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HAZARDS ANALYSIS
FOR THE BATTELLE-NORTHWEST EBR-II/TREAT
TRANSIENT IRRADIATION TEST SERIES

G. E. Culley and D. O. Sheppard

June 1970

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Engineering Drawing H-3-29368, Inner Capsule

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HAZARDS ANALYSIS FOR THE
BATTELLE-NORTHWEST
EBR-II/TREAT TRANSIENT IRRADIATION TEST-SERIES

By

G. E. Culley

D. O. Sheppard

FFTF Fuel Department
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June 1970

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ABSTRACT

This document describes the experimental equipment used and contains an analysis of potential hazards involved in the EBR-II/TREAT series of transient irradiations to be conducted in support of the Fast Flux Test Facility (FFTF) fuel development program. Approximately sixteen transient irradiations in TREAT will be conducted on prototypic FFTF fuel pins previously irradiated in EBR-II to **burnups** of 10,000 and 50,000 MWd/MTM. A hazards analysis was conducted on the maximum **burnup** fuel pin for both the expected and maximum accident transient conditions.

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HAZARDS ANALYSIS FOR THE
BATTELLE-NORTHWEST
EBR-II/TREAT TRANSIENT IRRADIATION TEST SERIES

1.0 INTRODUCTION

A series of transient tests will be conducted on prototype FTR (Fast Test Reactor) fuel pins in support of the FFTF driver fuel development program, to evaluate the pins response and failure thresholds for the postulated accident conditions shown in Table 1.1. The range of testing will include the thresholds between (a) the minor and major accidents and (b) the major and disruptive accidents. The transient tests will be designed to produce the appropriate degree of fuel melting and cladding temperature. The fuel pin response will be evaluated to determine if they meet plant safety criteria. This evaluation information will be used in the safety analysis reports required for construction and operation of the FFTF.

The overall BNW-FFTF transient testing program includes both pre-irradiated and nonirradiated fuel pins. Pre-irradiated pins will have been irradiated in either GETR or EBR-II. GETR-irradiated pins closely approximate the FTR's fuel length while EBR-II-irradiated pins (13-1/2 in. fuel column), more closely approximate burnup in the FTR's neutron flux spectrum.

The EBR-II/TREAT test series concern only those fuel pins pre-irradiated in EBR-II. The irradiated, mixed-oxide fuel pins will be de-encapsulated, nondestructively examined, and re-encapsulated. The pins will then be inserted into a testing capsule, transient irradiated, and finally nondestructively and destructively examined.

The hazards analysis presented for the EBR-II/TREAT series considers the test fuel pin and test conditions that will produce the most severe results.

TABLE 1.1. FTR Transient Overpower Damage Levels

Accident Severity Level	Damage to Fuel Matrix		Damage to Cladding	
	Damage	Threshold (Areal % Melting)	Damage	Threshold (Peak Temp.)
Disruptive Accident	Fuel Vaporization and Relocation Within Core	New Fuel (Unirradiated) Old-Fuel (Pre-Irradiated)	Cladding Ruptures With Violent Release of Fuel	
(B)		70% 50%		1480 °F
Major Accident	Gross Fuel Relocation and Segregation Within Cladding		Cladding Ruptures Releasing Fission Gas With Some Non-Violent Release of Fuel	
(A)		20%		1280 °F
Minor Accident	Some Local Fuel Migration Within Pin		No Loss of Cladding Integrity But Some Deformation	
		0		1200 °F
Operational Accident	No Fuel Melting or Migration		No Cladding Damage or Loss Of Continued Operational Capability	
Normal Operating Zone	Normal Fuel Restructuring, Fission Gas Release and Swelling		Normal Deformation Due to Steady-State Operation	1100 °F

1.2

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2.0 DESCRIPTION OF EXPERIMENTAL EQUIPMENT

2.1 FUEL PIN DESIGN

The fuel pins to be transient irradiated in TREAT have been pre-irradiated in EBR-II in subassemblies X031 (PNL-1) and X032 (PNL-2). Figure 2.1 shows the fuel pin and its sodium-bonded capsule used in EBR-II. The engineering drawing of the fuel pin (H-3-27688) is shown in Appendix A. After the EBR-II irradiation, the sodium-bonded capsule is removed, and the fuel pin is cleaned, gamma scanned, and profilometered. It is then re-encapsulated in an inner capsule called the EBR-II/TREAT capsule (discussed in Section 2.2).

The fuel material is a mixed oxide composed of 25 wt% PuO_2 and 75 wt% UO_2 containing 93% enriched uranium. The fuel is in the form of pressed and sintered solid and annular pellets, 0.25 in. long, 0.212 in. in diameter, at 93% of theoretical density. Nominal weight of the fuel is 80 grams in a column 13.5 in. long. Several of the fuel columns contain axial fuel motion restrictors which are 118-in. thick Type 304 SS wafers 0.215 in. in diameter with axial grooves on the periphery. Fuel cladding material is Type 304 SS 0.250 in. diameter with a 0.016 in. wall thickness. Nominal fuel-to-cladding cold diametral gap is 0.006 in. to yield a planar smeared density of approximately 88% TD. The as-fabricated smeared densities were somewhat less. Located directly above the fuel column is a spring-loaded extensometer. The cladding end closures were made by TIG welding Type 304 SS end caps in place. The internal fuel pin atmosphere is helium plus fission gases released during steady-state irradiation.(1) Table 2.1 shows the as-fabricated fuel parameters and steady-state irradiation conditions for EBR-II/TREAT test fuel pins.

FIGURE 2.1. Pre-Irradiated EBR-II/TREAT Test Fuel Pin and Capsule

TABLE 2.1. PNL 1 and 2 EBR-II Fuel Pins for TREAT Experiments

<u>PNL Capsule #</u>	<u>Fuel Pellet Density, % TD</u>	<u>Fuel Smeared Density, % TD</u>	<u>Fuel Weight, g</u>	<u>Pellet Hole Dia., in (Nominal)</u>	<u>Axial Motion Restrictor</u>	<u>Peak Steady State Power, kW/ft</u>	<u>Peak Burnup Mwd/MTM</u>
1-1	91.74	86.58	78.90	--	No	8.65	8,200
1-2	92.95	87.47	80.53	--	No	8.85	7,850
1-4	89.38	84.53	77.84	--	Yes	9.60	8,520
1-5	93.80	84.45	78.11	--	Yes	8.65	8,200
1-7	95.45	84.24	76.80	0.054	No	8.85	8,850
1-8	95.54	83.89	77.30	0.054	No	9.60	8,520
1-10	89.67	79.33	72.18	0.054	Yes	8.85	8,850
1-11	89.45	79.12	72.00	0.054	Yes	10.37	9,200
2-1	93.09	87.60	80.65	--	No	9.06	41,250
2-2	91.80	86.38	79.53	--	No	9.00	41,750
2-4	89.38	84.40	77.95	--	Yes	8.68	38,750
2-5	89.50	84.57	78.04	--	Yes	8.25	38,750
2-7	94.94	83.82	76.40	0.054	No	9.31	44,250
2-8	95.15	83.91	76.57	0.054	No	9.85	44,250
2-10	89.55	79.21	72.92	0.054	Yes	8.65	43,500
2-11	89.24	78.95	72.00	0.054	Yes	8.19	41,250

IN ALL CASES:

Fuel Column Length, in.	13.5
Fuel Pellet O.D., in.	0.212
Fuel Cladding Dia. Gap, in.	0.006
Cladding Diameter, in.	0.250
Cladding Thickness, in.	0.016

2.2 INNER CAPSULE DESIGN

The fuel pin is to be contained in a NaK-bonded, instrumented capsule as shown in Figure 2.2. The engineering drawing of this capsule (H-3-29368) appears in Appendix A.

The capsule was designed to:

- Provide an instrumented vehicle for the pre-irradiated fuel pin.
- Permit remote insertion of the fuel pin with subsequent NaK filling.
- Be compatible with the existing BNW-TREAT capsules.

The inner capsule design concept is based on that design proved in the G.E.-PA-10 program Task C transient tests. (2,3)

The capsule is constructed of Type 304 SS tubing with a 1-1/8 in. OD and a 0.049 in. wall. It contains an annular nickel thermal dam bonded to the fuel pin with eutectic NaK alloy. The capsule is instrumented with six Chromel-Alumel thermocouples spatially arranged as shown in Figure 2.2. Three thermocouple hot junctions are located in the NaK annulus between the fuel pin and nickel thermal dam, and the other three are located in the NaK annulus between the thermal dam and the capsule wall. The fuel pin will be attached at the bottom, have 10 in. of axial clearance and 0.090 in. of diametral clearance in the thermal dam. The overall capsule length is 63-7/8 in. with a maximum diameter of 1-3/4 in. at the instrumentation connector.

2.3 TREAT CAPSULE DESIGN

Cutaway drawing of the BNW-TREAT capsule is shown in Figure 2.3. This capsule was designed to replace the central fuel element in TREAT and to provide an interface between the inner capsule and the TREAT reactor for transient irradiation of mixed oxide fuel pins. The capsule consists of a cylindrical pressure vessel equipped with a TREAT gripping fixture

FIGURE 2.2. EBR-II/TREAT Inner Capsule Assembly

FIGURE 2.3. TREAT Capsule Assembly

at its upper end and a fuel element guide pin fitting at its lower end. The exterior of the capsule has the same dimensions as a TREAT driver element except that the central portion is round in cross section rather than square. The engineering drawing of this capsule (H-3-27729) is shown in Appendix A.

The primary design criteria for this capsule is that as a third and final containment barrier it must restrain the consequences of a gross failure of both the fuel pin and the inner capsule. The capsule is constructed of 3 in. OD by 0.250 in. wall, heat-treated 4130 steel tube with welded and heat-treated 4130 steel end fittings. The welds were subjected to X-radiography and dye penetrant testing and the completed capsule hydrostatically tested to 15,000 psi internal pressure. The inside of the capsule is lined with graphite to prevent molten materials from contacting the pressure vessel in the case of violent failure of both the fuel pin and inner capsule. An electric heating element thermally insulated from the pressure vessel is used to provide the desired pretransient temperature in the test capsule. Six thermocouples are incorporated in the heater can to monitor temperatures during the preheat period. A connector is provided to mate with the connector on the inner capsule, and electrical leads from both the heater can and inner capsule pass through a bulkhead sealed with epoxy resin. The capsule closure consists of a bolted flange with a pressure-assisted, plated, metal O ring seal.

Operating experience with capsules of similar design has been obtained under the Task C transient irradiations of the G.E.-conducted PA-10 program which shows the design to be adequate for the transient testing of mixed oxide fuel pins.⁽³⁾

2.4 EXPERIMENT INSTRUMENTATION

The instrumentation incorporated in these experiments is provided to:

- Measure and control the pre-transient temperature. Provide a history of inner capsule temperatures during the transient.
- Measure the post-transient equilibrium temperature of the inner capsule.

The failure of any instrumentation will not affect the test performance or safety features; however, such a failure could result in loss of useful information for test evaluation. The instrumentation consists of twelve stainless steel-sheathed Chromel-Alumel thermocouples. Six thermocouples are located within the inner capsule and six in the heater can in such a way that they rest against the outer surface of the inner capsule.

