5	0.02	0.4	0.02	0.4	2.3	5.3
Transient Overload		0.05	3.7	0.05	0.9	0.05	1.1	5.7	32.5
									203.6

$$\text{Uncertainty factor} = \sqrt{203.6} = 14.3$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 471.2 \\ &+ 14.3 \\ &= 485.5 \text{ } ^\circ\text{C} \\ &= 905.9 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

7E4 1-19 3.0 gpm		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)		
		Coolant $\Delta T = 61.3$ f d	Film $\Delta T = 16.6$ f d	Cap. Wall $1/2 \Delta T = 22.1$ f d	Σd	$(\Sigma d)^2$
56.25 MW						
Neutron + Gamma Flux		0.10 6.1	0.10 1.7	0.10 2.2	10.0	100.0
Flow Rate		0.05 3.1			3.1	9.6
Flow Profile		0.05 3.1			3.1	9.6
Mfg. Deviation	Cladding Dimension			0.05 1.1	1.1	1.2
	U, Pu Dimension		0.02 0.3	0.02 0.4	0.7	0.5
	^{235}U ^{239}Pu Dimension	0.01 0.6	0.01 0.2	0.01 0.2	1.0	1.0
Capsule Wall Thermal Conductivity				0.10 2.2	2.2	4.8
Film Coefficient			0.02			
Power Level		0.02 1.2	0.02 0.3	0.02 0.4	1.9	3.6
Transient Overload		0.05 3.1	0.05 0.8	0.05 1.1	5.0	25.0
					155.4	

$$\text{Uncertainty factor} = \sqrt{155.4} = 12.5$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 458.7 \\ &+ 12.5 \\ &= 471.2 \text{ } ^\circ\text{C} \\ &= 880.2 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
7E4 1-19 3.5 gpm		Out	In	(max)		(max value)			
		Coolant		Film		Cap. Wall			
		$\Delta T = 52.6$		$\Delta T = 15.6$		$1/2 \Delta T = 22.1$			
56.25 MW		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	5.3	0.10	1.6	0.10	2.2	9.1	82.8
Flow Rate		0.05	2.6					2.6	6.8
Flow Profile		0.05	2.6					2.6	6.8
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7	0.5
	$^{235}U, ^{239}Pu$ Dimension	0.01	0.5	0.01	0.2	0.01	0.2	0.9	0.8
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	1.0	0.02	0.3	0.02	0.4	1.7	2.9
Transient Overload		0.05	2.6	0.05	0.8	0.05	1.1	4.5	20.3
		127.0							

$$\text{Uncertainty factor} = \sqrt{127.0} = 11.3$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 450.0 \\ &+ 11.3 \\ &= 461.3 \text{ } ^\circ\text{C} \\ &= 862.3 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
7E4 1-19 4.0 gpm		Coolant $\Delta T = 46.0$		Film $\Delta T = 15.1$		Cap. Wall $1/2 \Delta T = 22.1$			
56.25 MW		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	4.6	0.10	1.5	0.10	2.2	8.3	68.9
Flow Rate		0.05	2.3					2.3	5.3
Flow Profile		0.05	2.3					2.3	5.3
Mfg. Deviation	Cladding Dimension					0.05	1.1	1.1	1.2
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7	0.5
	^{235}U ^{239}Pu Dimension	0.01	0.5	0.01	0.2	0.01	0.2	0.9	0.8
Capsule Wall Thermal Conductivity						0.10	2.2	2.2	4.8
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	0.9	0.02	0.3	0.02	0.4	1.6	2.6
Transient Overload		0.05	2.3	0.05	0.8	0.05	1.1	4.2	17.6
		107.1							

$$\text{Uncertainty factor} = \sqrt{107.1} = 10.4$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 443.5 \\ &+ 10.4 \\ &= 453.9 \text{ } ^\circ\text{C} \\ &= 849.0 \text{ } ^\circ\text{F} \end{aligned}$$

74E01	.8260+13	.4120+12	.1000+14	Fission Rates (Card 1)									
74E02	.8180+13	.3760+12	.9900+13										
74E03	.8090+13	.3380+12	.9780+13										
74E04	.8180+13	.3760+12	.9900+13										
74E05	.8100+13	.3430+12	.9800+13										
74E06	.8000+13	.3080+12	.9680+13										
74E07	.7900+13	.2710+12	.9550+13										
74E08	.8090+13	.3380+12	.9780+13										
74E09	.8000+13	.3080+12	.9680+13										
74E10	.7910+13	.2740+12	.9570+13										
74E11	.7800+13	.2450+12	.9430+13										
74E12	.7660+13	.2180+12	.9270+13										
74E13	.7900+13	.2710+12	.9550+13										
74E14	.7800+13	.2450+12	.9430+13										
74E15	.7680+13	.2200+12	.9290+13										
74E16	.7540+13	.1950+12	.9120+13										
74E17	.7550+13	.2180+12	.9270+13										
74E18	.7540+13	.1950+12	.9120+13										
74E19	.7390+13	.1750+12	.8940+13										
74E01	3.91	1.42	Gamma Heating (Card 2)										
74E02	3.68	1.25											
74E03	3.45	1.07	PART I. Computer Input & Output for Calculation										
74E04	3.68	1.25											
74E05	3.48	1.09	of Temperatures in Highest Heat Genera-										
74E06	3.26	.92											
74E07	3.02	.72	tion Specimen in Subassembly Position										
74E08	3.45	1.07											
74E09	3.26	.92	7E4/1, 56.25 MW Reactor Power, 1.685										
74E10	3.04	.75											
74E11	2.82	.56	GPM.										
74E12	2.57	.37											
74E13	3.02	.72											
74E14	2.82	.56											
74E15	2.60	.39											
74E16	2.37	.22											
74E17	2.57	.37											
74E18	2.37	.22											
74E19	2.14	.08											
1.90	.93	.81	.70	.63	.56	.50	.44	.37	.30	.18	.07	.00	(Card 3)
.66	.76	.87	.93	.97	.99	1.00	.99	.98	.95	.88	.77	.68	(Card 4)
56.3	375.	1.385	(Card 5)										

	34.29	34.29	34.29	34.29	34.29	84.06	(Card 6)
	.277	.277	.302	.425	.477		(Card 7)
	.554	.327	.554	.604	.851	.954	(Card 8)
	.041	13.600	.150	.041	.136	.500-02	(Card 9)
15-19	47.96	5.28	15.40	90.39	48.88	(Card 10).	
15-19	74E01	T1= 480.2	T2= 493.3	T3= 522.4	T6= 550.6	T7= 573.0	T8= 850.8
15-19	74E01	T1= 472.1	T2= 487.3	T3= 520.8	T6= 553.4	T7= 579.2	T8= 899.7
15-19	74E01	T1= 460.0	T2= 477.3	T3= 515.8	T6= 553.0	T7= 582.6	T8= 949.5
15-19	74E01	T1= 447.9	T2= 466.5	T3= 507.8	T6= 547.9	T7= 579.7	T8= 974.0
15-19	74E01	T1= 439.8	T2= 459.1	T3= 501.9	T6= 543.3	T7= 576.2	T8= 984.5
15-19	74E01	T1= 431.7	T2= 451.5	T3= 495.5	T6= 538.1	T7= 572.0	T8= 991.6
15-19	74E01	T1= 425.6	T2= 445.5	T3= 489.8	T6= 532.6	T7= 566.7	T8= 988.9
15-19	74E01	T1= 419.5	T2= 439.3	T3= 483.4	T6= 526.0	T7= 559.9	T8= 980.0
15-19	74E01	T1= 411.4	T2= 430.9	T3= 474.1	T6= 516.0	T7= 549.2	T8= 961.7
15-19	74E01	T1= 403.3	T2= 422.2	T3= 464.1	T6= 504.7	T7= 536.9	T8= 936.8
15-19	74E01	T1= 391.2	T2= 408.7	T3= 447.4	T6= 485.0	T7= 514.8	T8= 884.6
15-19	74E01	T1= 379.1	T2= 394.5	T3= 428.6	T6= 461.7	T7= 488.0	T8= 813.9
15-19	74E01	T1= 371.0	T2= 384.5	T3= 414.4	T6= 443.4	T7= 466.4	T8= 751.8
15-19	74E01	MAX WATTS/CM= 418.3		MAX KW/FT= 12.7		MAX CAL/SEC-CC= 414.8	
		FILM COEFFICIENT, CAL/SEC-CM2-C= 1.67			MAX CAPSULE HEAT FLUX, WATTS/CM2= 139.8		
		FUEL VOLUME, CM3= 8.26					

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FIGURE C-11. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 7E4/1, 56.25 MW Reactor Power, 12.7 kW/ft (peak)*

FIGURE C-12. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 7E3/1, 56.25 MW Reactor Power, 12.7 kW/ft (peak)*

74E01	.8280+13	.4140+12	.1002+14		
19	47.96	5.28	15.40	90.39	48.88
.554	.330	.554	.604	.851	.954
.590					
36.116	5.080	.850	.064	14.576	
575.2	34.0	22.4	44.2	29.1	755.
25.00	34.29	93.98	84.06		
.192+08	.192+08	.187-04	.187-04	.32	.32
56.3	273.	1.685			
.2724	.2620	.2318	1.00		
25.	100.	200.	300.	400.	482.
.0	.1	.2	.3	.4	.5
				.6	.7
				.8	.9
				1.0	

7E4/1, 56.25 MW Reactor Power
1.685 gpm Flow
21 in. Na Level

TOTAL FISSIONS=	.1031+23	AVERAGE BURNUP,A/O=	5.9	PEAK BURNUP,A/O=	6.7
TOTAL FISSION GAS GENERATED,CC=	101.57	TOTAL FISSION GAS RELEASED,CC=	101.57		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	975.5				
CLAD HOOP STRESS,PSI=	10632.3				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	8976.0	CLAD THERMAL STRESS AT TOP,PSI=	5913.6		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	11668.8	CAPSULE THERMAL STRESS AT TOP,PSI=	7682.4		
TEMP,C= 25.	NA VOL,CC= 14.58	NA LEVEL,CM= 53.34	VOID VOL,CC= 13.83	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 15.23	NA LEVEL,CM= 55.72	VOID VOL,CC= 13.18	CAP PRESS,PSI= 19.3	
TEMP,C= 200.	NA VOL,CC= 15.61	NA LEVEL,CM= 57.13	VOID VOL,CC= 12.80	CAP PRESS,PSI= 25.2	
TEMP,C= 300.	NA VOL,CC= 16.02	NA LEVEL,CM= 58.63	VOID VOL,CC= 12.39	CAP PRESS,PSI= 31.6	
TEMP,C= 400.	NA VOL,CC= 16.46	NA LEVEL,CM= 60.24	VOID VOL,CC= 11.95	CAP PRESS,PSI= 38.4	
TEMP,C= 482.	NA VOL,CC= 16.85	NA LEVEL,CM= 61.65	VOID VOL,CC= 11.56	CAP PRESS,PSI= 44.6	
TEMP,C= 500.	NA VOL,CC= 16.94	NA LEVEL,CM= 61.97	VOID VOL,CC= 11.47	CAP PRESS,PSI= 46.0	
TEMP,C= 550.	NA VOL,CC= 17.18	NA LEVEL,CM= 62.88	VOID VOL,CC= 11.22	CAP PRESS,PSI= 50.0	
CAPSULE PRESSURE AT 482C,PSI=	44.6	CORRESPONDING CAPSULE HOOP STRESS,PSI=	368.7		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	299.5	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2476.8		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 482C					
HEIGHT OF SODIUM,CM=	.000	CAPSULE PRESSURE,PSI=	18.1		
HEIGHT OF SODIUM,CM=	8.406	CAPSULE PRESSURE,PSI=	19.8		
HEIGHT OF SODIUM,CM=	16.812	CAPSULE PRESSURE,PSI=	21.8		
HEIGHT OF SODIUM,CM=	25.218	CAPSULE PRESSURE,PSI=	24.2		
HEIGHT OF SODIUM,CM=	33.624	CAPSULE PRESSURE,PSI=	27.2		
HEIGHT OF SODIUM,CM=	42.030	CAPSULE PRESSURE,PSI=	31.1		