3.0 HAZARDS ANALYSIS

3.1 NUCLEAR EFFECTS ON REACTOR PERFORMANCE

The nuclear Effects of performing EBR-II/TREAT experiments in the TREAT reactor were evaluated using the computer codes HRG,⁽⁴⁾ BATTELLE-REVISED-THERMOS,⁽⁵⁾ and DTF-IV.⁽⁶⁾ Because the capsule design is similar to the capsules used in the G.E.-PA-10 transient testing program (Series II and III in particular ^(2,3)), their effect upon reactor performance is expected to be similar. The knowledge obtained from the G.E. tests can be directly applied to the EBR-II/TREAT series of tests with a high degree of confidence.

Two neutron energy groups were used in the analysis since a significant portion of the power generated in the highly enriched test pin is due to nonthermal, neutron-induced fissions, as shown in Figure 3.1. This figure illustrates the radial power profile within the test fuel and shows that the nonthermal contribution accounts for approximately 70% of the power generated in the center of the pin.

Figure 3.2 shows the radial neutron flux perturbation in the TREAT core at the axial midplane caused by the presence of the EBR-II/TREAT capsule. The worth of the inner control rods located at a radius of 44-1/2 cm is decreased by only 5%. The outer control rods are not significantly affected. The worth of the EBR-II/TREAT capsule as compared to a standard driver element was calculated to be -2.7% $\Delta K/K$.

A calibration experiment utilizing the EBR-II/TREAT capsule geometry was performed in TREAT to determine the relationship between the test pin power and the reactor power. The results showed that this ratio is 1.00×10^{-4} W per cm³ of test fuel per watt of reactor power for nonirradiated fuel. For test pins irradiated to 50,000 MWd/MTM it was calculated that this ratio would be reduced by 3%. The fuel pin contains

FIGURE 3.1. Radial Power Distribution
in EBR-II/TREAT Fuel

FIGURE 3.2. Effect of EBR-II/TREAT Experiment Upon Radial Flux Distribution in TREAT

approximately 8 cm³ of fuel and therefore produces only 0.08% of the total power generated by the reactor. Since the experiment produces such a small portion of the total power and the majority of the test capsule materials will not undergo a large temperature change during the transient, there is not expected to be a significant effect upon the temperature coefficient of reactivity. The effect will be similar to that produced by the G.E. experiments previously performed over a wide range of reactor powers.

The nature of transient testing introduces the possibility of the fuel being rearranged within the test capsule. (See paragraph 3.3). Gross melting and movement of the fuel under the forces of gravity and fission gas pressure to the bottom of the inner capsule not only locates the fuel in a lower worth position in the reactor, but due to the self-shielding effect produces a fuel configuration that is worth less than the pin form. If this type of fuel movement were to occur during the transient it would not present any neutronic hazard to the reactor. The other extreme would be the idealized condition in which all the materials, and particularly the fuel, become homogenized within the inner capsule wall from the melting of all internal components. The analysis of this condition indicates that the experiment worth would increase by 0.4% $\Delta K/K$ upon homogenization. The increase is caused by the decrease in fuel density and associated reduction in the self-shielding effect.

Assuming a pessimistic mechanism whereby molten fuel released from the fuel pin completely melts and mixes with the nickel thermal dam, a step increase in effective reactivity would occur. If the homogenization were to take place before the negative temperature coefficient could feed back to the reactor, the maximum reactivity available would be the initial

reactor reactivity ($1.5\% \Delta K/K$ for the maximum accident case⁽⁸⁾) plus the reactivity upon homogenization ($0.4\% \Delta K/K$) or $1.9\% \Delta K/K$. A reactivity addition of this magnitude is well within the capacity of the TREAT reactor from the standpoint of maximum core temperature, peak reactor power, and integrated power.⁽⁹⁾

The likelihood of the homogenization process occurring is remote as was demonstrated in the C2C transient test in the G.E.-PA-10 program Series II transient. When molten fuel came in contact with the aluminum thermal dam (lower melting point and heat capacity than nickel) there was very little mixing of the fuel and aluminum even though the thermal dam was partially melted.⁽¹⁾ There was some slumping and relocation of the molten fuel, but a review of the reactor power versus time data did not reveal any significant effects upon the reactor performance.

It is concluded that the nuclear performance of the TREAT reactor containing the EBR-II/TREAT capsule will not be adversely changed during either the expected transient or one in which gross fuel relocation occurs.

3.2 EXPERIMENT RESPONSE WITH EXPECTED TRANSIENT

Several transient experiments are planned with various degrees of fuel melting for pins of two levels of steady state burnups. However, this discussion is concerned with the highest exposure pin and the maximum amount of fuel melting which will produce the most damaging conditions. The fuel pin data used for the calculations are from pin number PNL-2-8 which operated at 9.85 kW/ft peak power and achieved 44,250 MWd/MT exposure. The pre-irradiation physical characteristics of this pin are listed in Table 2.1. The post-irradiation characteristics were assumed from the destructive

analysis of PNL-1 fuel pins irradiated at the same conditions. The irradiated fuel (originally solid pellet) was characterized as having 0.015-in. diameter central void, an 0.080-in. diameter columnar grain region of 98% TD, with the remainder of the fuel being 93% TD and in contact with the cladding.⁽¹⁰⁾

The transient heat transfer program ARGUS⁽¹¹⁾ developed and modified⁽¹²⁾ at ANL was used to select a transient which would subject the fuel pin to the postulated FTR transient overpower case characterized by 50 areal% fuel melting (in a plane normal to the axis of the fuel pin) and a peak cladding temperature of 1480 °F. This is the defined threshold between a "major accident" and a "disruptive accident" shown in Table 1.1. The TREAT transient thus determined was used as the "expected transient" for the hazards analysis.

In addition to the fuel pin characteristics, the ARGUS input used a constant fuel-to-cladding gap coefficient of 2,000 Btu/hr-ft²-°F and a preheat temperature of 300 °F. The material properties in the test pin and capsule vary as a function of temperature. The nodal layout and material regions used to describe the test geometry are shown in Figure 3.3. TREAT transient No. 926 was found to produce the desired results on test pin PNL-2-8. Transient No. 926 had a peak power of 124 MW with an integrated power of 160.5 MW/sec as shown in Figure 3.4.

The temperature response of the fuel pin when subjected to TREAT transient No. 926, as predicted by ARGUS, is shown in Figure 3.5. The radial power profile of the pin (Figure 3.1) causes the maximum fuel temperature to first occur at a radius where r/r_0 equals approximately 0.5 and propagates inward. Ideally, the power profile would be flat to simulate a fast flux power profile, and melting would originate at the center and propagate outward. As shown in Figure 3.2, the fuel center

Q

1	2	3	4	5	6	7	8
VODD	FUEL 98% T.D.	FUEL 93% T.D.	CLAD 304 SS	NaK	Ni HEAT SINK	NaK	304 SS CAPSULE WALL

<u>Region</u>	<u>Radius, ft</u>	<u>Diameter, in.</u>	<u>No. of Nodes</u>	<u>Node Number</u>
1	6.250×10^{-4}	0.015	2	1-2
2	3.333×10^{-3}	0.080	6	3-8
3	9.083×10^{-3}	0.218	17	9-25
4	1.042×10^{-2}	0.250	3	26-28
5	1.417×10^{-2}	0.340	3	29-31
6	3.942×10^{-2}	0.946	11	32-42
7	4.275×10^{-2}	1.026	3	43-45
8	4.686×10^{-2}	1.125	3	46-48

FIGURE 3.3. EBR-II/TREAT Fuel Model for ARGUS

FIGURE 3.4. TREAT Transient Number 926

FIGURE 3.5. EBR-II/TREAT Fuel Pin Temperature Response
for TREAT Transient Number 926

temperature lags behind the peak fuel temperature by slightly more than 0.5 sec. The fuel center remains at the phase change temperature 4900 °F for 7 sec and is assumed to be melted. Figure 3.6 shows the maximum fuel temperature of 5170 °F occurring at a radius of 0.06 in. from centerline. The ARGUS program determined that slightly more than 50 areal% melting can be obtained from the expected transient with a maximum cladding temperature of 1480 °F. It is anticipated that the EBR-II/TREAT test with the expected transient will produce incipient fuel pin failure, as described in Table 1.1.

TREAT transient No. 926 was originally used on the G.E. test C3E involving a mixed oxide fuel pin irradiated to 65,000 MWd/MTM with the principal difference from the EBR-II/TREAT fuel pins being that the G.E. pin had 80% natural UO_2 instead of 75% fully enriched UO_2 in the mixed oxide fuel. The transient produced a 1300 °F maximum cladding temperature, a 4900 °F maximum fuel temperature and a 30 volume percent melting. The conclusions from the post-transient destructive examination are that no radial deformation or cracking of the cladding occurred, although there was some axial movement of the fuel.⁽³⁾ In comparison, the EBR-II/TREAT test to be performed with transient 926 is expected to develop higher power and temperatures due to the higher enrichment.

Because the test conditions may produce fuel pin failure, this aspect was studied as a part of the hazards evaluation using the computer code PECT-1.⁽¹³⁾ The code input utilized the cladding internal pressure and temperature history along with an assumed mode of failure to predict the time of fuel pin failure. The internal fuel pin pressure during the transient was determined from the amount of fission gas released by the fuel during melting and by the volume of the internal void. During steady-state operation it was assumed that the

EBR-II/TREAT TRANSIENT
FUEL MAXIMUM TEMPERATURE

FIGURE 3.6. Peak EBR-II/TREAT Fuel Temperatures
for Expected Transient

PNL-2-8 fuel released 80% of the generated fission gas to the plenum, based on data contained in Figure 3.7. The remaining fission gas entrapped in the fuel was assumed to be released when the fuel temperature reached the melting point during the transient. The fuel vapor pressure was ignored since the fuel reaches a maximum temperature of only 5170 °F, and the G.E Series II tests determined that fuel vapor pressure has an insignificant contribution to pin failure.⁽²⁾ The gas released during the transient was not allowed to expand into the plenum but was contained in the void within the fuel. This volume was based on the planar smeared density of the fuel pin, the differential thermal expansion between the fuel and the cladding, and the volume increase upon melting. (See Appendix C for detailed calculations.)

The internal fuel pin pressure-time relationship is shown in Figure 3.8, and the cladding temperature response is shown in Figure 3.9 for the expected transient. Cladding failure was assumed to occur when the cross section of the cladding at the axial peak power point on the fuel pin becomes plastic throughout. The PECT-1 analysis predicted fuel pin failure at 3.50 sec into the transient. The effective plastic diametral strain at failure was calculated to be 0.2% at the outer surface. This compares with burst test data of cladding sections from sibling PNL-1 fuel pins which exhibited failure strains between 0.2 and 18.4%.⁽¹⁴⁾

Assuming failure occurs at that point, 77% of the transient will have been completed, and the remaining 23% will generate 29 Btu in the fuel after the pin has failed. Assuming that all the heat given up by the fuel is instantly used to vaporize NaK in the inner capsule, a maximum pressure of 3210 psi will be generated within the inner capsule. This is less than the calculated 5500 psi containment capability

FIGURE 3.7. Fission Gas Release from Mixed Oxide DFR and APO Fuel Pins

FIGURE 3.8. Internal Fuel Pin Pressure for Maximum and Expected Transient

FIGURE 3.9. Cladding Temperature Versus Time for the Maximum and Expected Transient as Calculated by ARGUS

of the inner capsule. Assumptions and calculations are shown in Appendix C. Burst tests conducted on inner capsule specimens (including weld joints) have not been formally documented but have demonstrated internal pressure capabilities greater than 6000 psi at room temperature. (15)

It is concluded that by using pessimistic values for the fuel pin void volume and expected strain to failure, the fuel pin will fail during the expected transient and the inner capsule will contain the consequences of the pin failure.

3.3 MAXIMUM ACCIDENT CASE

As previously noted, normal conduct of the experiment may induce failure of the fuel pin cladding but should not cause failure of the inner capsule. In order to better evaluate the test assembly capabilities, more extreme conditions were postulated and analyzed. A maximum accident transient resulting from a reactivity insertion $0.3\% \Delta K/K$ greater than intended was provided by TREAT personnel. (8)

This maximum transient, shown in Figure 3.10, has a peak power of 510 MW with an integrated power of 345 MW/sec. The test pin response to this transient was evaluated with the ARGUS computer code, using the same fuel model and heat transfer assumptions as was used in Section 3.2 for the expected transient. Figure 3.11 shows the maximum temperatures calculated in the capsule, assuming that fuel pin integrity is maintained. Conclusions were that the fuel pin cladding will fail from excessive internal pressure before it could melt, and PECT-1 was used to determine when the failure would take place.

The internal pressure history for the fuel pin during the maximum transient was calculated using the same assumptions as in Section 3.2. Figure 3.8 shows the increase in internal

3.17

FIGURE 3.10 Maximum Accident Transient for EBR-II/TREAT Tests

FIGURE 3.11. Peak EBR-II/TREAT Capsule Temperatures for Maximum Accident Transient

pressure with time, and Figure 3.9 shows the associated cladding temperature history. Sample calculations are contained in Appendix D.

PECT-1 predicted failure of the cladding at 0.2% strain 2.15 sec after initiation of the transient, which leaves 70% of the transient power yet to be generated. Calculations of the amount of energy released by the fuel (328 Btu) after cladding failure are contained in Appendix D. Two approaches were taken to describe what happens following fuel pin failure.

The first approach assumes that molten fuel at 8500 °F is expelled to the bottom of the inner capsule and vaporizes all of the NaK contained in the inner capsule. Vaporization of the NaK requires 285 Btu and would cause the inner capsule pressure to increase to 20,200 psi if it did not fail. Calculations and assumptions are described in Appendix D. As explained in Section 3.2, the pressure containment capability of the inner capsule at 600 °F (wall temperature) is 5575 psi, and vaporization of all the NaK contained within will cause rupture. In this event, a mixture of molten fuel and NaK would be expelled into the TREAT capsule, causing the pressure in the capsule to increase to a maximum of 6500 psi. The TREAT capsule is capable of containing 15,000 psi and has a graphite liner to protect the capsule wall from contact with molten materials. The containment capabilities of the TREAT capsule from the standpoint of missile penetration and dynamic shock pressure loading were not analyzed here since the capsule and conditions are similar to those used in the G.E. tests. G.E. performed these analyses in their Series 2, 3 and 5 hazards evaluations^(16,17) and found that the missile problem is insignificant and that the dynamic shock pressure was within the containment capabilities of the TREAT capsule.

The second approach is based upon a more realistic evaluation of the post-failure process. This approach assumes that molten fuel is discharged from the axial center of the fuel pin at the hottest point. After vaporizing all of the NaK contained in the annulus between the fuel pin and the thermal dam, the fuel freezes in the annulus. The nickel thermal dam can conduct heat away from the fuel-nickel interface faster than it can get through the fuel. It has been shown experimentally⁽¹⁸⁾ that when molten fuel (still producing power) comes in contact with nickel, it solidifies before a significant portion of the nickel melts. Vaporization of all the NaK in the annulus requires only 15 Btu, causing the inner capsule pressure to increase to 1330 psi, which is within its containment capability. Assuming the remainder of the energy is transferred to the nickel thermal dam, its temperature would increase from 330 °F to 1480 °F, well below the 2300 °F melting point of nickel.

There is experimental evidence to indicate that the molten fuel-liquid metal interaction does not result in the extreme pressures assumed in the first approach. Work performed by Ivins⁽¹⁹⁾ has shown that oxide fuel pins which under severe transient irradiation expel molten fuel into the liquid metal bond (sodium in this case) do not produce extreme pressures within the capsule. The capsules were instrumented to detect pressure changes, and the maximum pressure pulse detected while performing five tests was 1800 psi. The G.E. tests also showed that pressures generated following fuel pin failure are within the containment capabilities of the inner capsule. In the C3C experiment, where the fuel temperatures approached 6000 °F, molten fuel expelled from the bottom of the fuel pin and collected in the bottom of the inner capsule (similar to the first approach assumption). No extreme pressure pulses were measured, and the inner capsule contained the consequences of the fuel pin failure.⁽³⁾

It is concluded that in the case of the maximum accident transient, fuel pin failure will occur early in the transient, and the inner capsule will remain intact. In the unlikely event that the inner capsule were to fail, the TREAT capsule would be more than adequate to contain the consequences.

3.4 HANDLING OF EXPERIMENTAL EQUIPMENT

The TREAT capsule containing the radioactive test pin will be shipped to and from the TREAT facility in the T-2 shipping cask. In order to facilitate horizontal loading of the capsule into the cask at BNW, a cask liner and capsule alignment clamp will be used as shown in Figure 3.12. This configuration will permit both horizontal loading at BNW and vertical handling at TREAT. The alignment clamp remains on the capsule and will not interfere with the loading into the reactor. (8)

While handling the capsule, care should be taken to not drop, bump, or otherwise jar the assembly, particularly in the longitudinal direction. Such mishandling could cause the inner capsule electrical connector to become disconnected. The capsule should remain in either the horizontal or vertical position (gripping fixture up) and must not be inverted. The vertical position is recommended for storage.

The chances of damage from handling causing a radioactive hazard are remote since the fuel has triple containment. However, damage such as fuel relocation prior to the test may produce errant response to the transient. Therefore, before they are inserted into the TREAT reactor, the capsules will be neutron radiographed to assess the condition of the internal* components.

FIGURE 3.12. TREAT Capsule Shipping Configuration

3.5 CHEMICAL REACTIONS

The possibility of deleterious chemical reactions between the materials present is remote. The materials to be used in the experiment do not react explosively, and special care will be exercised to assure the elimination of air and water vapor which might react with the sodium-potassium (NaK) alloy. Neither nickel nor any major constituents of stainless steel react to form alloys with NaK. There is no indication that the thermocouple insulator (MgO) or the Thermoflex insulation will be reduced by NaK.

The possibility of violent chemical reactions between the fuel, NaK and other capsule components is also quite remote. Chemical reactions between UO_2 and sodium were found to occur in a sealed container heated to 850 °C with an excess of oxygen (O/U ratio greater than 2).⁽²⁰⁾ A stable compound, Na_3UO_4 was found in concentrations of only 100 to 200 ppm indicating a very limited reaction.

UO_2 and PuO_2 can be reduced by alkali metals but the reaction is not significant unless the alkali metal oxide product is removed from the reaction. Since the capsule constitutes a closed system with no mechanism for removing sodium or potassium oxides, the reduction process is not considered probable.

Experimental evidence that no hazardous chemical reactions will occur was found in the postirradiation examination of the G.E. C2C experiment.⁽²⁾ Molten fuel was expelled from the fuel pin into the NaK annulus with no detectable chemical reactions. The aluminum thermal dam was melted and a portion of it found inside the fuel pin. Metallographic examination of the aluminum-containing section revealed a dendritic

structure in the aluminum typical of once-molten alloy materials. The associated autoradiography did not reveal any plutonium or uranium in the aluminum core.

The TREAT capsule will be argon-filled during assembly to a pressure of 1 atm (absolute) in order to minimize the possibility of a NaK reaction with air in the unlikely event of failure or leakage of the inner capsule.

3.6 RADIATION HAZARDS

The computer code RIBD⁽²¹⁾ was used to calculate fission and decay products, and ISOSHL⁽²²⁾ was used to calculate photon production and dose rates. Calculations were made on the basis of first pre-irradiating test pins to exposures of 10,000 and 50,000 MWd/MTM in EBR-II and then, after a substantial cooling time, subjecting them to a TREAT transient of 345 MW/sec.

Tables 3.1 and 3.2 present calculated dose rates versus time after the EBR-II irradiation with the fuel shielded by both the inner and TREAT capsules. Table 3.3 shows total curies as a function of cooling times. Because of the remote possibility of capsule rupture during shipment, fission gas activity is presented as a function of cooling time for both exposures in Tables 3.4 and 3.5. Fission gas activity drops off rapidly, and by the time the fuel pin is re-encapsulated, most of the radioactive fission gas isotopes have decayed away.

Tables 3.6 through 3.8 show the dose rate and curie content data resulting from a 345 MW/sec TREAT transient on green (nonpre-irradiated) fuel. The dose rate of a pre-irradiated transient test capsule is then determined by adding the appropriate dose rates for pre-irradiated and green fuel. The same procedure is used for the curie content. The highest

TABLE 3.1. 10,000 MWd/MTM EBR-II Exposure Roentgens/hr

Cooling Times (Days)	Distance from Centerline		
	(Treat Capsule Surface) 1.5 in.	6 in.	5 ft
0	1.715×10^6	1.731×10^5	3.677×10^3
30	5.114×10^4	4.920×10^3	1.102×10^2
75	2.465×10^4	2.348×10^3	5.297×10^1
180	8.906×10^3	8.53×10^2	1.908×10^1

TABLE 3.2. 50,000 MWd/MTM EBR-II Exposure Roentgens/hr

Cooling Times (Days)	Distance from Centerline		
	(Treat Capsule Surface) 1.5 in.	6 in.	5 ft
0	1.746×10^6	1.760×10^5	3.743×10^3
30	7.453×10^4	7.156×10^3	1.603×10^2
75	4.027×10^4	3.843×10^3	8.648×10^1
180	1.553×10^4	1.483×10^3	3.332×10^1

TABLE 3.3. Total Activity After Shutdown (Curies)

Cooling Times (Days)	10,000 MWd/MTM	50,000 MWd/MTM
0	2.726×10^4	2.823×10^4
30	1.311×10^3	2.088×10^3
75	6.818×10^2	1.266×10^3
180	2.648×10^2	6.23×10^2