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	36.3
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	43.5
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	54.4
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	72.5
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	108.6

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74E01	.8280+13	.4140+12	.1002+14
=19	47.96	5.28	15.40
.554	.330	.554	.604
.851	.954	.590	
36.116	5.080	.850	.064
15.966			
575.2	34.0	22.4	44.2
29.1	755.		
25.00	34.29	93.98	84.06
.192+08	.192+08	.187-04	.187-04
.32	.32		
56.3	273.	1.685	
.2724	.2620	.2318	1.00
25.	100.	200.	300.
400.	482.	500.	550.
.0	.1	.2	.3
.4	.5	.6	.7
.8	.9	1.0	

7E4/1, 56.25 MW Reactor Power
1.685 gpm Flow
23 in. Na Level

TOTAL FISSIONS=	.1031+23	AVERAGE BURNUP,A/O=	5.9	PEAK BURNUP,A/O=	6.7
TOTAL FISSION GAS GENERATED,CC=	101.57	TOTAL FISSION GAS RELEASED,CC=	101.57		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	975.5				
CLAD HOOP STRESS,PSI=	10632.3				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	8976.0	CLAD THERMAL STRESS AT TOP,PSI=	5913.6		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	11668.8	CAPSULE THERMAL STRESS AT TOP,PSI=	7682.4		
TEMP,C= 25.	NA VOL,CC= 15.97	NA LEVEL,CM= 58.42	VOID VOL,CC= 12.44	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 16.68	NA LEVEL,CM= 61.03	VOID VOL,CC= 11.73	CAP PRESS,PSI= 19.5	
TEMP,C= 200.	NA VOL,CC= 17.10	NA LEVEL,CM= 62.58	VOID VOL,CC= 11.31	CAP PRESS,PSI= 25.7	
TEMP,C= 300.	NA VOL,CC= 17.55	NA LEVEL,CM= 64.22	VOID VOL,CC= 10.86	CAP PRESS,PSI= 32.4	
TEMP,C= 400.	NA VOL,CC= 18.03	NA LEVEL,CM= 65.99	VOID VOL,CC= 10.38	CAP PRESS,PSI= 39.8	
TEMP,C= 482.	NA VOL,CC= 18.46	NA LEVEL,CM= 67.53	VOID VOL,CC= 9.95	CAP PRESS,PSI= 46.6	
TEMP,C= 500.	NA VOL,CC= 18.55	NA LEVEL,CM= 67.88	VOID VOL,CC= 9.86	CAP PRESS,PSI= 48.1	
TEMP,C= 550.	NA VOL,CC= 18.82	NA LEVEL,CM= 68.87	VOID VOL,CC= 9.59	CAP PRESS,PSI= 52.7	
CAPSULE PRESSURE AT 482C,PSI=	46.6	CORRESPONDING CAPSULE HOOP STRESS,PSI=	385.2		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	328.2	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2713.7		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 482C					
HEIGHT OF SODIUM,CM=	.000	CAPSULE PRESSURE,PSI=	16.3		
HEIGHT OF SODIUM,CM=	8.408	CAPSULE PRESSURE,PSI=	17.8		
HEIGHT OF SODIUM,CM=	16.812	CAPSULE PRESSURE,PSI=	19.6		
HEIGHT OF SODIUM,CM=	25.218	CAPSULE PRESSURE,PSI=	21.8		
HEIGHT OF SODIUM,CM=	33.624	CAPSULE PRESSURE,PSI=	24.5		
HEIGHT OF SODIUM,CM=	42.030	CAPSULE PRESSURE,PSI=	28.0		

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	32.6
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	39.1
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	48.9
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	65.2
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	97.7

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FIGURE C-13. Sodium Level Versus Sodium Temperature in Hottest Specimen in Sub-assembly. Position 7E4/1, 56.25 MW Reactor Power, 1.685 gal/min Flow

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FIGURE C-14. Internal Void Volume as a Function of Capsule Temperature.
Position 7E4/1, 56.25 MW Reactor Power, 1.685 gal/min Flow

FIGURE C-15. Internal Capsule Pressure as a Function of Capsule Temperature.
Position 7E4/1, 56.25 MW Reactor Power, 1.685 gal/min Flow

FIGURE C-16. Sodium Level Versus Reactor Coolant Pressure in Capsule with Leak at the Lower End for Hottest Specimen in Subassembly. Position 7E4/1, 56.25 MW Reactor Power, 1.685 gal/min Flow

FIGURE C-17. Capsule and Specimen Temperatures During Loss of Flow Accident for Hottest Specimen in Subassembly. Position 7E4/1, 56.25 MW Initial Reactor Power, 1.685 gal/min Initial Flow Rate

APPENDIX D

FUEL SPECIMEN AND SUBASSEMBLY PERFORMANCE DATA
FOR POSITION 6E3

APPENDIX D - PART I

PREDICTED PERFORMANCE AT STANDARD OPERATING CONDITIONS
(45 MW REACTOR POWER)PERFORMANCE PREDICTIONS FOR SPECIMENS IN CORE POSITION 6E3

These calculations evaluate the performance of the subassembly in core position 6E3. This position is an alternate to the desired 7E4 position in the event that the 7E4 position, for some reason, is determined unsuitable for the irradiation.

The calculational techniques employed to evaluate subassembly performance in the 6E3 position were the same as those discussed in Appendix C. The data in this appendix are presented in the same manner as in Appendix C.

TABLE D-I. Specimen Heat Generation Rates and Temperatures*
at 45 MW Reactor Power

Position Number	Specimen Number	Maximum Heat Generation Rate, kW/ft	Maximum** Mean Capsule Wall Temp, °C	Maximum Capsule Surface Heat Flux, W/cm ²	Coolant** ΔT, C°
1	19	11.2	476 [†]	123	81
2	7	11.0	474	120.6	80
3	8	10.8	472	118.3	78
4	9	11.1	475	121.5	80
5	10	10.9	473	119.3	79
6	15	10.7	471	117.1	77
7	16	10.5	469	115.2	76
8	11	11.0	474	120.6	80
9	12	10.7	471	117.7	77
10	17	10.6	470	115.9	76
11	1	10.8	472	118.8	78
12	2	10.7	471	117.7	77
13	18	10.6	470	116.2	76
14	4	10.9	473	119.1	79
15	5	10.8	472	118.0	78
16	13	10.6	470	116.8	76
17	3	10.8	472	117.9	78
18	6	10.7	471	117.1	77
19	14	10.6	470	115.9	76
		Σ = 205.0		Σ = 1479	

$$\text{Total heat generation} = \frac{205.0}{1.14} \times \frac{13.5}{12} = 202.0 \text{ kW};$$

$$\text{Mean coolant } \Delta T = \frac{1479}{19} = 78.9 \text{ C}^\circ$$

* Core position 6E3

** Based on coolant flow rate of 2 gpm (total flow = 38 gpm)

† Uncertainty factor of 15 C° increases this to 491 °C.

FIGURE D-1. Coolant ΔT Versus Flow for Highest and Lowest Specimen Linear Heat Generation in Subassembly. Position 6E3, 45 MW Reactor Power

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FIGURE D-2. Maximum Mean Capsule Wall Temperature Versus Flow for Highest and Lowest Specimen Heat Generation in Subassembly. Position 6E3, 45 MW Reactor Power

FIGURE D-3. *Maximum Mean Capsule Wall Temperature with Uncertainty Factors Versus Flow for Highest Heat Generation Specimen in Subassembly. Position 6E3/1, 11.2 kW/ft (peak), 45 MW Reactor Power*

FIGURE D-4. Coolant-Capsule Film ΔT for Highest Specimen Linear Heat Generation in Subassembly. Position 6E3/1, 45 MW Reactor Power, 11.2 kW/ft (peak)

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FIGURE D-5. *Uncertainty Factor Versus Flow for Highest Heat Generation Specimen in Subassembly. Position 6E3/1, 11.2 kW/ft (peak), 45 MW Reactor Power*

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
6E3 1-19 45 MW		Coolant		Film		Cap. Wall			
1.0 gpm		$\Delta T = 161.8$ f d		$\Delta T = 20.3$ f d		$1/2 \Delta T = 19.5$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	16.2	0.10	2.0	0.10	2.0	20.2	408.0
Flow Rate		0.05	8.1					8.1	65.6
Flow Profile		0.05	8.1					8.1	65.6
Mfg. Deviation	Cladding Dimension					0.05	1.0	1.0	1.0
	U_2 Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01	1.6	0.01	0.2	0.01	0.2	2.0	4.0
Capsule Wall Thermal Conductivity						0.10	2.0	2.0	4.0
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	3.2	0.02	0.4	0.02	0.4	4.0	16.0
Transient Overload		0.05	8.1	0.05	1.0	0.05	1.0	10.1	102.0

667.0

$$\text{Uncertainty factor} = \sqrt{667.0} = 25.8 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 559.0 \\ &+ 25.8 \\ &\underline{584.8 \text{ } ^\circ\text{C}} \\ &= 1084.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

6E3 1-19 45 MW		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)		
1.5 gpm		Coolant $\Delta T = 107.9$ f d	Film $\Delta T = 18.2$ f d	Cap. Wall $1/2 \Delta T = 19.5$ f d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10 10.8	0.10 1.8	0.10 2.0	14.6	213.2
Flow Rate		0.05 5.4			5.4	29.2
Flow Profile		0.05 5.4			5.4	29.2
Mfg. Deviation	Cladding Dimension			0.05 1.0	1.0	1.0
	U, Pu Dimension		0.02 0.4	0.02 0.4	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01 1.1	0.01 0.2	0.01 0.2	1.5	2.3
Capsule Wall Thermal Conductivity				0.10 2.0	2.0	4.0
Film Coefficient			0.02 0.4		0.4	0.2
Power Level		0.02 2.1	0.02 0.4	0.02 0.4	2.9	88.4
Transient Overload		0.05 5.4	0.05 0.9	0.05 1.0	7.3	53.3

341.4

$$\text{Uncertainty factor} = \sqrt{341.4} = 18.5 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 503.6 \\ &+ 18.5 \\ &= 522.1 \text{ } ^\circ\text{C} \\ &= 971.8 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
6E3 1-19 45 MW		Out	In	(max)		(max value)			
2.0 gpm		Coolant $\Delta T=80.9$		Film $\Delta T=16.6$		Cap. Wall $1/2\Delta T=19.5$			
		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	8.1	0.10	0.7	0.10	2.0	11.8	139.2
Flow Rate		0.05	4.0					4.0	16.0
Flow Profile		0.05	4.0					4.0	16.0
Mfg. Deviation	Cladding Dimension					0.05	1.0	1.0	1.0
	U, Pu Dimension			0.02	0.4	0.02	0.4	0.8	0.6
	^{235}U , ^{239}Pu Dimension	0.01	0.8	0.01	0.2	0.01	0.2	1.2	1.4
Capsule Wall Thermal Conductivity						0.10	2.0	2.0	4.0
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	0.2	0.02	0.2	0.02	0.4	0.8	0.6
Transient Overload		0.05	4.0	0.05	0.8	0.05	1.0	5.8	33.6
		212.6							

$$\text{Uncertainty factor} = \sqrt{212.6} = 14.6 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 475.7 \\ &+ 14.6 \\ &= 490.3 \text{ } ^\circ\text{C} \\ &= 914.5 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

6E3 1-19 45 MW		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (max value)		
2.5 gpm		Coolant $\Delta T = 65.0$ f d	Film $\Delta T = 15.5$ f d	Cap. Wall $1/2 \Delta T = 19.5$ f d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10 6.5	0.10 1.6	0.10 2.0	10.1	102.0
Flow Rate		0.05 3.3			3.3	10.9
Flow Profile		0.05 3.3			3.3	10.9
Mfg. Deviation	Cladding Dimension			0.05 1.0	1.0	1.0
	U, Pu Dimension		0.02 0.3	0.02 0.4	0.7	0.5
	^{235}U ^{239}Pu Dimension	0.01 0.6	0.01 0.2	0.01 0.2	1.0	1.0
Capsule Wall Thermal Conductivity				0.10 2.0	2.0	4.0
Film Coefficient			0.02 0.3		0.3	0.1
Power Level		0.02 1.2	0.02 0.3	0.02 0.4	1.9	3.6
Transient Overload		0.05 3.3	0.05 0.8	0.05 1.0	5.1	26.0

160.0

Uncertainty factor = 160.0 = 12.6 °C

Maximum capsule mean wall temperature = 458.7
+ 12.6
= 471.3 °C
= 880.3 °F

UNCERTAINTY FACTOR CALCULATIONS

6E3 1-19 45 MW		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
3.0 gpm		Coolant $\Delta T = 53.9$ f d		Film $\Delta T = 14.6$ f d		Cap. Wall $1/2 \Delta T = 20.0$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	5.4	0.10	1.5	0.10	2.0	8.9	79.2
Flow Rate		0.05	2.7					2.7	7.3
Flow Profile		0.05	2.7					2.7	7.3
Mfg. Deviation	Cladding Dimension					0.05	1.0	1.0	1.0
	U ₂ Pu Dimension			0.02	0.3	0.02	0.4	0.7	0.5
	²³⁵ U ²³⁹ Pu Dimension	0.01	0.5	0.01	0.1	0.01	0.2	0.8	0.6
Capsule Wall Thermal Conductivity						0.10	2.0	2.0	4.0
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	1.1	0.02	0.3	0.02	0.4	1.8	3.2
Transient Overload		0.05	2.7	0.05	0.7	0.05	1.0	4.4	19.4
								122.6	

$$\text{Uncertainty factor} = \sqrt{122.6} = 11.0 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 447.4 \\ &+ 11.0 \\ &= 458.4 \text{ } ^\circ\text{C} \\ &= 857.1 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

6E3 1-19 45 MW		$T_1 - T_1$ Out In	$T_2 - T_1$ (max)	$\frac{T_3 - T_2}{2}$ (Max Value)		
3.5 gpm		Coolant $\Delta T = 46.2$ f d	Film $\Delta T = 13.9$ f d	Cap. Wall $1/2 \Delta T = 19.5$ f d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10 4.6	0.10 1.4	0.10 2.0	8.0	64.0
Flow Rate		0.05 2.3			2.3	5.3
Flow Profile		0.05 2.3			2.3	5.3
Mfg. Deviation	Cladding Dimension			0.05 1.0	1.0	1.0
	U, Pu Dimension		0.02 0.3	0.02 0.4	0.7	0.5
	$^{235}\text{U}^{239}\text{Pu}$ Dimension	0.01 0.5	0.01 0.1	0.01 0.2	0.8	0.6
Capsule Wall Thermal Conductivity				0.10 2.0	2.0	4.0
Film Coefficient			0.02 0.3		0.3	0.1
Power Level		0.02 0.9	0.02 0.3	0.02 0.4	1.6	2.6
Transient Overload		0.05 2.3	0.05 0.7	0.05 1.0	4.0	16.0

99.4

$$\text{Uncertainty factor} = \sqrt{99.4} = 10.0 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 439.2 \\ &+ 10.0 \\ &= 449.2 \text{ } ^\circ\text{C} \\ &= 840.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
6E 1-19 45 MW		Out	In	(max)		(max value)			
4.0 gpm		Coolant $\Delta T = 40.4$		Film $\Delta T = 13.2$		Cap. Wall $1/2 \Delta T = 19.5$		Σd	$(\Sigma d)^2$
		f	d	f	d	f	d		
Neutron + Gamma Flux		0.10	4.0	0.10	1.3	0.10	2.0	7.3	51.3
Flow Rate		0.05	2.0					2.0	4.0
Flow Profile		0.05	2.0					2.0	4.0
Mfg. Deviation	Cladding Dimension					0.05	1.0	1.0	1.0
	U, Pu Dimension			0.02	0.3	0.02	0.4	0.7	0.5
	^{235}U ^{239}Pu Dimension	0.01	0.4	0.01	0.1	0.01	0.2	0.7	0.5
Capsule Wall Thermal Conductivity						0.10	2.0	2.0	4.0
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	0.8	0.02	0.3	0.02	0.4	1.5	2.3
Transient Overload		0.05	2.0	0.05	0.7	0.05	1.0	3.7	12.7

80.4

$$\text{Uncertainty factor} = \sqrt{80.4} = 9.0 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 432.9 \\ &+ 9.0 \\ &= 441.9 \text{ } ^\circ\text{C} \\ &= 827.4 \text{ } ^\circ\text{F} \end{aligned}$$

63E01	.3970+13	.5960+12	.1088+14	Fission Rates (Card 1)									
63E02	.3810+13	.5760+12	.1066+14										
63E03	.3650+13	.5530+12	.1046+14										
63E04	.3870+13	.5830+12	.1072+14										
63E05	.3720+13	.5630+12	.1054+14										
63E06	.3530+13	.5400+12	.1037+14										
63E07	.3450+13	.5150+12	.1025+14										
63E08	.3350+13	.5670+12	.1057+14										
63E09	.3620+13	.5480+12	.1041+14										
63E10	.3500+13	.5250+12	.1029+14										
63E11	.3430+13	.4990+12	.1020+14										
63E12	.3370+13	.4620+12	.1013+14										
63E13	.3520+13	.5300+12	.1031+14										
63E14	.3440+13	.5070+12	.1023+14										
63E15	.3390+13	.4780+12	.1015+14										
63E16	.3320+13	.4370+12	.1008+14										
63E17	.3400+13	.4240+12	.1006+14										
63E18	.3340+13	.4420+12	.1010+14										
63E19	.3270+13	.4090+12	.1002+14										
63E01	4.37	2.08		Gamma Heating (Card 2)									
63E02	4.19	1.96											
63E03	3.99	1.81											
63E04	4.24	1.99		PART I. Computer Input & Output for Calculation									
63E05	4.07	1.86											
63E06	3.88	1.73		of Temperatures in Highest Heat Genera-									
63E07	3.68	1.58											
63E08	4.10	1.89		tion Specimen in Subassembly Position									
63E09	3.94	1.76											
63E10	3.76	1.63		6E3/1, 45 MW Reactor Power 2.0 GPM Flow									
63E11	3.56	1.50											
63E12	3.36	1.34											
63E13	3.79	1.66											
63E14	3.62	1.54											
63E15	3.43	1.40											
63E16	3.24	1.24											
63E17	3.46	1.43											
63E18	3.29	1.28											
63E19	3.11	1.12											
1.00	.95	.81	.70	.63	.56	.50	.44	.37	.30	.15	.07	.00	(Card 3)
.65	.76	.87	.93	.97	.99	1.00	.99	.98	.95	.88	.77	.68	(Card 4)
45.0	200.	2.000											(Card 5)

34.29	34.29	34.29	34.29	34.29	84.06
.277	.277	.302	.425	.477	
.554	.327	.554	.604	.851	.954
.041	13.600	.150	.041	.136	.597-02
15-19	47.96	5.28	15.40	90.39	48.88

15-19	63E01	T1=	451.9	T2=	462.8	T3=	486.5	T6=	513.2	T7=	532.9	T8=	777.2
15-19	63E01	T1=	445.9	T2=	458.5	T3=	488.1	T6=	516.7	T7=	539.4	T8=	821.2
15-19	63E01	T1=	436.9	T2=	451.4	T3=	485.2	T6=	518.0	T7=	544.0	T8=	866.6
15-19	63E01	T1=	428.0	T2=	443.5	T3=	479.8	T6=	519.0	T7=	543.0	T8=	889.7
15-19	63E01	T1=	422.0	T2=	438.1	T3=	475.7	T6=	512.1	T7=	541.1	T8=	900.1
15-19	63E01	T1=	416.0	T2=	432.5	T3=	471.2	T6=	508.7	T7=	538.4	T8=	907.4
15-19	63E01	T1=	411.5	T2=	428.1	T3=	467.0	T6=	504.7	T7=	534.6	T8=	905.9
15-19	63E01	T1=	406.9	T2=	423.5	T3=	462.2	T6=	499.7	T7=	529.5	T8=	898.9
15-19	63E01	T1=	400.9	T2=	417.2	T3=	455.2	T6=	492.0	T7=	521.3	T8=	884.0
15-19	63E01	T1=	394.9	T2=	410.7	T3=	447.6	T6=	483.2	T7=	511.6	T8=	863.2
15-19	63E01	T1=	386.0	T2=	400.5	T3=	434.6	T6=	467.6	T7=	493.9	T8=	819.1
15-19	63E01	T1=	377.0	T2=	389.8	T3=	419.9	T6=	449.0	T7=	472.1	T8=	758.7
15-19	63E01	T1=	371.0	T2=	382.2	T3=	406.6	T6=	434.0	T7=	454.3	T8=	705.2

MAX WATTS/CM=	367.8	MAX KW/FT=	11.2	MAX CAL/SEC-CC=	364.8
FILM COEFFICIENT, CAL/SEC-CM2-C=	1.76	MAX CAPSULE HEAT FLUX, WATTS/CM2=	123.0		
FUEL VOLUME, CM3=	8.26

FIGURE D-6. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 6E3/1, 45 MW Reactor Power, 11.2 kW/ft (peak)*

FIGURE D-7. Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 6E3/1, 45 MW Reactor Power 11.2 kW/ft (peak)

63E01	.0970+13	.5960+12	.1088+14
19	47.96	5.28	15.40
.554	.330	.554	.604
.851	.954	.590	
36.116	5.080	.850	.064
14.576			
532.9	29.9	19.7	38.9
25.7	725.		
25.00	34.29	93.98	84.06
.192+08	.192+08	.187-04	.187-04
.32	.32		
45.0	310.	2.000	
.2724	.2620	.2318	1.00
25.	100.	200.	300.
400.	452.	500.	550.
.0	.1	.2	.3
.4	.5	.6	.7
.8	.9	1.0	

6E3/1, 45 MW Reactor Power
2.00 gpm Flow
21 in. Na Level

TOTAL FISSIONS=	.1017+23	AVERAGE BURNUP,A/O=	5.8	PEAK BURNUP,A/O=	6.6
TOTAL FISSION GAS GENERATED,CC=	100.14	TOTAL FISSION GAS RELEASED,CC=	100.14		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	914.5				
CLAD HOOP STRESS,PSI=	9967.9				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	7893.6	CLAD THERMAL STRESS AT TOP,PSI=	5200.8		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	10269.6	CAPSULE THERMAL STRESS AT TOP,PSI=	6784.8		
TEMP,C= 25.	NA VOL,CC= 14.58	NA LEVEL,CM= 53.34	VOID VOL,CC= 13.83	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 15.23	NA LEVEL,CM= 55.72	VOID VOL,CC= 13.18	CAP PRESS,PSI= 19.3	
TEMP,C= 200.	NA VOL,CC= 15.61	NA LEVEL,CM= 57.13	VOID VOL,CC= 12.80	CAP PRESS,PSI= 25.2	
TEMP,C= 300.	NA VOL,CC= 16.02	NA LEVEL,CM= 58.63	VOID VOL,CC= 12.39	CAP PRESS,PSI= 31.6	
TEMP,C= 400.	NA VOL,CC= 16.46	NA LEVEL,CM= 60.24	VOID VOL,CC= 11.95	CAP PRESS,PSI= 38.4	
TEMP,C= 452.	NA VOL,CC= 16.71	NA LEVEL,CM= 61.13	VOID VOL,CC= 11.70	CAP PRESS,PSI= 42.3	
TEMP,C= 500.	NA VOL,CC= 16.94	NA LEVEL,CM= 61.97	VOID VOL,CC= 11.47	CAP PRESS,PSI= 46.0	
TEMP,C= 550.	NA VOL,CC= 17.18	NA LEVEL,CM= 62.88	VOID VOL,CC= 11.22	CAP PRESS,PSI= 50.0	
CAPSULE PRESSURE AT 452C,PSI=	42.3	CORRESPONDING CAPSULE HOOP STRESS,PSI=	349.5		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	284.1	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2349.5		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 452C					
HEIGHT OF SODIUM,CM=	.000	CAPSULE PRESSURE,PSI=	17.4		
HEIGHT OF SODIUM,CM=	8.406	CAPSULE PRESSURE,PSI=	19.0		
HEIGHT OF SODIUM,CM=	16.812	CAPSULE PRESSURE,PSI=	20.9		
HEIGHT OF SODIUM,CM=	25.218	CAPSULE PRESSURE,PSI=	23.2		
HEIGHT OF SODIUM,CM=	33.624	CAPSULE PRESSURE,PSI=	26.1		
HEIGHT OF SODIUM,CM=	42.030	CAPSULE PRESSURE,PSI=	29.8		