**TABLE 3.4. 10,000 MWd/MTM EBR-II Exposure Fission
Gas Activity (Curies) After Removal
from EBR-II**

Isotope & Half-life		0	30 days	75 days	180 days
Br ⁸²	35.5 h	2.432×10^{-2}	1.767×10^{-8}	1.065×10^{-17}	0.0
Br ⁸³	2.3 h	3.752×10^1	0.0	0.0	0.0
Br ^{84m}	6 m	8.025×10^{-1}	0.0	0.0	0.0
Br ⁸⁴	32 m	$4.463 \times 10^{+1}$	0.0	0.0	0.0
Br ⁸⁵	3 m	5.983×10^1	0.0	0.0	0.0
Br ⁸⁶	54 s	8.348×10^1	0.0	0.0	0.0
Br ⁸⁷	56 s	1.039×10^2	0.0	0.0	0.0
Br ⁸⁸	16 s	1.167×10^2	0.0	0.0	0.0
Br ⁸⁹	4.5 s	1.044×10^2	0.0	0.0	0.0
Br ⁹⁰	1.6 s	8.112×10^1	0.0	0.0	0.0
Total		6.325×10^2	1.767×10^{-8}	1.065×10^{-17}	0.0
Kr ^{83m}	1.86 h	3.752×10^1	0.0	0.0	0.0
Kr ^{85m}	4.6 h	6.010×10^1	0.0	0.0	0.0
Kr ⁸⁵	10.1 y	2.066×10^{-1}	2.060×10^{-1}	2.043×10^{-1}	2.005×10^{-1}
Kr ⁸⁷	75 m	1.203×10^2	0.0	0.0	0.0
Kr ⁸⁸	2.79 h	1.590×10^2	0.0	0.0	0.0
Kr ⁸⁹	3.1 m	1.838×10^2	0.0	0.0	0.0
Kr ⁹⁰	34 s	1.950×10^2	0.0	0.0	0.0
Kr ⁹¹	9.9 s	1.676×10^2	0.0	0.0	0.0
Kr ⁹²	3 s	1.237×10^2	0.0	0.0	0.0
Kr ⁹³	2 s	7.713×10^1	0.0	0.0	0.0
Kr ⁹⁴	1.4 s	3.724×10^1	0.0	0.0	0.0
Kr ⁹⁵	short	1.704×10^1	0.0	0.0	0.0
Total		1.179×10^3	2.060×10^{-1}	2.043×10^{-1}	2.005×10^{-1}

TABLE 3.4. (contd)

Isotope & Half-life		0	30 days	75 days	180 days
I ¹²⁸	25 m	1.405×10^{-1}	0.0	0.0	0.0
I ¹²⁹	1.7×10^1 y	8.030×10^{-7}	8.118×10^{-7}	8.157×10^{-7}	8.181×10^{-7}
I ¹³⁰	12.5 h	6.374×10^{-1}	3.049×10^{-18}	0.0	0.0
I ¹³¹	8.1 d	1.985×10^2	$1.566 \times 10^{+1}$	3.244×10^{-1}	3.851×10^{-5}
I ¹³²	2.3 h	2.715×10^2	4.468×10^{-1}	3.019×10^{-5}	5.709×10^{-15}
I ¹³³	20.5 h	3.073×10^2	1.536×10^{-8}	4.979×10^{-24}	0.0
I ¹³⁴	52.5 m	3.225×10^2	0.0	0.0	0.0
I ¹³⁵	6.7 h	3.179×10^2	1.442×10^{-30}	0.0	0.0
I ¹³⁶	84 s	2.329×10^2	0.0	0.0	0.0
I ¹³⁷	23 s	1.597×10^2	0.0	0.0	0.0
I ¹³⁸	5.9 s	8.438×10^1	0.0	0.0	0.0
I ¹³⁹	2.3 s	3.543×10^1	0.0	0.0	0.0
Total		1.931×10^3	1.61×10^1	3.244×10^{-1}	3.932×10^{-5}
Xe ^{131m}	11.9 d	1.181	4.584×10^{-1}	4.271×10^{-2}	9.791×10^{-5}
Xe ^{133m}	2.26 d	7.767	1.481×10^{-3}	1.897×10^{-9}	3.455×10^{-23}
Xe ¹³³	5.3 d	3.078×10^2	7.482	2.074×10^{-2}	2.262×10^{-8}
Xe ^{135m}	15.6 m	1.011×10^2	4.049×10^{-31}	0.0	0.0
Xe ¹³⁵	9.16 h	3.482×10^2	3.328×10^{-21}	0.0	0.0
Xe ¹³⁷	3.8 m	3.183×10^2	0.0	0.0	0.0
Xe ¹³⁸	17 m	2.592×10^2	0.0	0.0	0.0
Xe ¹³⁹	42 s	1.793×10^2	0.0	0.0	0.0
Xe ¹⁴⁰	16 s	1.087×10^2	0.0	0.0	0.0
Xe ¹⁴¹	2.3 s	5.243×10^1	0.0	0.0	0.0
Xe ¹⁴²	1.5 s	2.254×10^1	0.0	0.0	0.0
Xe ¹⁴³	1.0 s	7.227	0.0	0.0	0.0
Xe ¹⁴⁴	1.0 s	4.791×10^{-1}	0.0	0.0	0.0
Total		1.714×10^3	7.942	6.345×10^{-2}	9.793×10^{-5}
Grand Total		5.457×10^3	2.425×10^1	5.922×10^{-1}	2.005×10^{-1}

**TABLE 3.5. 50,000 MWd/MTM EBR-II Exposure Fission
Gas Activity (Curies) After Removal
from EBR-II**

Isotope & Half-life		0	30 days	75 days	180 days
Br ⁸²	35.5 h	2.475×10^{-2}	1.798×10^{-8}	1.084×10^{-17}	0.0
Br ⁸³	2.3 h	3.752×10^1	0.0	0.0	0.0
Br ^{84m}	6 m	8.025×10^{-1}	0.0	0.0	0.0
Br ⁸⁴	32 m	$4.463 \times 10^{+1}$	0.0	0.0	0.0
Br ⁸⁵	3 m	5.983×10^1	0.0	0.0	0.0
Br ⁸⁶	54 s	$8.348 \times 10^{+1}$	0.0	0.0	0.0
Br ⁸⁷	56 s	1.039×10^2	0.0	0.0	0.0
Br ⁸⁸	16 s	1.167×10^2	0.0	0.0	0.0
Br ⁸⁹	4.5 s	1.044×10^2	0.0	0.0	0.0
Br ⁹⁰	1.6 s	8.112×10^1	0.0	0.0	0.0
Total		6.325×10^2	1.798×10^{-8}	1.084×10^{-17}	0.0
Kr ^{83m}	1.86 h	3.752×10^1	0.0	0.0	0.0
Kr ^{85m}	4.6 h	6.010×10^1	0.0	0.0	0.0
Kr ⁸⁵	10.1 y	9.989×10^{-1}	9.941×10^{-1}	9.861×10^{-1}	9.677×10^{-1}
Kr ⁸⁷	75 m	1.203×10^2	0.0	0.0	0.0
Kr ⁸⁸	2.79 h	1.590×10^2	0.0	0.0	0.0
Kr ⁸⁹	3.1 m	1.838×10^2	0.0	0.0	0.0
Kr ⁹⁰	34 s	1.950×10^2	0.0	0.0	0.0
Kr ⁹¹	9.9 s	1.676×10^2	0.0	0.0	0.0
Kr ⁹²	3 s	1.237×10^2	0.0	0.0	0.0
Kr ⁹³	2 s	7.713×10^1	0.0	0.0	0.0
Kr ⁹⁴	1.4 s	3.724×10^1	0.0	0.0	0.0
Kr ⁹⁵	short	1.704×10^1	0.0	0.0	0.0
Total		1.180×10^3	9.941×10^{-1}	9.861×10^{-1}	9.677×10^{-1}

TABLE 3.5. (contd)

<u>Isotope & Half-life</u>		<u>0</u>	<u>30 days</u>	<u>75 days</u>	<u>180 days</u>
I ¹²⁸	25 m	1.606×10^{-1}	0.0	0.0	0.0
I ¹²⁹	1.7×10^7 y	4.075×10^{-6}	4.084×10^{-6}	4.089×10^{-6}	4.092×10^{-6}
I ¹³⁰	12.5 h	6.453×10^{-1}	3.087×10^{-18}	0.0	0.0
I ¹³¹	8.1 d	1.986×10^2	1.568×10^1	3.245×10^{-1}	3.852×10^{-5}
I ¹³²	2.3 h	2.714×10^2	4.468×10^{-1}	3.020×10^{-5}	5.710×10^{-15}
I ¹³³	20.5 h	3.073×10^2	1.536×10^{-8}	4.979×10^{-24}	0.0
I ¹³⁴	52.5 m	3.225×10^2	0.0	0.0	0.0
I ¹³⁵	6.7 h	3.179×10^2	1.442×10^{-30}	0.0	0.0
I ¹³⁶	84 s	2.329×10^2	0.0	0.0	0.0
I ¹³⁷	23 s	1.597×10^2	0.0	0.0	0.0
I ¹³⁸	5.9 s	8.438×10^1	0.0	0.0	0.0
I ¹³⁹	2.3 s	3.543×10^1	0.0	0.0	0.0
Total		1.931×10^3	1.611×10^1	3.246×10^{-1}	4.261×10^{-5}
Xe ^{131m}	11.9 d	1.191	4.603×10^{-1}	4.284×10^{-2}	9.819×10^{-5}
Xe ^{133m}	2.26 d	7.781	1.483×10^{-3}	1.899×10^{-9}	3.459×10^{-23}
Xe ¹³³	5.3 d	3.079×10^2	7.483	2.075×10^{-2}	2.262×10^{-8}
Xe ^{135m}	15.6 m	1.011×10^2	4.049×10^{-31}	0.0	0.0
Xe ¹³⁵	9.16 h	3.482×10^2	3.328×10^{-21}	0.0	0.0
Xe ¹³⁷	3.8 m	3.183×10^2	0.0	0.0	0.0
Xe ¹³⁸	17 m	2.592×10^2	0.0	0.0	0.0
Xe ¹³⁹	42 s	1.793×10^2	0.0	0.0	0.0
Xe ¹⁴⁰	16 s	1.087×10^2	0.0	0.0	0.0
Xe ¹⁴¹	2.3 s	5.243×10^1	0.0	0.0	0.0
Xe ¹⁴²	1.5 s	2.254×10^1	0.0	0.0	0.0
Xe ¹⁴³	1.0 s	7.227	0.0	0.0	0.0
Xe ¹⁴⁴	1.0 s	4.791×10^{-1}	0.0	0.0	0.0
Total		1.714×10^3	7.945	6.359×10^{-2}	9.821×10^{-5}
Grand Total		5.458×10^3	2.505×10^1	1.376	9.677×10^{-1}

TABLE 3.6. TREAT Transient Exposure Green Fuel Roentgens/hr

Cooling times (days)	Distance from Centerline		
	(Treat Capsule Surface) 1.5 in.	6 in.	5 feet
0	7.947×10^4	1.073×10^4	1.531×10^2
1 hour	1.831×10^3	2.468×10^2	3.524×10^0
1 day	2.523×10^1	3.281×10^0	4.778×10^{-2}
10 days	1.510×10^0	2.323×10^{-1}	3.359×10^{-3}

TABLE 3.7. Total Activity After Shutdown (Curies)

Cooling Times	Activity
0	3.962×10^4
1 hour	2.365×10^1
1 day	7.271×10^{-1}
10 days	4.99×10^{-2}
30 days	1.496×10^{-2}