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	34.8
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	41.8
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	52.2
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	69.6
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	104.2

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BNWL-650

63E01	.8970+13	.5960+12	.1088+14
=19	47.96	5.28	15.40
.554	.330	.554	.604
.851	.954	.590	
36.116	5.080	.850	.064
15.966			
532.9	29.9	19.7	38.9
25.7	725.		
25.00	34.29	93.98	84.06
.192+08	.192+08	.187-04	.187-04
.32	.32		
45.0	310.	2.000	
.2724	.2620	.2318	1.00
25.	100.	200.	300.
400.	452.	500.	550.
.0	.1	.2	.3
.4	.5	.6	.7
.8	.9	1.0	

6E3/1, 45 MW Reactor Power
2.0 gpm Flow
23 in. Na Level

TOTAL FISSIONS=	.1017+23	AVERAGE BURNUP,A/O=	5.8	PEAK BURNUP,A/O=	6.6
TOTAL FISSION GAS GENERATED,CC=	100.14	TOTAL FISSION GAS RELEASED,CC=	100.14		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	914.5				
CLAD HOOP STRESS,PSI=	9967.9				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	7893.6	CLAD THERMAL STRESS AT TOP,PSI=	5200.8		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	10269.6	CAPSULE THERMAL STRESS AT TOP,PSI=	6784.8		
TEMP,C= 25.	NA VOL,CC= 15.97	NA LEVEL,CM= 58.42	VOID VOL,CC= 12.44	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 16.68	NA LEVEL,CM= 61.03	VOID VOL,CC= 11.73	CAP PRESS,PSI= 19.5	
TEMP,C= 200.	NA VOL,CC= 17.10	NA LEVEL,CM= 62.58	VOID VOL,CC= 11.31	CAP PRESS,PSI= 25.7	
TEMP,C= 300.	NA VOL,CC= 17.55	NA LEVEL,CM= 64.22	VOID VOL,CC= 10.86	CAP PRESS,PSI= 32.4	
TEMP,C= 400.	NA VOL,CC= 18.03	NA LEVEL,CM= 65.99	VOID VOL,CC= 10.38	CAP PRESS,PSI= 39.8	
TEMP,C= 452.	NA VOL,CC= 18.30	NA LEVEL,CM= 66.95	VOID VOL,CC= 10.11	CAP PRESS,PSI= 44.0	
TEMP,C= 500.	NA VOL,CC= 18.55	NA LEVEL,CM= 67.88	VOID VOL,CC= 9.86	CAP PRESS,PSI= 48.1	
TEMP,C= 550.	NA VOL,CC= 18.62	NA LEVEL,CM= 68.87	VOID VOL,CC= 9.59	CAP PRESS,PSI= 52.7	
CAPSULE PRESSURE AT 452C,PSI=	44.0	CORRESPONDING CAPSULE HOOP STRESS,PSI=	363.9		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	311.2	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2573.8		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 452C					
HEIGHT OF SODIUM,CM=	.000	CAPSULE PRESSURE,PSI=	15.7		
HEIGHT OF SODIUM,CM=	9.406	CAPSULE PRESSURE,PSI=	17.1		
HEIGHT OF SODIUM,CM=	16.812	CAPSULE PRESSURE,PSI=	18.8		
HEIGHT OF SODIUM,CM=	25.218	CAPSULE PRESSURE,PSI=	20.9		
HEIGHT OF SODIUM,CM=	33.624	CAPSULE PRESSURE,PSI=	23.5		
HEIGHT OF SODIUM,CM=	42.030	CAPSULE PRESSURE,PSI=	26.8		

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	31.3
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	37.6
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	47.0
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	62.6
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	93.8

FIGURE D-8. Sodium Level Versus Sodium Temperature in Hottest Specimen in Sub-assembly. Position 6E3/1, 45 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-9. Internal Capsule Pressure as a Function of Capsule Temperature.
Position 6E3/1, 45 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-10. Internal Void Volume as a Function of Capsule Temperature.
Position 6E3/1, 45 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-11. Sodium Level Versus Reactor Coolant Pressure in Capsule with Leak at the Lower End, for Hottest Specimen in Subassembly. Position 6E3/1, 45 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-12. Capsule and Specimen Temperatures During Loss of Flow Accident for Hottest Specimen in Subassembly. Position 6E3/1, 45 MW Initial Reactor Power, 2.0 gal/min Initial Flow Rate

APPENDIX D - PART II

PREDICTED PERFORMANCE AT 25% OVERPOWER CONDITIONS

TABLE D-II. Specimen Heat Generation Rates and Temperatures at 56.25 MW Reactor Power (25% overpower from normal 45 MW reactor power)*

Position Number	Specimen Number	Maximum Heat Generation Rate, kW/ft	Maximum** Mean Capsule Wall Temp, °C	Maximum Capsule Surface Heat Flux, W/cm ²	Coolant** ΔT, C°
1	19	14.0	502 [†]	153.7	101
2	7	13.7	499	150.8	99
3	8	13.5	497	147.8	97
4	9	13.8	500	151.8	99
5	10	13.6	498	149.1	98
6	15	13.3	495	146.3	96
7	16	13.1	493	144.0	95
8	11	13.7	499	150.8	99
9	12	13.4	496	147.2	97
10	17	13.2	495	144.9	95
11	1	13.5	497	148.5	97
12	2	13.4	496	147.1	97
13	18	13.2	495	145.3	95
14	4	13.6	498	148.8	98
15	5	13.5	497	147.6	97
16	13	13.3	495	146.0	96
17	3	13.4	496	147.4	97
18	6	13.4	496	146.5	97
19	14	13.2	495	144.9	95
		Σ = 254.5		Σ = 1745	

$$\text{Total heat generation} = \frac{254.5}{1.14} \times \frac{13.5}{12} = 251 \text{ kW};$$

$$\text{Mean coolant } \Delta T = \frac{1745}{19} = 91.8 \text{ C}^\circ$$

$$\text{Mean coolant exit temp} = 371 + 91.8 = 462.8 \text{ }^\circ\text{C} \text{ (865 }^\circ\text{F)}.$$

* Core position 6E3

** Based on coolant flow rate of 2 gpm

† Uncertainty factor of 18.3 C° increases this to 520.3 °C.

FIGURE D-13. Coolant ΔT Versus Flow Rate for Highest and Lowest Specimen Linear Heat Generation in Subassembly. Position 6E3, 56.25 MW Reactor Power

FIGURE D-14. Maximum Mean Capsule Wall Temperature Versus Coolant Flow Rate for Highest and Lowest Specimen Linear Heat Generation in Subassembly. Position 6E3, 56.25 MW Reactor Power

FIGURE D-15. Maximum Mean Capsule Wall Temperature with Uncertainty Factors Versus Flow for Highest Linear Heat Generation Specimen in Sub-assembly. Position 6E3/1, 56.25 MW Reactor Power, 14.0 kW/ft (peak)

FIGURE D-16. Coolant-Capsule Film ΔT for Highest Specimen Linear Heat Generation in Subassembly. Position 6E3/1, 56.25 MW Reactor Power, 14.0 kW/ft (peak)

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FIGURE D-17. *Uncertainty Factor Versus Flow for Highest Specimen Linear Heat Generation Rate in Subassembly. Position 6E3/1, 56.25 MW Reactor Power, 14.0 kW/ft (peak)*

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
6E3 1-19 56.25 MW		Out In		(max)		(max value)			
		Coolant		Film		Cap. Wall			
		$\Delta T = 202.3$		$\Delta T = 25.4$		$1/2 \Delta T = 24.3$			
1.0 gpm		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	20.2	0.10	2.5	0.10	2.4	25.1	630.0
Flow Rate		0.05	10.1					10.1	102.0
Flow Profile		0.05	10.1					10.1	102.0
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.5	0.02	0.5	1.0	1.0
	^{235}U ^{239}Pu Dimension	0.01	2.0	0.01	0.3	0.01	0.2	2.5	6.3
Capsule Wall Thermal Conductivity						0.10	2.3	2.3	5.8
Film Coefficient				0.02	0.5			0.5	0.3
Power Level		0.02	4.0	0.02	0.5	0.02	0.5	5.0	25.0
Transient Overload		0.05	10.1	0.05	1.3	0.05	1.2	12.6	158.7
		1032.5							

$$\text{Uncertainty factor} = \sqrt{1032.5} = 32.1 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 606.0 \\ &+ \frac{32.1}{638.1} \text{ } ^\circ\text{C} \\ &= 1180.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
6E3 1-19 56.25 MW		Coolant $\Delta T=134.8$		Film $\Delta T=22.7$		Cap. Wall $1/2\Delta T=24.3$			
1.5 gpm		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	13.5	0.10	2.3	0.10	2.4	18.2	331.2
Flow Rate		0.05	6.7					6.7	44.9
Flow Profile		0.05	6.7					6.7	44.9
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.5	0.02	0.5	1.0	1.0
	^{235}U ^{239}Pu Dimension	0.01	1.3	0.01	0.2	0.01	0.2	1.7	2.9
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.5			0.5	0.3
Power Level		0.02	2.7	0.02	0.5	0.02	0.5	3.7	13.7
Transient Overload		0.05	6.7	0.05	1.1	0.05	1.2	9.0	81.0

527.1

$$\text{Uncertainty factor} = \sqrt{527.1} = 23.0 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 536.8 \\ &+ 23.0 \\ &= 559.8 \text{ } ^\circ\text{C} \\ &= 1039.6 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$		$T_2 - T_1$		$\frac{T_3 - T_2}{2}$			
6E3 1-19 56.25 MW		Out	In	(max)		(max value)			
2.0 gpm		Coolant $\Delta T = 101.1$		Film $\Delta T = 20.8$		Cap. Wall $1/2 \Delta T = 24.3$		Σd	$(\Sigma d)^2$
		f	d	f	d	f	d		
Neutron + Gamma Flux		0.10	10.1	0.10	2.1	0.10	2.4	14.6	213.2
Flow Rate		0.05	5.0					5.0	25.0
Flow Profile		0.05	5.0					5.0	25.0
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.4	0.02	0.5	0.9	0.8
	^{235}U , ^{239}Pu Dimension	0.01	1.0	0.01	0.2	0.01	0.2	1.4	1.9
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	2.0	0.02	0.4	0.02	0.5	2.9	8.4
Transient Overload		0.05	5.0	0.05	1.0	0.05	1.2	7.2	51.8

333.5

$$\text{Uncertainty factor} = \sqrt{333.5} = 18.3 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 501.8 \\ &+ 18.3 \\ &= 520.1 \text{ } ^\circ\text{C} \\ &= 968.2 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max) d		$\frac{T_3 - T_2}{2}$ (max value) f d			
6E3 1-19 56.25 MW		Coolant $\Delta T=80.9$		Film $\Delta T=19.3$		Cap. Wall $1/2 \Delta T=24.3$			
2.5 gpm		f d		f d		f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	8.1	0.10	1.9	0.10	2.4	12.4	153.7
Flow Rate		0.05	4.0					4.0	16.0
Flow Profile		0.05	4.0					4.0	16.0
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.3
	U, Pu Dimension			0.02	0.4	0.02	0.5	0.9	0.8
	^{235}U ^{239}Pu Dimension	0.01	0.8	0.01	0.2	0.01	0.2	1.2	1.4
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	1.6	0.02	0.4	0.02	0.5	2.5	6.3
Transient Overload		0.05	4.0	0.05	1.0	0.05	1.2	6.2	38.4
								240.0	

$$\text{Uncertainty factor} = \sqrt{240.0} = 15.5 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 480.7 \\ &+ 15.5 \\ &= 496.2 \text{ } ^\circ\text{C} \\ &= 925.2 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
6E3 1-19 56.25 MW		Coolant $\Delta T = 67.4$		Film $\Delta T = 18.3$		Cap. Wall $1/2 \Delta T = 24.3$			
30 gpm		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	6.7	0.10	1.8	0.10	2.4	10.9	118.8
Flow Rate		0.05	3.3					3.3	10.9
Flow Profile		0.05	3.3					3.3	10.9
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.4	0.02	0.5	0.9	0.8
	^{235}U ^{239}Pu Dimension	0.01	0.7	0.01	0.2	0.01	0.2	1.1	1.2
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.4			0.4	0.2
Power Level		0.02	1.3	0.02	0.4	0.02	0.5	2.2	4.8
Transient Overload		0.05	3.3	0.05	0.9	0.05	1.2	5.4	29.2
								184.0	

$$\text{Uncertainty factor} = \sqrt{184.0} = 13.6 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 466.4 \\ &+ 13.6 \\ &= 480.0 \text{ } ^\circ\text{C} \\ &= 896.0 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
6E3 1-19 56.25 MW		Coolant $\Delta T = 57.8$		Film $\Delta T = 17.3$		Cap. Wall $1/2 \Delta T = 24.3$			
3.5 gpm		f	d	f	d	f	d	Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	5.8	0.10	1.7	0.10	2.4	9.9	98.0
Flow Rate		0.05	2.9					2.9	8.4
Flow Profile		0.05	2.9					2.9	8.4
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.3	0.02	0.5	0.8	0.6
	$^{235}\text{U}^{239}\text{Pu}$ Dimension	0.01	0.6	0.01	0.2	0.01	0.2	1.0	1.0
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	1.2	0.02	0.3	0.02	0.5	2.0	4.0
Transient Overload		0.05	2.9	0.05	0.9	0.05	1.2	5.0	25.0
		152.7							

$$\text{Uncertainty factor} = \sqrt{152.7} = 12.4 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 456.2 \\ &+ 12.4 \\ &= 468.6 \text{ } ^\circ\text{C} \\ &= 875.5 \text{ } ^\circ\text{F} \end{aligned}$$

UNCERTAINTY FACTOR CALCULATIONS

6E3 1-19 56.25 MW		$T_1 - T_1$ Out In		$T_2 - T_1$ (max)		$\frac{T_3 - T_2}{2}$ (max value)			
4.0 gpm		Coolant $\Delta T=50.6$ f d		Film $\Delta T=16.5$ f d		Cap. Wall $1/2\Delta T=24.3$ f d		Σd	$(\Sigma d)^2$
Neutron + Gamma Flux		0.10	5.1	0.10	1.7	0.10	2.4	9.2	84.6
Flow Rate		0.05	2.5					2.5	6.3
Flow Profile		0.05	2.5					2.5	6.3
Mfg. Deviation	Cladding Dimension					0.05	1.2	1.2	1.4
	U, Pu Dimension			0.02	0.3	0.03	0.5	0.8	0.6
	^{235}U ^{239}Pu Dimension	0.01	0.5	0.01	0.2	0.01	0.2	0.9	0.8
Capsule Wall Thermal Conductivity						0.10	2.4	2.4	5.8
Film Coefficient				0.02	0.3			0.3	0.1
Power Level		0.02	1.0	0.02	0.3	0.02	0.5	1.8	3.2
Transient Overload		0.05	2.5	0.05	0.8	0.05	1.2	4.5	20.3

129.4

$$\text{Uncertainty factor} = \sqrt{129.4} = 11.4 \text{ } ^\circ\text{C}$$

$$\begin{aligned} \text{Maximum capsule mean wall temperature} &= 448.5 \\ &+ 11.4 \\ &= 459.9 \text{ } ^\circ\text{C} \\ &= 859.8 \text{ } ^\circ\text{F} \end{aligned}$$

63E01	.6976+13	.5860+12	.1088+14
63E02	.8516+13	.5760+12	.1066+14
63E03	.8680+13	.5530+12	.1046+14
63E04	.8876+13	.5136+12	.1072+14
63E05	.8720+13	.5630+12	.1054+14
63E06	.8560+13	.5400+12	.1037+14
63E07	.8050+13	.5150+12	.1025+14
63E08	.8050+13	.5670+12	.1057+14
63E09	.8620+13	.5480+12	.1041+14
63E10	.8500+13	.5250+12	.1029+14
63E11	.8430+13	.4990+12	.1020+14
63E12	.8370+13	.4620+12	.1013+14
63E13	.8520+13	.5300+12	.1031+14
63E14	.8450+13	.5070+12	.1023+14
63E15	.8590+13	.4720+12	.1015+14
63E16	.8520+13	.4370+12	.1008+14
63E17	.8400+13	.4610+12	.1006+14
63E18	.8560+13	.4420+12	.1010+14
63E19	.8270+13	.6020+12	.1002+14

Fission Rates (Card 1)

63E01	5.46	2.61
63E02	5.24	2.45
63E03	4.99	2.26
63E04	5.30	2.49
63E05	5.09	2.33
63E06	4.35	2.16
63E07	4.60	1.98
63E08	5.13	2.37
63E09	4.92	2.21
63E10	4.70	2.04
63E11	4.46	1.87
63E12	4.19	1.67
63E13	4.74	2.08
63E14	4.55	1.93
63E15	4.29	1.76
63E16	4.05	1.55
63E17	4.33	1.79
63E18	4.11	1.60
63E19	3.39	1.40

Gamma Heating (Card 2)

PART 1. Computer Input & Output for Calculation
of Temperatures in Highest Heat Generation Specimen in Subassembly Position
6E3/1, 56.25 MW Reactor Power, 2.0 GPM Flow.

1.00	.93	.81	.70	.63	.56	.50	.44	.37	.30	.18	.07	.00
.65	.71	.87	.93	.97	.99	1.00	.99	.98	.95	.88	.77	.68

56.3 200. 2.000 (Card 5)

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34.29	34.29	34.29	34.29	34.29	84.06
.277	.277	.302	.425	.477	
.554	.327	.354	.604	.651	.954
.041	13.600	.159	.041	.136	.597-02
15-19	47.96	5.26	15.60	90.39	43.38

15-19	63E01	T1=	472.1	T2=	475.2	T3=	517.3	T6=	548.3	T7=	575.4	T8=	673.8
15-19	63E01	T1=	469.6	T2=	469.0	T3=	517.2	T6=	543.1	T7=	581.5	T8=	933.7
15-19	63E01	T1=	453.6	T2=	471.5	T3=	513.8	T6=	554.7	T7=	587.2	T8=	990.5
15-19	63E01	T1=	442.2	T2=	461.5	T3=	507.1	T6=	531.0	T7=	595.0	T8=	1019.4
15-19	63E01	T1=	431.7	T2=	454.8	T3=	501.9	T6=	547.4	T7=	583.6	T8=	1032.4
15-19	63E01	T1=	427.2	T2=	447.9	T3=	496.3	T6=	543.1	T7=	580.3	T8=	1041.6
15-19	63E01	T1=	421.5	T2=	442.4	T3=	491.9	T6=	533.1	T7=	575.5	T8=	1039.6
15-19	63E01	T1=	415.3	T2=	436.6	T3=	485.0	T6=	531.9	T7=	569.1	T8=	1030.9
15-19	63E01	T1=	408.4	T2=	428.7	T3=	476.3	T6=	522.3	T7=	558.8	T8=	1012.2
15-19	63E01	T1=	399.9	T2=	420.6	T3=	466.7	T6=	511.3	T7=	546.7	T8=	996.2
15-19	63E01	T1=	392.7	T2=	407.9	T3=	450.5	T6=	491.8	T7=	524.6	T8=	931.1
15-19	63E01	T1=	373.5	T2=	394.5	T3=	432.1	T6=	468.5	T7=	497.3	T8=	855.6
15-19	63E01	T1=	371.0	T2=	385.1	T3=	417.9	T6=	449.8	T7=	475.1	T8=	788.8
15-19	63E01	MAX WATTS/CM= 459.7 MAX KW/FT= 16.0 MAX CAL/SEC-CM= 455.9 FILM COEFFICIENT, CAL/SEC-CM2-C= 1.76 MAX CAPSULE HEAT FLUX, WATTS/CM2= 153.7 FUEL VOLUME, CM3= 3.26

FIGURE D-18. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 6E3/1, 56.25 MW Reactor Power, 14.0 kW/ft (peak)*

FIGURE D-19. *Calculated Axial Temperature Profiles for Hottest Specimen in Subassembly. Position 6E3/1, 56.25 MW Reactor Power, 14.0 kW/ft (peak)*

63E01	.8970+13	.5960+12	.1088+14
=19	47.96	5.28	15.40
.554	.330	.554	.604
.851	.954	.590	
36.116	5.680	.850	.064
14.576			
573.4	37.4	24.6	48.6
32.0	745.		
25.00	34.29	93.98	84.06
.192+08	.192+08	.187-04	.187-04
.32	.32		
56.3	248.	2.000	
.2724	.2620	.2318	1.00
25.	100.	200.	300.
400.	472.	500.	550.
.0	.1	.2	.3
.4	.5	.6	.7
.8	.9	1.0	

6E3/1, 56.25 MW Reactor Power

2.0 gpm Flow

21 in. Na Level

TOTAL FISSIONS= .1017+23 AVERAGE BURNUP,A/O= 5.8 PEAK BURNUP,A/O= 6.6

TOTAL FISSION GAS GENERATED,CC= 100.14 TOTAL FISSION GAS RELEASED,CC= 100.14

INITIAL VOID OF PLENUM,CC= 5.05 TOTAL INITIAL VOID,CC= 6.29

FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI= 960.5

CLAD HOOP STRESS,PSI= 10468.9

CLAD THERMAL STRESS AT MIDPLANE,PSI= 9673.6 CLAD THERMAL STRESS AT TOP,PSI= 6494.4

CAPSULE THERMAL STRESS AT MIDPLANE,PSI= 12830.4 CAPSULE THERMAL STRESS AT TOP,PSI= 8448.0

TEMP,C= 25.	NA VOL,CC= 14.58	NA LEVEL,CM= 53.34	VOID VOL,CC= 13.83	CAP PRESS,PSI= 14.7
TEMP,C= 100.	NA VOL,CC= 15.23	NA LEVEL,CM= 55.72	VOID VOL,CC= 13.18	CAP PRESS,PSI= 19.3
TEMP,C= 200.	NA VOL,CC= 15.61	NA LEVEL,CM= 57.13	VOID VOL,CC= 12.80	CAP PRESS,PSI= 25.2
TEMP,C= 300.	NA VOL,CC= 16.02	NA LEVEL,CM= 58.63	VOID VOL,CC= 12.39	CAP PRESS,PSI= 31.6
TEMP,C= 400.	NA VOL,CC= 16.46	NA LEVEL,CM= 60.24	VOID VOL,CC= 11.95	CAP PRESS,PSI= 38.4
TEMP,C= 472.	NA VOL,CC= 16.80	NA LEVEL,CM= 61.48	VOID VOL,CC= 11.61	CAP PRESS,PSI= 43.8
TEMP,C= 500.	NA VOL,CC= 16.94	NA LEVEL,CM= 61.97	VOID VOL,CC= 11.47	CAP PRESS,PSI= 46.0
TEMP,C= 550.	NA VOL,CC= 17.18	NA LEVEL,CM= 62.88	VOID VOL,CC= 11.22	CAP PRESS,PSI= 50.0