**TABLE 3.8. Fission Gas Activity (Curies) from TREAT
Transient Exposure (Green Fuel)**

Isotope and Half-Life		0	1 hour	1 day	10 days	30 days
Br ⁸²	35.5 h	1.064 x 10 ⁻¹¹	1.043 x 10 ⁻¹¹	6.643 x 10 ⁻¹²	9.567 x 10 ⁻¹⁴	7.723 x 10 ⁻¹²
Br ⁸³	2.3 h	3.581 x 10 ⁻³	5.244 x 10 ⁻²	7.617 x 10 ⁻⁵	6.157 x 10 ⁻³²	0.0
Br ^{84m}	6 m	6.705 x 10 ⁻²	6.552 x 10 ⁻⁵	0.0	0.0	0.0
Br ⁸⁴	32 m	5.993 x 10 ⁻¹	1.755 x 10 ⁻¹	1.825 x 10 ⁻¹⁴	0.0	0.0
Br ⁸⁵	3 m	2.750 x 10 ⁰	1.125 x 10 ⁻⁵	0.0	0.0	0.0
Br ⁸⁶	54 s	2.302 x 10 ¹	4.692 x 10 ⁻¹⁹	0.0	0.0	0.0
Br ⁸⁷	56 s	8.576 x 10 ⁻¹	1.648 x 10 ⁻¹⁸	0.0	0.0	0.0
Br ⁸⁸	16 s	2.178 x 10 ²	0.0	0.0	0.0	0.0
Br ⁸⁹	4.5 s	5.103 x 10 ¹	0.0	0.0	0.0	0.0
Br ⁹⁰	1.6 s	1.716 x 10 ³	0.0	0.0	0.0	0.0
Total		2.012 x 10 ³	2.280 x 10 ⁻¹	7.617 x 10 ⁻⁵	9.563 x 10 ⁻¹⁴	7.723 x 10 ⁻¹⁸
Kr ^{83m}	1.86 h	1.229 x 10 ⁻⁷	1.602 x 10 ⁻²	2.920 x 10 ⁻⁴	2.736 x 10 ⁻³¹	0.0
Kr ^{85m}	4.6 h	4.064 x 10 ⁻⁵	9.695 x 10 ⁻²	2.586 x 10 ⁻³	4.289 x 10 ⁻¹⁸	0.0
Kr ⁸⁵	10.1 y	1.202 x 10 ⁻⁷	2.537 x 10 ⁻⁷	1.090 x 10 ⁻⁶	1.112 x 10 ⁻⁶	1.108 x 10 ⁻⁶
Kr ⁸⁷	75 m	8.962 x 10 ⁻⁶	4.196 x 10 ⁻¹	1.436 x 10 ⁻⁶	0.0	0.0
Kr ⁸⁸	2.79 h	1.024 x 10 ⁻¹	3.503 x 10 ⁻¹	1.181 x 10 ⁻³	7.194 x 10 ⁻²⁷	0.0
Kr ⁸⁹	3.1 m	2.793 x 10 ¹	6.604 x 10 ⁻⁵	0.0	0.0	0.0
Kr ⁹⁰	34 s	1.081 x 10 ²	2.806 x 10 ⁻³¹	0.0	0.0	0.0
Kr ⁹¹	9.9 s	4.260 x 10 ²	0.0	0.0	0.0	0.0
Kr ⁹²	3 s	1.071 x 10 ³	0.0	0.0	0.0	0.0
Kr ⁹³	2 s	7.500 x 10 ²	0.0	0.0	0.0	0.0
Kr ⁹⁴	1.4 s	5.240 x 10 ²	0.0	0.0	0.0	0.0
Kr ⁹⁵	short	3.591 x 10 ³	0.0	0.0	0.0	0.0
Kr ⁹⁷	1.0 s	2.703 x 10 ³	0.0	0.0	0.0	0.0
Total		9.212 x 10 ³	8.829 x 10 ⁻¹	4.062 x 10 ⁻³	1.112 x 10 ⁻⁶	1.108 x 10 ⁻⁶

BNWL

TABLE 3.8. (contd)

Isotope and Half-Life		0	1 hour	1 day	10 days	30 days
I ¹³¹	8.1 d	3.112×10^{-4}	3.286×10^{-3}	5.709×10^{-3}	2.934×10^{-3}	5.250×10^{-4}
I ¹³²	2.3 h	1.193×10^{-4}	5.992×10^{-3}	1.968×10^{-2}	2.889×10^{-3}	4.058×10^{-5}
I ¹³³	20.5 h	8.121×10^{-3}	8.056×10^{-2}	5.490×10^{-2}	4.398×10^{-5}	5.790×10^{-12}
I ¹³⁴	52.5 m	3.073×10^{-1}	1.413×10^0	1.018×10^{-7}	0.0	0.0
I ¹³⁵	6.7 h	1.690×10^{-1}	3.416×10^{-1}	3.164×10^{-2}	6.261×10^{-12}	1.710×10^{-33}
I ¹³⁶	84 s	4.934×10^1	4.325×10^{-12}	0.0	0.0	0.0
I ¹³⁷	23 s	2.938×10^2	0.0	0.0	0.0	0.0
I ¹³⁸	5.9 s	8.129×10^2	0.0	0.0	0.0	0.0
I ¹³⁹	2.3 s	1.139×10^3	0.0	0.0	0.0	0.0
Total		2.295×10^3	1.844×10^0	1.119×10^{-1}	5.367×10^{-3}	5.656×10^{-4}
Xe ^{131m}	11.9 d	3.406×10^{-6}	3.426×10^{-6}	5.380×10^{-6}	1.507×10^{-5}	1.046×10^{-5}
Xe ^{133m}	2.26 d	1.951×10^{-10}	1.315×10^{-5}	4.066×10^{-4}	7.121×10^{-5}	1.731×10^{-7}
Xe ¹³³	5.3 d	4.928×10^{-4}	7.721×10^{-4}	9.970×10^{-3}	6.482×10^{-3}	4.715×10^{-4}
Xe ^{135m}	15.6 m	9.689×10^{-3}	8.872×10^{-2}	8.897×10^{-3}	1.760×10^{-12}	4.810×10^{-34}
Xe ¹³⁵	9.16 h	3.961×10^{-7}	2.376×10^{-2}	8.209×10^{-2}	1.433×10^{-8}	2.817×10^{-24}
Xe ¹³⁷	3.8 m	8.651×10^0	9.860×10^{-4}	0.0	0.0	0.0
Xe ¹³⁸	17 m	3.190×10^0	6.860×10^{-1}	2.446×10^{-25}	0.0	0.0
Xe ¹³⁹	42 s	1.071×10^2	6.095×10^{-25}	0.0	0.0	0.0
Xe ¹⁴⁰	16 s	3.191×10^2	0.0	0.0	0.0	0.0
Xe ¹⁴¹	2.3 s	1.046×10^3	0.0	0.0	0.0	0.0
Xe ¹⁴²	1.5 s	2.802×10^2	0.0	0.0	0.0	0.0
Xe ¹⁴³	1.0 s	3.434×10^3	0.0	0.0	0.0	0.0
Xe ¹⁴⁴	1.0 s	2.964×10^3	0.0	0.0	0.0	0.0
Total		8.162×10^3	8.002×10^{-1}	1.014×10^{-1}	6.569×10^{-3}	4.821×10^{-4}
Grand total		2.168×10^4	3.755×10^0	2.174×10^{-1}	1.244×10^{-2}	1.049×10^{-3}

levels that could be encountered during this series of tests would be for a PNL-2 (50,000 MWd/MTM) fuel pin subjected to a TREAT transient 75 days after removal from EBR-II. PNL-2 was removed from EBR-II on August 31, 1969 and PNL-1 on June 14, 1968. The curie content immediately after the transient would be 6.2×10^4 curies (including fission gas) and the dose rate at the surface of the capsule would be 1.2×10^5 R/hr.

Specific calculations were not made of stainless steel activation in EBR-II, but activity per cm^3 of 304 SS steel was calculated for the FTR.⁽²³⁾ Using these, total activation is found to be only 66 curies for the amount of stainless steel in the EBR-II test pin cladding, assuming saturated concentration of all nuclides at 0.025 MW (see Table 3.9). This is negligible when compared to the fission product activity even at 180 days cooling time. The ^{60}Co , however, may contribute a significant part of the activity at much longer times because of its long half-life. The spectrum of EBR-II and FTR do not differ enough to change this result appreciably. Also, the FTR spectrum intensity on which the results are based is about double that of EBR-II, thus lending even more conservatism to this result.

Appendix E contains several tables showing photon production rates and fission gas concentrations for the 10,000 MWd/MTM, 50,000 MWd/MTM, and the TREAT transient cases.

3.7 DISPOSAL OF RADIOACTIVE MATERIALS

All radioactive materials generated during these tests will be contained within the TREAT capsule which will be returned to BNW in the T-2 shipping cask.

TABLE 3.9. 304 SS Activation (Based on 304 SS Concentrations)

<u>Nuclide</u>	<u>T 1/2</u>	<u>Dis/cm³-sec</u>
Co ⁶⁰	5.2 y	7.2×10^{10}
Cr ⁵¹	27.8 d	2.8×10^{11}
Fe ⁵⁹	45 d	6×10^{10}
⁵⁸ Ni(n,p) ⁵⁸ Co	71 d	3.3×10^{11}
⁵⁴ Fe(n,p) ⁵⁴ Mn	300 d	1.6×10^{11}
⁵⁶ Fe(n,p) ⁵⁶ Mn	2.6 h	3.6×10^{10}
⁵² Cr(n,p) ⁵² V	3.8 m	1.6×10^{10}
²⁸ Si(n,p) ²⁸ Al	2.3 m	6.5×10^9
⁵⁴ Fe(n,a) ⁵¹ Cr	287 d	4.9×10^9
⁶⁰ Ni(n,p) ⁶⁰ Co	5.2 y	1.2×10^9
TOTAL		96.66×10^{10} dis/cm ³ -sec

for 304 SS in EBR-II $\longrightarrow$ 66.356 Curies.

4.0 CONCLUSIONS

This document discusses the possible hazards associated with the expected transient and maximum accident transient for a maximum exposure fuel pin (44,250 MWd/MTM). It is concluded that even with the most severe response credible, the experiment will be contained within the TREAT capsule and will impose no undue hazards to the testing facilities or personnel. This document will be referenced as the hazards analysis for the complete EBR-II/TREAT transient test series which includes fuel pins and transients of equal or less irradiation exposure.