CAPSULE PRESSURE AT 472C,PSI= 43.8 CORRESPONDING CAPSULE HOOP STRESS,PSI= 362.2

CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI= 292.0 CORRESPONDING CAPSULE HOOP STRESS,PSI= 2414.6

SODIUM LEVEL WITH LEAK IN CAPSULE AT 472C

HEIGHT OF SODIUM,CM= .090	CAPSULE PRESSURE,PSI= 17.9
HEIGHT OF SODIUM,CM= 3.406	CAPSULE PRESSURE,PSI= 19.5
HEIGHT OF SODIUM,CM= 16.812	CAPSULE PRESSURE,PSI= 21.5
HEIGHT OF SODIUM,CM= 25.218	CAPSULE PRESSURE,PSI= 23.9
HEIGHT OF SODIUM,CM= 33.624	CAPSULE PRESSURE,PSI= 26.8
HEIGHT OF SODIUM,CM= 42.030	CAPSULE PRESSURE,PSI= 30.7

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HEIGHT OF SODIUM, CM=	50.436	CAPSULE PRESSURE, PSI=	35.8
HEIGHT OF SODIUM, CM=	58.842	CAPSULE PRESSURE, PSI=	42.9
HEIGHT OF SODIUM, CM=	67.248	CAPSULE PRESSURE, PSI=	53.6
HEIGHT OF SODIUM, CM=	75.654	CAPSULE PRESSURE, PSI=	71.5
HEIGHT OF SODIUM, CM=	84.060	CAPSULE PRESSURE, PSI=	107.1

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BNWL-650

63E01	.8970+13	.5960+12	.1088+14	
=19	47.96	5.28	15.40	90.39 48.88
.554	.330	.554	.604	.851 .954 .590
36.116	5.080	.850	.064	15.966
573.4	37.4	24.6	48.6	32.0 745.
25.00	34.29	93.98	84.06	
.192+08	.192+08	.187-04	.187-04	.32 .32
56.3	248.	2.000		
.2724	.2620	.2318	1.00	
25. 100. 200. 300. 400. 472. 500. 550.				
.0 .1 .2 .3 .4 .5 .6 .7 .8 .9 1.0				

6E3/1, 56.25 MW Reactor Power
2.0 gpm Flow
23 in. Na Level

TOTAL FISSIONS=	.1017+23	AVERAGE BURNUP,A/O=	5.8	PEAK BURNUP,A/O=	6.6
TOTAL FISSION GAS GENERATED,CC=	100.14	TOTAL FISSION GAS RELEASED,CC=	100.14		
INITIAL VOID OF PLENUM,CC=	5.05	TOTAL INITIAL VOID,CC=	6.29		
FINAL GAS PRESSURE AT TOP OF FUEL PIN,PSI=	960.5				
CLAD HOOP STRESS,PSI=	10468.9				
CLAD THERMAL STRESS AT MIDPLANE,PSI=	9873.6	CLAD THERMAL STRESS AT TOP,PSI=	6494.4		
CAPSULE THERMAL STRESS AT MIDPLANE,PSI=	12830.4	CAPSULE THERMAL STRESS AT TOP,PSI=	8448.0		
TEMP,C= 25.	NA VOL,CC= 15.97	NA LEVEL,CM= 58.42	VOID VOL,CC= 12.4.	CAP PRESS,PSI= 14.7	
TEMP,C= 100.	NA VOL,CC= 16.68	NA LEVEL,CM= 61.03	VOID VOL,CC= 11.73	CAP PRESS,PSI= 19.5	
TEMP,C= 200.	NA VOL,CC= 17.10	NA LEVEL,CM= 62.58	VOID VOL,CC= 11.31	CAP PRESS,PSI= 25.7	
TEMP,C= 300.	NA VOL,CC= 17.55	NA LEVEL,CM= 64.22	VOID VOL,CC= 10.86	CAP PRESS,PSI= 32.4	
TEMP,C= 400.	NA VOL,CC= 18.03	NA LEVEL,CM= 65.99	VOID VOL,CC= 10.38	CAP PRESS,PSI= 39.8	
TEMP,C= 472.	NA VOL,CC= 18.40	NA LEVEL,CM= 67.34	VOID VOL,CC= 10.01	CAP PRESS,PSI= 45.7	
TEMP,C= 500.	NA VOL,CC= 18.55	NA LEVEL,CM= 67.88	VOID VOL,CC= 9.86	CAP PRESS,PSI= 48.1	
TEMP,C= 550.	NA VOL,CC= 18.82	NA LEVEL,CM= 68.87	VOID VOL,CC= 9.59	CAP PRESS,PSI= 52.7	
CAPSULE PRESSURE AT 472C,PSI=	45.7	CORRESPONDING CAPSULE HOOP STRESS,PSI=	377.9		
CAPSULE PRESSURE WITH RUPTURED SPECIMEN,PSI=	319.9	CORRESPONDING CAPSULE HOOP STRESS,PSI=	2645.1		
SODIUM LEVEL WITH LEAK IN CAPSULE AT 472C					
HEIGHT OF SODIUM,CM=	.000	CAPSULE PRESSURE,PSI=	16.1		
HEIGHT OF SODIUM,CM=	8.406	CAPSULE PRESSURE,PSI=	17.6		
HEIGHT OF SODIUM,CM=	16.812	CAPSULE PRESSURE,PSI=	19.3		
HEIGHT OF SODIUM,CM=	25.218	CAPSULE PRESSURE,PSI=	21.5		
HEIGHT OF SODIUM,CM=	33.624	CAPSULE PRESSURE,PSI=	24.1		
HEIGHT OF SODIUM,CM=	42.030	CAPSULE PRESSURE,PSI=	27.6		

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HEIGHT OF SODIUM,CM=	50.436	CAPSULE PRESSURE,PSI=	32.2
HEIGHT OF SODIUM,CM=	58.842	CAPSULE PRESSURE,PSI=	38.6
HEIGHT OF SODIUM,CM=	67.248	CAPSULE PRESSURE,PSI=	48.3
HEIGHT OF SODIUM,CM=	75.654	CAPSULE PRESSURE,PSI=	64.3
HEIGHT OF SODIUM,CM=	84.060	CAPSULE PRESSURE,PSI=	96.4

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FIGURE D-20. Sodium Level Versus Sodium Temperature in Hottest Specimen in Sub-assembly. Position 6E3/1, 56.25 MW Reactor Power, 2.0 gal/min Flow

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FIGURE D-21. *Internal Capsule Pressure as a Function of Capsule Temperature.
Position 6E3/1, 56.25 MW Reactor Power, 2.0 gal/min Flow*

FIGURE D-22. Internal Void Volume as a Function of Capsule Temperature. Position 6E3/1, 56.25 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-23. Sodium Level Versus Reactor Coolant Pressure in Capsule with Leak at the Lower End, for Hottest Specimen in Subassembly. Position 6E3/1, 56.25 MW Reactor Power, 2.0 gal/min Flow

FIGURE D-24. Capsule and Specimen Temperatures During Loss of Flow Accident for Hottest Specimen in Subassembly. Position 6E3/1, 56.25 MW Initial Reactor Power, 2.0 gal/min Initial Flow Rate

APPENDIX E

RADIAL DISTANCES - CENTER OF CAPSULES TO CENTER OF REACTOR

Radial Distances from Core Center Line to Each Capsule in EBR-II

Pin Number	Distance, cm						
	Position 2E1	Position 3E1	Position 3E2	Position 4E1	Position 4E2	Position 6E1	Position 6E2
1	3.9291	9.6958	7.7100	15.5652	13.1352	27.3235	24.6345
2	5.1609	10.8623	8.4090	16.7051	14.0043	28.4416	25.6050
3	6.3986	12.0424	9.2179	17.8559	14.9210	29.5659	26.5929
4	3.7300	9.6200	8.4090	15.5200	13.5804	27.3000	24.8913
5	4.8479	10.7157	8.9600	16.6092	14.3606	28.3843	25.8141
6	6.0212	11.8461	9.6459	17.7241	15.2079	29.4865	26.7680
7	7.2167	12.9953	10.4360	18.8530	16.1062	30.5972	27.7436
8	3.9291	9.6959	9.2179	15.5652	14.1136	27.3235	25.1977
9	4.8479	10.7157	9.6459	16.6092	14.8175	28.3844	26.0845
10	5.8900	11.7800	10.2100	17.6800	15.5900	29.4600	27.0000
11	7.0005	12.8777	10.8889	18.7729	16.4214	30.5489	27.9415
12	8.1515	14.0010	11.6627	19.8844	17.3033	31.6498	28.9063
13	5.1609	10.8623	10.4360	16.7051	15.3629	28.4416	26.4125
14	6.0212	11.8461	10.8889	17.7241	16.0605	29.4865	27.2873
15	7.0005	12.8777	11.4600	18.7729	16.8224	30.5489	28.1915
16	8.0500	13.9400	12.1288	19.8400	17.6352	31.6200	29.1163
17	6.3986	12.0424	11.6627	17.8559	16.6120	29.5659	27.6282
18	7.2167	12.9953	12.1288	18.8530	17.3005	30.5972	28.4906
19	8.1515	14.0010	12.7100	19.8844	18.0572	31.6498	29.3878

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Radial Distances from Core Center Line to Each Capsule in EBR-II

Pin Number	Distance, cm				
	Position 6E3	Position 7E1	Position 7E2	Position 7E3	Position 7E4
1	23.2052	33.2185	30.4728	28.7213	28.1200
2	23.9762	34.3310	31.4651	29.5580	28.7697
3	24.7786	35.4483	32.4694	30.4158	29.4497
4	23.7330	33.2000	30.6851	29.1596	28.7697
5	24.4472	34.2832	31.6383	29.9498	29.3700
6	25.2075	35.3821	32.6162	30.7749	30.0145
7	26.0047	36.4880	33.6101	31.6267	30.6961
8	24.3048	33.2185	30.9367	29.6350	29.4497
9	24.9758	34.2832	31.8626	30.3916	30.0145
10	25.6900	35.3600	32.8100	31.1800	30.6200
11	26.4441	36.4479	33.7771	31.9978	31.2637
12	27.2347	37.5459	34.7623	32.8428	31.9435
13	25.5547	34.3310	32.1354	30.8783	30.6961
14	26.2233	35.3821	33.0500	31.6293	31.2638
15	26.9335	36.4479	33.9873	32.4117	31.8700
16	27.6772	37.5200	34.9392	33.2179	32.5086
17	26.8043	35.4483	33.3352	32.1211	31.9435
18	27.4668	36.4880	34.2384	32.8640	32.5086
19	28.1775	37.5459	35.1696	33.6448	33.1200

Radial Distances from Core Center Line to Each Capsule in EBR-II

Pin Number	Distance, cm			
	Position 8E1	Position 8E2	Position 8E3	Position 8E4
1	39.1049	36.3243	34.3672	33.3475
2	40.2135	37.3315	35.2472	34.0771
3	41.3257	38.3472	36.1421	34.8282
4	39.0900	36.5055	34.7418	33.9069
5	40.1723	37.4792	35.5824	34.5936
6	41.2689	38.4736	36.4511	35.3156
7	42.3713	39.4804	37.3401	36.0658
8	39.1049	36.7185	35.1474	34.4936
9	40.1723	37.6708	35.9615	35.1510
10	41.2500	38.6400	36.8000	35.8400
11	42.3371	39.6250	37.6614	36.5587
12	43.4331	40.6245	38.5440	37.3055
13	40.2135	37.9042	36.3804	35.7436
14	41.2689	38.8460	37.1876	36.3998
15	42.3371	39.8062	38.0197	37.0866
16	43.4100	40.7771	38.8692	37.7977
17	41.3257	39.0909	37.6128	36.9934
18	42.3713	40.0224	38.4115	37.6443
19	43.4331	40.9774	39.2413	38.3335

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APPENDIX F

DESCRIPTION OF INPUT FOR EBR-II HAZARDS ANALYSIS PROGRAM

APPENDIX F

DESCRIPTION OF INPUT FOR EBR-II HAZARDS ANALYSIS PROGRAM

A detailed description of the input data required by the EBR-II Hazards Analysis program and the format in which it must be presented are given here.

The Fortran symbol used in the program is followed by its definition and the column numbers of the IBM card on which it is submitted.

The format for each input is in parenthesis following the title description.

<u>PART I - AXIAL AND RADIAL TEMPERATURE PROFILE</u>		<u>Columns</u>
Card 1 - Fission Rates (1H1, 9X, 2X, A5, 3E15.4)		
	- Not used	1 - 1
	- Not used	2 - 10
	- Not used	11 - 12
DENT	- Capsule location in EBR-II	13 - 17
FRU35	- Fission rate of ^{235}U (fissions/g-sec)	18 - 32
FRU38	- Fission rate of ^{238}U (fissions/g-sec)	33 - 47
FRPU	- Fission rate of Pu (fissions/g-sec)	48 - 62
	- Not used	63 - 80
Card 2 - Gamma Heating Coefficients (10X, 2X, A5, 2F10.2)		
	- Not used	1 - 10
	- Not used	11 - 12
DENT	- Capsule location in EBR-II	13 - 17
GHF	- Gamma heating of fuel (W/g)	18 - 27
GHC	- Gamma heating of capsule (W/g)	28 - 37
	- Not used	38 - 80
Card 3 - Axial Description (13F5.2)		
DIST	- Relative axial position for determining local flux factors	1 - 65
	- Not used	66 - 80

PART I (contd)Columns

Card 4 - Flux Factors (13F5.2)

FAC ~ Ratio of local flux factors to peak for 1 - 65
 average cycle

 ~ Not used 66 - 80

Card 5 - Irradiation Data (10X, F5.1, F6.0, F7.3)

 ~ Not used 1 - 10

RP ~ Reactor power (MW) 11 - 15

T ~ Irradiation time (days) 16 - 21

FLORTV ~ Reactor coolant flow rate (gpm) 22 - 28

 ~ Not used 29 - 80

Card 6 - Length Measurements (10X, 6F10.2)

 ~ Not used 1 - 10

XLU35 ~ Length of ^{235}U (cm) 11 - 20

XLU38 ~ Length of ^{238}U (cm) 21 - 30

XLPU ~ Length of Pu (cm) 31 - 40

XLFU ~ Length of fuel (cm) 41 - 50

XLCA ~ Length of capsule (cm) for gamma heating 51 - 60

XLSPEC ~ Length of specimen (cm) 61 - 70

 ~ Not used 71 - 80

Card 7 - Radial Measurements (10X, 5F10.3)

 ~ Not used 1 - 10

R1 ~ Fuel radius (cm) 11 - 20

R3 ~ Inner cladding radius (cm) 21 - 30

R4 ~ Outer cladding radius (cm) 31 - 40

R5 ~ Inner capsule radius (cm) 41 - 50

R6 ~ Outer capsule radius (cm) 51 - 60

 ~ Not used 61 - 80

Card 8 - Diameter Measurement (10X, 6F7.3)

 ~ Not used 1 - 10

D1 ~ Cladding ID (cm) 11 - 17

D2 ~ Restrainer rod OD (cm) 18 - 24

D3 ~ Fuel OD (cm) 25 - 31

PART I Card 8 (contd)Columns

D4	- Cladding OD (cm)	32 - 38
D5	- Capsule ID (cm)	39 - 45
D6	- Capsule OD (cm)	46 - 52
	- Not used	53 - 80
	Card 9 - Thermal Conductivity, Coefficients (10X, 5F7.3, E10.3)	
	- Not used	1 - 10
CWK	- K of capsule (cal/sec-cm-°C)	11 - 17
CB	- Contact resistance of sodium interface (cal/sec-cm ² -°C)	18 - 24
COK	- K of coolant (cal/sec-cm-°C)	25 - 31
CLK	- K of cladding (cal/sec-cm-°C)	32 - 38
CA	- Contact resistance of fuel to clad (cal/sec-cm ² -°C)	39 - 45
FUK	- K of fuel (cal/sec-cm-°C)	46 - 55
	- Not used	56 - 80
	Card 10 - Weight Measurements (8X, 2X, A5, 5F10.2)	
	- Not used	1 - 8
	- Not used	9 - 10
DNAME	- Capsule identification	11 - 15
GMU35	- Weight of ²³⁵ U (grams)	16 - 25
GMU38	- Weight of ²³⁸ U (grams)	26 - 35
GMPU	- Weight of Pu (grams)	36 - 45
GMFU	- Weight of Fuel (grams)	46 - 55
GMCAP	- Weight of capsule (grams) for gamma heating	56 - 65
	- Not used	66 - 80
	Card 11 - Case Card (I5)	
K	- Case number (0 or blank = continue, 1 = end)	1 - 5
	- Not used	6 - 80

PART II - STRESS CALCULATIONSColumns

Card 1 - Irradiation Data (10X, F5.1, F6.0, F7.3)		
	- Not used	1 - 10
RP	- Reactor power (MW)	11 - 15
T	- Irradiation time (days)	16 - 21
FLORTV	- Reactor coolant flow rate (gpm)	22 - 28
	- Not used	29 - 80
Card 2 - Gas Ratio (10X, 3F10.4, F5.2)		
	- Not used	1 - 10
YPU39	- Factor for determining number of fissions that produce gas	11 - 20
YU235	- Factor for determining number of fissions that produce gas	21 - 30
YU238	- Factor for determining number of fissions that produce gas	31 - 40
PR	- Ratio of gas released to gas generated	41 - 45
	- Not used	46 - 80
Card 3 - Temperature Descriptions (10X, 8F5.0)		
	- Not used	1 - 10
TE	- Temperatures at which information as to capsule conditions are calculated (°C)	11 - 50
	- Not used	51 - 80
Card 4 - Percentage Factors (10X, 11F5.1)		
	- Not used	1 - 10
FAC	- Ratio of height of sodium after loss from leak in capsule to original height of sodium	11 - 65
	- Not used	66 - 80
Card 5 - Fission Rates (10X, 2X, A5, 3E15.4)		
	- Not used	1 - 10
	- Not used	11 - 12
DENT	- Capsule location in EBR-II	13 - 17
FRU35	- Fission rate of ^{235}U (fissions/g-sec)	18 - 32

PART II Card 5 (contd)Columns

FRU38	- Fission rate of ^{238}U (fissions/g-sec)	33 - 47
FRPU	- Fission rate of Pu (fissions/g-sec)	48 - 62
	- Not used	63 - 80
Card 6 - Weight Measurements (9X, 2X, A5, 5F10.2)		
	- Not used	1 - 8
	- Not used	9 - 10
DNAME	- Capsule identification	11 - 15
GMU35	- Weight of ^{235}U (g)	16 - 25
GMU38	- Weight of ^{238}U (g)	26 - 35
GMPU	- Weight of Pu (g)	36 - 45
GMFU	- Weight of fuel (g)	46 - 55
GMCAP	- Weight of capsule (g) for gamma heating	56 - 65
	- Not used	66 - 80
Card 7 - Diameter Measurements (10X, 6F7.3)		
	- Not used	1 - 10
D1	- Cladding ID (cm)	11 - 17
D2	- Restrainer rod OD (cm)	18 - 24
D3	- Fuel OD (cm)	25 - 31
D4	- Cladding OD (cm)	32 - 38
D5	- Capsule ID (cm)	39 - 45
D6	- Capsule OD (cm)	46 - 52
	- Not used	53 - 80
Card 8 - Other Measurements (10X, 5F10.3)		
	- Not used	1 - 10
XLP	- Plenum length (cm)	11 - 20
XLS	- Spring length (cm)	21 - 30
PE	- Packing efficiency ($\% \times 10^{-2}$)	31 - 40
VS	- Volume of spring (cc)	41 - 50
RTVOL	- Volume of sodium in capsule (cc)	51 - 60
	- Not used	61 - 80

<u>PART II</u> (contd)	<u>Columns</u>
Card 9 - Other Temperatures (10X, .5F7.1, F6.0)	
Not used	1 - 10
T7 Cladding temperature at top of fuel column ($^{\circ}\text{C}$)	11 - 17
DT6M Maximum cladding ΔT	18 - 24
DT6T Cladding ΔT at top of fuel column	25 - 31
DT2M Maximum capsule wall ΔT , ($^{\circ}\text{C}$)	32 - 38
DT2T Capsule wall ΔT at top of fuel column ($^{\circ}\text{C}$)	39 - 45
TG Temperature when capsule leaks ($^{\circ}\text{K}$)	46 - 51
Not used	52 - 80
Card 10 - Other Measurements (10X, 4F10.2)	
Not used	1 - 10
VOLSP Volume of specimen (cc)	11 - 20
XLFU Length of fuel (cm)	21 - 30
XLCA Length of capsule cavity (cm)	31 - 40
XLSPEC Length of specimen (cm)	41 - 50
Not used	51 - 80
Card 11 - E, α and μ for Capsule and Cladding (10X, 4E10.3, 2F5.2)	
Not used	1 - 10
E1 E of cladding (psi)	11 - 20
E2 E of capsule (psi)	21 - 30
CTE1 α of cladding (in./in./ $^{\circ}\text{C}$)	31 - 40
CTE2 α of capsule (in./in./ $^{\circ}\text{C}$)	41 - 50
U1 μ of cladding (c)	51 - 55
U2 μ of capsule (c)	56 - 60
Not used	61 - 80
Card 12 - Case Card (10X, II0)	
Not used	1 - 10
K Case number (0 or blank = continue, 1 = end)	11 - 20
Not used	21 - 80

PART III - DECAY HEAT AND CURIE CONTENTColumns

	Card 1 - Cooling Time (10X, 5F5.0)	
	- Not used	1 - 10
TC(I)	- Cooling time (days)	11 - 35
	- Not used	36 - 80
	Card 2 - Energy Group (10X, 6F5.1)	
	- Not used	1 - 10
ENGRP(N)	- Energy group (MeV/photon)	11 - 40
	- Not used	41 - 80
	Card 3 - Activity Value (10X, 6F5.2)	
	- Not used	1 - 10
VAL1(N)	- Energy group activity value after 1 day of cooling (C)	11 - 40
	- Not used	41 - 80
	Card 4 - Activity Value (10X, 6F5.2)	
	- Not used	1 - 10
VAL7(N)	- Energy group activity value after 7 days of cooling (C)	11 - 40
	- Not used	41 - 80
	Card 5 - Activity Value (10X, 6F5.2)	
	- Not used	1 - 10
VAL14(N)	- Energy group activity value after 14 days of cooling (C)	11 - 40
	- Not used	41 - 80
	Card 6 - Activity Value (10X, 6F5.3)	
	- Not used	1 - 10
VAL30(N)	- Energy group activity value after 30 days of cooling (C)	11 - 40
	- Not used	41 - 80
	Card 7 - Other Data (10X, 2X, A5, 5X, A5, F10.0, 5X, F10.1)	
	- Not used	1 - 10
	- Not used	11 - 12