APPENDIX A

ENGINEERING DRAWINGS OF EXPERIMENT APPARATUS

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

REV		PT. NO.	DESCRIPTION
1	1	1	ASSEMBLY
2	1	2	
3	1	3	
4	1	4	
5	1	5	
6	1	6	
7	1	7	PEDESTAL
8	1	8	TUBE .250 DIA. x .016 WALL x 3/32 LG 304 SST
9	1	9	SPACER TUBE .188 OD x .148 ID x 10 7/32 LG 304 SST
10	1	10	TUBE .040 OD x .034 ID x 32 27/32 LG 304 SST
11	1	11	EXTENSOMETER 4-B-15274
12	2	12	INSULATOR PELLET .214 DIA. x 1/2 LG DEPLETED UO ₂
13	53	13	FUEL, SOLID PELLET, 25% PUO ₂ , 75% UO ₂ , 1/4 LG (FULLY ENRICHED)
14	53	14	FUEL, ANNULAR PELLET, 25% PUO ₂ , 75% UO ₂ , 1/4 LG FULLY ENRICHED
15	1	15	
16	1	16	FUEL MOTION RESTRICTOR
17	1	17	END CAP
18	1	18	ROD .712 ± .001 DIA. x 2 7/32 LG NICKEL
19	1	19	FLUX MONITOR WIRE .020 DIA x 28 3/8 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 $63 \sqrt{R}$ (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

WIRING SCHEDULE		
T.C. NO.	POLARITY	CONDUCTOR P-W. NO.
1	+	A
	-	B
2	-	D
3	+	E
	-	F
4	+	G
	-	H
5	-	I
	+	J
6	+	K
	-	L
7	+	N
	-	O
8	+	P
	-	S
GROUND		T
N/C		U
N/C		V
N/C		W

LEGEND

INDICATE THERMOCOUPLE IDENTIFICATION

1 ASSEMBLY

[illegible]

650-7341 ACCTEC

- GENERAL INSTRUCTIONS
(UNLESS OTHERWISE SPECIFIED)
1. TOLERANCES: FRACTIONS - $\pm$ DECIMALS - .030
INCHES - .010 ANGLES - $\pm$.030°
 2. ALL MACHINED SURFACES $\pm$.0015 (ASME B46 1982)
 3. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS
 4. ALL MATL. TO BE AS SPEC'D OR APPROVED EQUAL QUALITY
 5. FINISH, COMPRESSION - FREE LENGTH - 6.000
AIR/D.O., NUMBER OF COILS - 19, WIRE DIA - .177
RATE - 70%/IN. SOLID HEIGHT - 3.673. MATL. SPRING STEEL WIRE 10
 6. PT #25 PLUG, ELIMINATE LOCKING COLLAR
AND NICKON BOLT
 7. PT #27 PLUG, ELIMINATE LOCKING COLLAR
AND ENVIRONMENT RESISTANT RING GRAMMET.

LEGEND

(--) INDICATE THERMOCOUPLE NUMBER

△	2	32	SCREW	54 NO 3
△	AR	33	GRID WIRE PVC INSULATED	COPPER
	1	34	PISTON	54 NO 3
	2	35	SCR, 10-24 UNC-2, 1/2 LG SOC SET	54 NO 3
	3	36	SCR, 1/2-20 UNC-2, 1/2 LG SOC NO CAP	54 NO 3
	4	37	SCR, 1/2-20 UNC-2, 1/2 LG NO CAP	54 NO 3
	5	38	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	6	39	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	7	40	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	8	41	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	9	42	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	10	43	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	11	44	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	12	45	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	13	46	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	14	47	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	15	48	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	16	49	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	17	50	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	18	51	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	19	52	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	20	53	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	21	54	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	22	55	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	23	56	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	24	57	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	25	58	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	26	59	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	27	60	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	28	61	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	29	62	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	30	63	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	31	64	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	32	65	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	33	66	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	34	67	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	35	68	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	36	69	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	37	70	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	38	71	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	39	72	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	40	73	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	41	74	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	42	75	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	43	76	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	44	77	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	45	78	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	46	79	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	47	80	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	48	81	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	49	82	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	50	83	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	51	84	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	52	85	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	53	86	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	54	87	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	55	88	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	56	89	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	57	90	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	58	91	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	59	92	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	60	93	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	61	94	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	62	95	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	63	96	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	64	97	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	65	98	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	66	99	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST
	67	100	LOCKWASHER, 1/2 SPLIT (AVER STD)	304 SST

4	70,71,72 671 10W F.S 2N 2-29 CHANGED PARTS 67 D-11 3) ADDED PT 70,72 (7) ADDED PARTS 49 (7) 2N	27 6/2/70				
5	DESCRIPTION	REV	APP'D			
	REVISIONS	CONT'D	DATE	BY	FOR	DATE

[illegible]

APPENDIX B

FISSION GAS PRESSURE BUILDUP
DURING STEADY-STATE IRRADIATION AND
PRESSURE CAPABILITY OF INNER CAPSULE

APPENDIX BFISSION GAS PRESSURE IN PNL 2-8 BEFORE TREAT
IRRADIATION

76.57 g fuel in PNL 2-8

$$\begin{aligned} \text{g of metal} &= 76.57 \left(\frac{239}{271} \right) = 67.5 \text{ g} \\ &= 6.75 \times 10^{-5} \text{ MIM} \end{aligned}$$

Peak

$$\text{Burnup} = 4.425 \times 10^4 \text{ MWd/MTM} \quad \text{Axial Peak/Ave} = 1.125$$

$$\frac{4.425 \times 10^4 \text{ MWd}}{1.125} \cdot \frac{6.75 \times 10^{-5} \text{ MIM}}{\text{MIM}} = 2.65 \text{ MWd}$$

$$\begin{aligned} 2.65 \text{ MWd} & \left(3.1 \times 10^{10} \frac{\text{fissions}}{\text{Watt-sec}} \right) \left(\frac{10^6 \text{ Watt}}{\text{MW}} \right) \left(\frac{24 \text{ hr}}{\text{day}} \right) \left(\frac{3600 \text{ sec}}{\text{hr}} \right) \\ &= 7.1 \times 10^{21} \text{ fissions} \end{aligned}$$

Assuming 27% of fissions form gas in fast flux

$$7.1 \times 10^{21} \text{ fissions} \left(\frac{0.27 \text{ gas atoms}}{\text{fission}} \right) = 1.92 \times 10^{21} \text{ atoms}$$

$$\begin{aligned} \text{Volume} &= \frac{(1.92 \times 10^{21} \text{ atoms})(2.24 \times 10^4 \text{ cm}^3/\text{mole})}{6.02 \times 10^{23} \text{ atoms/mole}} \\ (\text{STP}) & \end{aligned}$$

$$= 71.3 \text{ cm}^3 \text{ fission gas generated (STP)}$$

Only 80% of the gas is actually released therefore

$$\begin{aligned} v_{\text{STP}} &= 0.80(71.3) \text{ cm}^3 \\ &= 57.0 \text{ cm}^3 \end{aligned}$$

Volume of He at STP = 7.1 cm³

$$\text{Combined volume} = 57.0 + 7.1 = 64.1 \text{ cm}^3 \text{ (STP)}$$

Assume pre-transient temperature of 300 °F

$$\begin{aligned} P_2 &= \frac{P_1 v_1 T_2}{T_1 v_2} = \frac{(1 \text{ atm})(64.1 \text{ cm}^3)(460 + 300)^\circ\text{R}}{(460 + 32)^\circ\text{R} (7.1 \text{ cm}^3)} \\ &= \frac{(71.42)(760)}{(492)(7.1)} \text{ atm} \\ &= 13.9 \text{ atm} \\ &= 204 \text{ psi} \leftarrow \text{Initial fuel pin pressure} \end{aligned}$$

PRESSURE CAPABILITY OF INNER CAPSULE

Estimates of the internal pressure capability of the inner capsule are based on hoop stress equation.

$$\text{Yield pressure } P_y = \frac{2 \sigma_y t}{d}$$

$$\text{Burst pressure } P_b = \frac{2 \sigma_u t}{d}$$

where σ_y = yield strength

σ_u = ultimate strength

$t = 0.049$ in. $d = 1.125$ in.

Temperature of the wall (inner surface, wax) at (ARGUS)

5 sec 600 °F

10 sec 1000 °F

at 600 °F $\sigma_y = 23,000$ psi: (17,000-27,000)

$\sigma_u = 64,000$ psi: (60,000-67,000)

at 1000 °F $\sigma_y = 23,000$ psi: (17,000-28,000) (BNW 891)

$\sigma_u = 58,000$ psi: (54,000-62,000)

600 °F $P_y = 2004$ psi

$P_b = 5575$ psi

1000 °F $P_y = 2004$ psi

$P_b = 5052$ psi

APPENDIX C

CALCULATIONS FOR EXPECTED TRANSIENT

APPENDIX CINTERNAL FUEL PIN PRESSURE DUE TO FUEL MELTING AND RETAINED FISSION GAS RELEASE DURING EXPECTED TRANSIENTBASIS:

1. Negligible fission gas is produced during transient irradiation.
2. Fission gas retention during steady state is assumed as follows: columnar grain region 0.080 in. diam has 5% gas retention, remainder has 90% of retention.
3. The fission gas released during steady state is allowed to equilibrate with the plenum at all times.
4. 100% fission gas retained in the fuel during steady state is released when the fuel melts and occupies the void region as determined by the smeared density. The transient released gas does not equilibrate with the plenum.
5. The fuel is assumed melted when it reaches 4900 °F.
6. As the fuel melts, it expands and reduces the void available for transient released fission gases.

Time After Transient	Nodes Melted (see Figure 3.4)	Volume of Fuel Melted (in. 3)	Cumulative Volume of Fuel Melted (in. 3)	Volume of gas Released (cm ³)	Size of Void Available (cm ³)*
3.1	None	None	None	None	1.93
3.2	17-19	0.0575	0.0575	7.32	1.84
3.3	15-19	0.0512	0.1086	6.51	1.76
3.4	13-19	0.0449	0.1536	5.72	1.69
3.5	11-19	0.0391	0.1926	4.98	1.63
3.6	8-19	0.0323	0.2249	4.10	1.58
3.7	6-19	0.0369	0.2618	0.52*	1.52
3.8	3-19	0.0285	0.2903	0.40*	1.47

Sample calculations for this table follow

* 10% released during melt

** Assumes the melted fuel volume increases 10%. The other fuel expansion up to 4900 °F is compensated for by cladding expansion.