PART III Card 7 (contd)Columns

DNAME	~ Capsule identification	13 - 17
	~ Not used	18 - 22
DENT	~ Capsule location in EBR-II	23 - 27
T	~ Irradiation time (days)	28 - 37
	~ Not used	38 - 42
P	~ Average element power (Watts)	43 - 52
K	~ Case number (0 or blank = continue, 1 = end)	53 - 57
	~ Not used	58 - 80

PART IV - LOSS OF COOLANT

Card 1 ~ Fission Rates (1H1, 9X, 2X, A5, 3E15.4)

	~ Not used	1 - 1
	~ Not used	2 - 10
	~ Not used	11 - 12
DENT	~ Capsule location in EBR-II	13 - 17
FRU35	~ Fission rate of ^{235}U (fissions/g-sec)	18 - 32
FRU38	~ Fission rate of ^{238}U (fissions/g-sec)	33 - 47
FRPU	~ Fission rate of Pu (fissions/g-sec)	48 - 62
	~ Not used	63 - 80

Card 2 ~ Gamma Heating Coefficients

(10X, 2X, A5, 2F10.2)

	~ Not used	1 - 10
	~ Not used	11 - 12
DENT	~ Capsule location in EBR-II	13 - 17
GHF	~ Gamma heating of fuel (Watts/g)	18 - 27
GHC	~ Gamma heating of capsule (Watts/g)	28 - 37
	~ Not used	38 - 80

Card 3 ~ Axial Description (13 F5.2)

DIST	~ Relative axial position for determining local flux factors	1 - 65
	~ Not used	66 - 80

PART IV (contd)Columns

	Card 4 - Flux Factors (13 F5.2)	
FAC	- Ratio of local flux factors to peak for average cycle	1 - 65
	- Not used	66 - 80
	Card 5 - Time After Scram (10X, 12F5.1)	
	- Not used	1 - 10
TYMAS	- Time after scram (sec) 1-12	11 - 70
	- Not used	71 - 80
	Card 6 - Time After Scram (10X, 4F5.1, 2F6.1)	
	- Not used	1 - 10
TYMAS	- Time after scram (sec) 13-18	11 - 42
	- Not used	43 - 80
	Card 7 - Power Factor (10X, 6F5.3)	
	- Not used	1 - 10
CNTPWR	- Fraction of maximum power (C) 1-12	11 - 70
	- Not used	71 - 80
	Card 8 - Power Factor (10X, 6F5.3)	
	- Not used	1 - 10
CNTPWR	- Fraction of maximum power (C) 13-18	11 - 40
	- Not used	41 - 80
	Card 9 - Flow Factor (9X, F6.3)(11F5.3)	
	- Not used	1 - 9
CNTFLO	- Fraction of maximum coolant flow (C) 1-12	10 - 70
	- Not used	71 - 80
	Card 10 - Flow Factor (10X, 6F5.3)	
	- Not used	1 - 10
CNTFLP	- Fraction of maximum coolant flow (C) 13-18	11 - 40
	- Not used	41 - 80
	Card 11 - Irradiation Data (10X, F5.1, F6.0, F7.3)	
	- Not used	1 - 10
RP	- Reactor power (MW)	11 - 15

PART IV Card 11 (contd)Columns

T	= Irradiation time (days	16 - 21
FLORTV	= Reactor coolant flow rate (gpm)	22 - 28
	= Not used	29 - 80
Card 12 - Length Measurements (10X, 6F10.2)		
	= Not used	1 - 10
XLU35	= Length of ^{235}U (cm)	11 - 20
XLU38	= Length of ^{238}U (cm)	21 - 30
XLPU	= Length of Pu (cm)	31 - 40
XLFU	= Length of fuel (cm)	41 - 50
XLCA	= Length of capsule (cm) for gamma heating	51 - 60
XLSPEC	= Length of specimen (cm)	61 - 70
	= Not used	71 - 80
Card 13 - Radial Measurements (10X, 5F10.3)		
	= Not used	1 - 10
R1	= Fuel radius (cm)	11 - 20
R3	= Inner cladding radius (cm)	21 - 30
R4	= Outer cladding radius (cm)	31 - 40
R5	= Inner capsule radius (cm)	41 - 50
R6	= Outer capsule radius (cm)	51 - 60
	= Not used	61 - 80
Card 14 - Diameter Measurement (10X, 6F7.3)		
	= Not used	1 - 10
D1	= Cladding ID (cm)	11 - 17
D2	= Restrainer rod OD (cm)	18 - 24
D3	= Fuel OD (cm)	25 - 31
D4	= Cladding OD (cm)	32 - 38
D5	= Capsule ID (cm)	39 - 45
D6	= Capsule OD (cm)	46 - 52
	= Not used	53 - 80
Card 15 - Thermal Conductivity Coefficient (10X, 5F7.3, E10.3)		
	= Not used	1 - 10
CWK	= K of capsule (cal/sec-cm-°C)	11 - 17

PART IV Card 15 (contd)

CB	- Contact resistance of sodium interface (cal/sec-cm ² -°C)	18	-	24
COK	- K of coolant (cal/sec-cm-°C)	25	-	31
CLK	- K of cladding (cal/sec-cm-°C)	32	-	38
CA	- Contact resistance of fuel to clad (cal/sec-cm ² -°C)	39	-	45
FUK	- K of fuel (cal/sec-cm-°C)	46	-	55

Card 16 - Weight Measurements (8X, 2X, A5,
5F10.2)

	- Not used	1	-	8
	- Not used	9	-	10
DNAME	- Capsule identification	11	-	15
GMU35	- Weight of ²³⁵ U (g)	16	-	25
GMU38	- Weight of ²³⁸ U (g)	26	-	35
GMPU	- Weight of Pu (g)	36	-	45
GMFU	- Weight of fuel (g)	46	-	55
GMCAP	- Weight of capsule (g) for gamma heating	56	-	65
	- Not used	66	-	80

Card 17 - Case Card (I5)

K	- Case number (0 or blank = continue, 1 = end)	1	-	5
	- Not used	6	-	80

PART V - UNCERTAINTY FACTOR CALCULATIONS

Card 1 - Fission Rates (1H1, 9X, 2X, A5, 3E15.4)

	- Not used	1	-	1
	- Not used	2	-	10
	- Not used	11	-	12
DENT	- Capsule location in EBR-II	13	-	17
FRU35	- Fission rate of ²³⁵ U (fissions/g-sec)	18	-	32
FRU38	- Fission rate of ²³⁸ U (fissions/g-sec)	33	-	47
FRPU	- Fission rate of Pu (fissions/g-sec)	48	-	62
	- Not used	63	-	80

PART V (contd)Card 2 - Gamma Heating Coefficients (10X, 2X, A5,
2F10.2)

	Not used	1	10
	Not used	11	12
DENT	Capsule location in EBR-II	13	17
GHF	Gamma heating in fuel (W/g)	18	27
GHC	Gamma heating in capsule (W/g)	28	37
	Not used	38	80

Card 3 - Axial Description (11F6.3)

DENT	Relative axial position for determining local flux factors	1	66
	Not used	67	80

Card 4 - Flux Factors (11F6.3)

FAC	Ratio of local factors to peak for average cycle	1	66
	Not used	67	80

Card 5 - Irradiation Data (10X, F5.2, F6.0, F7.3)

	Not used	1	10
RP	Reactor power (MW)	11	15
T	Irradiation time (days)	16	21
FLORTV	Reactor coolant flow rate (gpm)	22	28
	Not used	29	80

Card 6 - Length Measurements (10X, 6F10.2)

	Not used	1	10
XLU35	Length of ^{235}U (cm)	11	20
XLU38	Length of ^{238}U (cm)	21	30
XLPU	Length of Pu (cm)	31	40
XLFU	Length of fuel (cm)	41	50
XLCA	Length of capsule (cm) for gamma heating	51	60
XLSPEC	Length of specimen (cm)	61	70
	Not used	71	80

PART V (contd)

Card 7 - Radial Measurements (10X, 5F10.3)		
	- Not used	1 - 10
R1	- Fuel radius (cm)	11 - 20
R3	- Inner cladding radius (cm)	21 - 30
R4	- Outer cladding radius (cm)	31 - 40
R5	- Inner capsule radius (cm)	41 - 50
R6	- Outer capsule radius (cm)	51 - 60
	- Not used	61 - 80
Card 8 - Diameter Measurements (10X, 6F7.3)		
	- Not used	1 - 10
D1	- Cladding ID (cm)	11 - 17
D2	- Restrainer rod OD (cm)	18 - 24
D3	- Fuel OD (cm)	25 - 31
D4	- Cladding OD (cm)	32 - 38
D5	- Capsule ID (cm)	39 - 45
D6	- Capsule OD (cm)	46 - 52
	- Not used	53 - 80
Card 9 - Thermal Conductivity Coefficients (10X, 6F7.3)		
	- Not used	1 - 10
CWK	- K of capsule (cal/sec-cm-°C)	11 - 17
CB	- Contact resistance of sodium interface (cal/sec-cm ² -°C)	18 - 24
COK	- K of coolant (cal/sec-cm-°C)	25 - 31
CLK	- K of cladding (cal/sec-cm-°C)	32 - 38
CA	- Contact resistance of fuel to clad (cal/sec-cm ² -°C)	39 - 45
FUK	- K of fuel (cal/sec-cm-°C)	46 - 52
	- Not used	53 - 80
Card 10 - Uncertainty Factor (16F4.2)		
UCTY	- Percent of uncertainty (% x 10 ⁻²)	1 - 64
	- Not used	65 - 80

PART V (contd)

Card 11 - Weight Measurements (8X, 2X, A5, 5F10.2)

	Not used	1 - 8
	Not used	9 - 10
DNAME	Capsule identification	11 - 15
GMU35	Weight of ^{235}U (g)	16 - 25
GMU38	Weight of ^{238}U (g)	26 - 35
GMPU	Weight of Pu (g)	36 - 45
GMFU	Weight of fuel (g)	46 - 55
GMCAP	Weight of capsule (g) for gamma heating	56 - 65
	Not used	66 - 80

PART VI - RADIAL TEMPERATURE PROFILE - SPECIMEN AND IRRADIATION
CAPSULE COMPONENTS

Card 1 - Heat Generation (7F6.1)

Q1	Heat generation due to fission and gamma heat in the fuel (cal/sec-cm)	1 - 6
Q2	Gamma heating in cladding (cal/sec-cm)	7 - 12
Q3	Q1 + Q2 (cal/sec-cm)	13 - 18
Q4	Gamma heating in coolant (cal/sec-cm)	19 - 24
Q5	Q3 + Q4 (cal/sec-cm)	25 - 30
Q6	Gamma heating in capsule (cal/sec-cm)	31 - 36
Q7	Q5 + Q6 (cal/sec-cm) "film drop"	37 - 42
	Not used	43 - 80

Card 2 - Radial Measurements (5F6.3)

R1	Fuel radius (cm)	1 - 6
R2	Inner cladding radius (cm)	7 - 12
R3	Outer cladding radius (cm)	13 - 18
R4	Inner capsule radius (cm)	19 - 24
R5	Outer capsule radius (cm)	25 - 30
	Not used	31 - 80

PART IV (contd)

	Card 3 - Outer Data (5X, A5, 5X, A5, F10.1, I5)	
	- Not used	1 - 5
Cap	- Capsule identification	6 - 10
	- Not used	11 - 15
SPEC	- Capsule location in EBR-II	16 - 20
T1	- Initial bulk reactor coolant temperature (°C)	21 - 30
K	- Case number (0 or blank = continue, 1 = end)	31 - 35
	- Not used	36 - 80

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