The original total volume of fuel

$$V = \pi r^2 L$$

$$= \pi (0.109)^2 (13.5) = 0.504 \text{ in.}^3$$

Volume of fission gas generated in melted fuel at 3.2 sec

$$V_f = \left[\frac{0.0575 \text{ in}^3}{0.504 \text{ in}^3} \right] [71.3 \text{ cm}^3] = 8.13 \text{ cm}^3$$

90% is released

$$V_{fr} = (8.13 \text{ cm}^3) (0.90) = 7.32 \text{ cm}^3$$

Size of void available at 3.1 smeared density = 76.57%

$$V = 0.504 \text{ in.}^3 \times 0.234 = 0.118 \text{ in}^3$$

$$= (0.118 \text{ in}^3) \left(16.4 \frac{\text{cm}^3}{\text{in}^3} \right) = 1.93 \text{ cm}^3$$

Size of void available at 3.2

$$V_{\text{melt}} = 0.0575 \text{ in.}^3 = 0.943 \text{ cm}^3$$

$$\text{Volume increase} = (0.943) (0.1) = 0.094 \text{ cm}^3$$

$$\text{Void decrease} = 1.93 - 0.094 = 1.84 \text{ cm}^3$$

Sample calculation of pressure due to temperature increase of steady-state released fission gas at 3.1 sec

Model

Initial temperature (pre-transient) of both regions 300 °F

Total initial fission gas volume at STP (both regions)

64.1 cm³ (Appendix B)

Constant values

$$\begin{aligned}v &= 5.17 \text{ cm}^3 \quad R = 670.5 \text{ psia cm}^3/\text{g-mole } ^\circ\text{R} \\C_p &= 0.00274 \text{ Btu/g } ^\circ\text{F} \quad MW_{\text{ave}} = 25.34 \text{ g/g-mole} \\&= 0.0694 \text{ Btu/g-mole } ^\circ\text{F}\end{aligned}$$

at Time 0 pressure $P = 204 \text{ psi}$ (Appendix B)

at Time 3.1:

Grams of gas in each region at $300 ^\circ\text{F}$ before heat up

$$\begin{aligned}(64.1 \text{ cm}^3) \frac{25.23 \text{ g}}{\text{g-mole}} \frac{\text{g-mole}}{22,400 \text{ cm}^3} &= 0.072 \text{ g} \\37\% \text{ in void } 63\% \text{ in plenum}\end{aligned}$$

$$n_o = 0.045 \text{ g} \quad n_o = 0.027 \text{ g}$$

Phase 1: $T' = 4400 ^\circ\text{F}$ (bulk fuel temp, ARGUS)

$$\begin{aligned}q &= (M_o T_o + M_o' T') c_p \\&= [(0.045 \text{ g})(300 ^\circ\text{F}) + (0.027 \text{ g})(4400 ^\circ\text{F})] 0.00274 \frac{\text{B}}{\text{g}} ^\circ\text{F} \\&= 0.363 \text{ Btu}\end{aligned}$$

$$\begin{aligned}\text{Phase 2: } nT &= \frac{n' T' v}{v'} \\&= n' \frac{(4400 ^\circ\text{F})(5.17 \text{ cm}^3)}{1.93 \text{ cm}^3} = 11,786 \text{ n } ^\circ\text{F}\end{aligned}$$

$$q = 0.363 \text{ Btu} = [11,786 ^\circ\text{F n}' + 4400 ^\circ\text{F n}'] \frac{0.0694 \text{ B}}{\text{g-mole } ^\circ\text{F}}$$

$$n' = \frac{0.363 \text{ Btu g-mole } ^\circ\text{F}}{0.0694 \text{ Btu } 16,186 ^\circ\text{F}} = 0.000323 \text{ g-mole}$$

$$\begin{aligned}p &= \frac{n' T' R}{v'} = \frac{(0.000324 \text{ g-mole})(4860 ^\circ\text{R})(670.5) \text{ psi cm}^3}{1.93 \text{ cm}^3 \text{ g-mole } ^\circ\text{R}} \\&= 547.4 \text{ psi}\end{aligned}$$

at Time 3.2: 11.3% fuel melts-void size reduces

$$T' = 4500 ^\circ\text{F}$$

Phase 1:

$$\begin{aligned}q &= [(0.00285) 1340 + 0.000324(4500)] 0.0694 \\&= 0.366 \text{ Btu}\end{aligned}$$

Phase 2:

$$nT = \frac{n'T'v}{v'} = \frac{[4500][5.17 \text{ cm}^3]}{1.84 \text{ cm}^3}$$

$$= 12,644 \text{ n'}$$

$$q = 0.366 = 0.0694 (12,644 + 4500) \text{ n'}$$

$$n' = 0.000307$$

$$p = \frac{n'T'R}{v'} = \frac{(0.000307)(4960)(670.5)}{1.84}$$

$$= 555 \text{ psi}$$

To this must be added the pressure due to entrapped fission gas release during melt

$$P = \frac{P_{\text{STP}} v_{\text{STP}}^T}{v_{\text{STP}}^T} = \frac{(14.7 \text{ psi})(7.32 \text{ cm}^3)(4900 + 460)^\circ\text{R}}{(1.84 \text{ cm}^3)(460 + 32)^\circ\text{R}}$$

$$= 637 \text{ psi}$$

Total pressure at 3.2 sec $637 + 555 = 1192 \text{ psi}$

Table of Pressure Calculated as Above

<u>Time</u>	<u>Pressure Due to Entrapped Fission Gas Release</u>	<u>Pressure Due to Pre- transient Released Gas Heatup</u>	<u>Total as Shown in Figure 3.7</u>
3.1	0	547	547
3.2	637	555	1192
3.3	1229	561	1790
3.4	1771	567	2338
3.5	2260	574	2834
3.6	2676	580	3256
3.7	2731	587	3318
3.8	2775	593	3368

POWER TO BE GENERATED IN FUEL AFTER CLAD
FAILURE DURING EXPECTED TRANSIENT

Time of failure 3.50 sec after initiation of transient.

77% of transient has passed leaving 23% or 37 MW-sec.

Power factor = 1.004×10^{-4} watts fuel power/cm³ fuel
per watt of reactor power

Volume of fuel at failure = 8.23 cm⁵

$$Q = \frac{1.004 \text{ watts fuel power } 8.23 \text{ cm}^3 \text{ fuel } 37 \text{ W-sec} \times 10^6 \times 10^{-4}}{\text{cm}^3 \text{ fuel watts reactor power}}$$

$$Q = (1.004)(8.23)(37) \times 10^2 \text{ W-sec}$$

$$Q = 3.06 \times 10^4 \text{ W-sec} \times 948 \times 10^{-4} \text{ Btu/W-sec}$$

$$= 29 \text{ Btu generated in fuel after failure}$$

Heat released from cooling the fuel to 4900 °F (freezing point)

$$Q = MC_p \Delta T \quad C_p = 0.15 \text{ B/lb}_m \text{ } ^\circ\text{F}$$

$$Q = 76.6 \text{ g } (0.15 \frac{\text{B}}{\text{lb}_m} ^\circ\text{F}) (\frac{5100-4900}{^\circ\text{F}}) (\frac{1\text{lb}_m}{454}) \text{ g}$$

$$= 5.05 \text{ Btu}$$

Total heat available $29 + 5 = 34 \text{ Btu}$

PRESSURE WITHIN INNER CAPSULE FROM VAPORIZING NaK

Weight of NaK vaporized by 34 Btu

Heat of vaporization

$$\text{Na} = 1811 \text{ Btu/lb}_m$$

$$\text{K} = 872 \text{ Btu/lb}_m$$

NaK composition 78% K 22% Na

$$\text{Weight NaK} = \frac{(0.7 \text{ g})(34 \text{ Btu})}{872 \text{ Btu/lb}_m} - \frac{(0.22)(34 \text{ Btu})}{1811 \text{ Btu/lb}_m}$$

$$= 0.0345 \text{ lb}_m$$

Volume of NaK vapor at STP

$$V_{\text{NaK}} = \frac{(0.0345 \text{ lb}_m)(454 \text{ g/lb}_m)}{1.043 \times 10^{-3} \text{ g/cm}^3} = 1.5 \times 10^4 \text{ cm}^3$$

Available void for vapor expansion 267 cm³

Pressure after expansion with final temperature 1450 °F
(vaporization temperature)

$$\begin{aligned} P_{\text{NaK}} &= (14.7 \text{ psi}) \left(\frac{1.5 \times 10^4 \text{ cm}^3}{267 \text{ cm}^3} \right) \left(\frac{1910 \text{ }^\circ\text{R}}{492 \text{ }^\circ\text{R}} \right) \\ &= 3,210 \text{ psi} \end{aligned}$$

APPENDIX D
CALCULATIONS FOR MAXIMUM
ACCIDENT TRANSIENT

APPENDIX DINTERNAL FUEL PIN PRESSURE DUE TO FUEL MELTING
AND FISSION GAS RELEASE DURING MAXIMUM
ACCIDENT TRANSIENT

The basis for these calculations are the same as those in Appendix C for the expected transient, and the calculations are the same.

Time After Transient	Nodes Melted (see Figure 3.4)	Volume of Fuel Melted (in. 3)	Cumulative Volume of Fuel Melted (in. 3)	Volume of Fission Gas Released (cm ³)	Size of Void Available (cm ³)**
2.10	None	None	None	None	1.93
2.15	19-21	0.065	0.065	8.28	1.82
2.20	14-24	0.239	0.304	30.43	1.43
2.25	10-24	0.078	0.382	9.93	1.30
2.30	5-25	0.104	0.486	1.47*	1.13
2.35	3-35(Au)	0.018	0.504	0.25*	1.10

* 10% released during melt

** Assumes the melted fuel volume increases 10%. The other fuel expansion up to 4900 °F is compensated for by cladding expansion.

Table of Pressures Calculated as in Appendix C

Time	Pressure Due to Entrapped Fission Gas Released	Pressure Due to Pretransient Released Gas Heatup	Total as Shown in Figure 3.7
2.10	0 psi	505 psi	505 psi
2.15	729	625	1354
2.20	4136	742	4878
2.25	5359	865	6224
2.30	5567	984	6551
2.35	5603	1106	6709

POWER TO BE GENERATED IN FUEL AFTER CLAD
FAILURE DURING MAXIMUM ACCIDENT TRANSIENT

Time of failure = 2.15 sec

Power to fuel = 270 MW-sec

Power factor $\frac{1.004 \times 10^{-4} \text{ watts fuel power}}{\text{cm}^3 \text{ fuel watts reactor power}}$

Volume of fuel at failure = 8.23 cm³

$$\begin{aligned} Q &= \frac{1.004 \times 10^{-4} \text{ watts fuel power } 8.23 \text{ cm}^3 \text{ fuel } 270 \times 10^6 \text{ W-sec}}{\text{cm}^3 \text{ fuel watts reactor power}} \\ &= 2.235 \times 10^5 \text{ W-sec} \times 9.48 \times 10^{-4} \frac{\text{Btu}}{\text{W-sec}} \\ &= 2.2 \text{ Btu} \end{aligned}$$

Heat released from cooling the fuel to 4900 °F maximum
bulk fuel temperature 7600 °F (freezing point) (ARGUS)

$$\begin{aligned} Q &= MC_p \Delta T \\ &= 76.6 \text{ g} \left(0.15 \frac{\text{B}}{\text{lb}_m} \right)_{\text{°F}} \left(7600 - 4900 \text{ °F} \right) \left(\frac{1 \text{ lb}_m}{454 \text{ g}} \right) \\ &= 68.5 \text{ Btu} \end{aligned}$$

Total heat generation 212 + 68.5 = 280.5 Btu

PRESSURE WITHIN INNER CAPSULE FROM VAPORIZING
NaK AFTER FAILURE CAUSED BY MAXIMUM ACCIDENT
TRANSIENT

Weight of NaK to be vaporized (all NaK)

$$W_{\text{NaK}} = 0.2175 \text{ lb}_m$$

Volume of NaK vapor

$$V_{\text{NaK}} = \frac{(0.2175 \text{ lb}_m)(454 \text{ g/lb}_m)}{1043 \times 10^{-3} \text{ g/cm}^3} = 9.45 \times 10^4 \text{ cm}^3$$

Void = 267 cm³

Pressure after expansion

$$P_{\text{NaK}} = (14.7 \text{ psi}) \left(\frac{9.45 \times 10^4 \text{ cm}^3}{267 \text{ cm}^3} \right) \left(\frac{1910^\circ}{492^\circ} \right)$$

$$= 20,200 \text{ psi vaporizing all NaK}$$

Resultant pressure when all NaK in annulus between the fuel pin and the heat sink is vaporized and the remaining fuel freezes on the heat sink and raises the heat sink temperature.

Weight of NaK in annulus = 0.0143 lb_m

Volume NaK vapor

$$V_{\text{NaK}} = \frac{(0.0143 \text{ lb}_m)(454 \text{ g/lb}_m)}{1.043 \times 10^{-3} \text{ g/cm}^3} = 6.2 \times 10^3 \text{ cm}^3$$

$$P_{\text{NaK}} = (14.7 \text{ psi}) \left(\frac{6.2 \times 10^3 \text{ cm}^3}{267 \text{ cm}^3} \right) \left(\frac{1910^\circ}{492^\circ} \right)$$

$$= 1,325 \text{ psi}$$

FINAL BULK HEAT SINK TEMPERATURE IF THE ENERGY REMAINING AFTER VAPORIZATION OF THE NaK IN THE ANNULUS AS ABSORBED BY THE HEAT SINK.

Energy required to vaporize 0.0143 lb NaK

$$W_{\text{NaK}} = 0.0143 \text{ lb} = X \text{ Btu} \left[\frac{0.75}{894 \text{ B/lb}} + \frac{0.25}{1865 \text{ B/lb}} \right]$$

$$= 0.0143 \text{ lb} = X \text{ B} (9.74 \times 10^{-4} \text{ B/lb})$$

$$X = 14.7 \text{ Btu to vaporize the NaK}$$

280.5 Btu total energy available less 14.7 Btu absorbed in the Ni heat sink.

265.8 BtuVolume of Ni heat sink = 8.25 in.³

Initial temperature heat sink = 330 °F (ARGUS)

 $C_{pNi} = 0.103 \text{ B/lb}_m \text{ } ^\circ\text{F}$ $e = 0.321 \text{ lb/in}^3$

$$Q = eVC_p \Delta T = eVC_p (T - T_o)$$

$$T = \frac{Q}{eVC_p} + T_o = \frac{265.8 \text{ B in.}^3 \text{ lb}_m \text{ } ^\circ\text{F}}{0.321 \text{ lb}_m \text{ 8.25 in.}^3 \text{ 103 B}} + 330 \text{ } ^\circ\text{F}$$

$$= (974 + 330) \text{ } ^\circ\text{F}$$

= 1,304 °F final Ni temperature

melting temperature of Ni = 2250 °F

APPENDIX E
TABLES OF PHOTON PRODUCTION RATES
AND FISSION GAS CONCENTRATIONS

TABLE I. 10,000 MWd/MTM EBR-II Exposure

<u>Group</u>	<u>Group Average Energy MeV</u>	<u>Group Production Rate; Photons</u>			
		<u>Shutdown</u>	<u>30 days</u>	<u>75 days</u>	<u>180 days</u>
1	.15	7.173×10^{13}	3.277×10^{12}	1.246×10^{12}	2.726×10^{11}
2	.25	6.106×10^{13}	1.918×10^{11}	7.561×10^{10}	2.370×10^{10}
3	.35	5.40×10^{13}	1.196×10^{12}	7.113×10^{10}	2.202×10^8
4	.475	1.003×10^{14}	5.719×10^{12}	1.844×10^{12}	3.470×10^{11}
5	.65	1.247×10^{14}	3.794×10^{12}	2.318×10^{12}	8.797×10^{11}
6	.825	9.07×10^{13}	8.711×10^{12}	6.380×10^{12}	2.592×10^{12}
7	1.00	4.868×10^{13}	3.980×10^{11}	5.615×10^{10}	1.058×10^{10}
8	1.225	5.316×10^{13}	5.314×10^{10}	1.430×10^{10}	4.385×10^9
9	1.475	7.283×10^{13}	2.685×10^{12}	2.409×10^{11}	6.051×10^9
10	1.700	7.855×10^{12}	7.138×10^8	3.440×10^8	1.034×10^8
11	1.900	1.491×10^{13}	2.993×10^9	3.059×10^8	2.360×10^6
12	2.100	9.011×10^{12}	1.756×10^{10}	1.353×10^{10}	1.019×10^{10}
13	2.300	8.063×10^{12}	6.185×10^8	4.980×10^8	4.077×10^8
14	2.500	1.036×10^{13}	7.996×10^{10}	6.986×10^9	2.376×10^7
15	2.700	6.626×10^{12}	3.286×10^6	2.225×10^2	4.226×10^{-8}
16	3.00	1.778×10^{13}	2.993×10^9	2.615×10^8	8.897×10^5
TOTAL		7.518×10^{14}	2.613×10^{13}	1.227×10^{13}	4.147×10^{12}

TABLE II. 50,000 MWd/MTM EBR-II Exposure

Group	Group Average Energy MeV	Grwp Production Rate; Photons			
		Shutdown	30 days	75 days	180 days
1	.15	7.298×10^{13}	4.167×10^{12}	1.860×10^{12}	6.514×10^{11}
2	.25	6.115×10^{13}	2.531×10^{11}	1.133×10^{11}	3.601×10^{10}
3	.35	5.402×10^{13}	1.200×10^{12}	7.150×10^{10}	2.355×10^8
4	.475	1.019×10^{14}	6.794×10^{12}	2.495×10^{12}	6.710×10^{11}
5	.65	1.279×10^{14}	6.325×10^{12}	4.168×10^{12}	1.975×10^{12}
6	.825	9.918×10^{13}	1.519×10^{13}	1.060×10^{13}	4.061×10^{12}
7	1.00	4.873×10^{13}	4.338×10^{11}	8.236×10^{10}	2.824×10^{10}
8	1.225	5.317×10^{13}	6.644×10^{10}	2.405×10^{10}	1.038×10^{10}
9	1.475	7.291×10^{13}	2.716×10^{12}	2.578×10^{11}	1.810×10^{10}
10	1.700	7.855×10^{12}	1.077×10^9	5.627×10^8	1.729×10^8
11	1.900	1.491×10^{13}	3.016×10^9	3.088×10^8	2.382×10^6
12	2.100	9.047×10^{12}	5.074×10^{10}	4.322×10^{10}	3.315×10^{10}
13	2.300	8.064×10^{12}	1.997×10^9	1.764×10^9	1.445×10^9
14	2.500	1.036×10^{13}	8.034×10^{10}	7.018×10^9	2.387×10^7
15	2.700	6.626×10^{12}	3.286×10^6	2.225×10^2	4.227×10^{-8}
16	3.00	1.778×10^{13}	3.007×10^9	2.628×10^8	8.938×10^5
TOTAL		7.666×10^{14}	3.729×10^{13}	1.973×10^{13}	7.492×10^{12}

TABLE III. TREAT Transient Exposure of Green Fuel

<u>GROUP</u>	Group Avg. Energy	<u>Shutdown</u>	Group Production Rate: Photons		
	MeV		<u>1 Hour</u>	<u>1 day</u>	<u>10 days</u>
1	.15	9.889×10^{12}	8.793×10^{10}	1.568×10^9	2.394×10^8
2	.25	3.939×10^{12}	5.804×10^{10}	4.560×10^9	1.077×10^8
3	.35	2.936×10^{12}	9.180×10^{10}	4.196×10^8	1.435×10^8
4	.475	6.047×10^{12}	1.206×10^{11}	3.007×10^9	2.406×10^8
5	.65	4.190×10^{12}	1.123×10^{11}	6.988×10^9	2.237×10^8
6	.825	6.810×10^{12}	1.576×10^{11}	1.372×10^9	1.659×10^8
7	1.00	9.909×10^{11}	1.058×10^{11}	9.381×10^8	4.125×10^7
8	1.225	5.436×10^{12}	4.396×10^{10}	9.842×10^8	8.794×10^6
9	1.475	4.392×10^{12}	1.078×10^{11}	5.837×10^8	2.087×10^8
10	1.200	1.176×10^{12}	1.431×10^{10}	4.257×10^7	9.078×10^5
11	1.900	8.033×10^{11}	1.637×10^{10}	4.326×10^8	3.016×10^6
12	2.100	3.366×10^{11}	1.891×10^{10}	7.184×10^7	7.055×10^5
13	2.300	6.324×10^{11}	1.923×10^{10}	3.218×10^6	4.307×10^5
14	2.500	3.513×10^{11}	2.391×10^{10}	1.961×10^7	5.823×10^6
15	2.700	1.106×10^{12}	7.215×10^9	1.335×10^6	2.122×10^4
16	3.00	8.109×10^{11}	1.781×10^{10}	3.799×10^5	2.161×10^5
Total		4.98×10^{13}	1.003×10^{12}	2.099×10^{10}	1.391×10^9

TABLE IV. Fission Gas Concentration (grams)

10 000 MWD/MTM EBR-II

Element	Exposure			
	0	30 days	75 days	180 days
Br	6.563×10^{-4}	6.538×10^{-4}	6.533×10^{-4}	6.533×10^{-4}
Kr	8.260×10^{-3}	8.238×10^{-3}	8.234×10^{-3}	8.224×10^{-3}
I	8.047×10^{-3}	6.338×10^{-3}	6.284×10^{-3}	6.362×10^{-3}
Xe	8.059×10^{-2}	8.124×10^{-2}	8.132×10^{-2}	8.133×10^{-2}
Total	9.756×10^{-2}	9.647×10^{-2}	9.648×10^{-2}	9.656×10^{-2}

50 000 MWD/MTM EBR-II

Element	Exposure			
	0	30 days	75 days	180 days
Br	3.269×10^{-3}	3.266×10^{-3}	3.266×10^{-3}	3.266×10^{-3}
Kr	4.113×10^{-2}	4.110×10^{-2}	4.108×10^{-2}	4.103×10^{-2}
I	3.354×10^{-2}	3.187×10^{-2}	3.187×10^{-2}	3.202×10^{-2}
Xe	4.059×10^{-1}	4.065×10^{-1}	4.066×10^{-1}	4.066×10^{-1}
Total	4.838×10^{-1}	4.828×10^{-1}	4.829×10^{-1}	4.829×10^{-1}

TABLE V. Fission Gas Concentration (grams)

Element	TREAT Transient Exposure (Green Fuel)				
	0	1 hour	1 day	10 days	30 days
Kr	2.128×10^{-7}	8.105×10^{-8}	3.588×10^{-8}	3.553×10^{-8}	3.552×10^{-8}
Br	7.343×10^{-8}	6.738×10^{-9}	1.044×10^{-9}	1.040×10^{-9}	1.040×10^{-9}
I	2.919×10^{-7}	2.513×10^{-7}	1.167×10^{-7}	3.866×10^{-8}	2.135×10^{-8}
Xe	3.708×10^{-7}	1.782×10^{-7}	3.563×10^{-7}	3.864×10^{-7}	3.828×10^{-7}
Total	16.095×10^{-7}	5.173×10^{-7}	5.099×10^{-7}	4.616×10^{-7}	4.407×10^{-7}

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