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REPORT

**INVESTIGATION OF THE COMBINED FAILURE
OF A PRESSURE TUBE
AND DEFECTED FUEL ROD IN PRTR**

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INVESTIGATION OF THE COMBINED FAILURE
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AND DEFECTED FUEL ROD IN PRTR

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INVESTIGATION OF THE COMBINED FAILURE OF A PRESSURE TUBE AND DEFECTED FUEL ROD IN PRTR

INTRODUCTION

On September 29, 1965, a pressure tube ruptured in the Fuel Element Rupture Test Facility (FERTF),⁽¹⁾ of the Plutonium Recycle Test Reactor (PRTR).⁽²⁾ The incident released substantial amounts of fission products into the reactor core and containment vessel, and resulted in gross degradation of the heavy water moderator by light water from the FERTF. Damage to the reactor was slight. A technical review of the incident was instigated to determine major contributing factors to the incident because of the value of the data which were produced and the technical implications of operating oxide fuel at high specific powers. Such a review is reported here.

The fuel testing program in the PRTR preceding the incident included an evaluation of the irradiation performance of vibrationally compacted mixed-oxide fuels operating at high specific powers.⁽³⁾ Nondefected and defected fuel element tests were being conducted in the FERTF at high specific powers with maximum fuel temperatures above the melting point. At the time of the incident, a fuel element with an intentionally defected UO_2 -4 wt% PuO_2 fuel rod was being irradiated in the FERTF. After the incident, it was found that the defected rod had ruptured and had lost an appreciable amount of fuel material.

An important factor in determining the cause of the incident was recognized to be a detailed evaluation of the fuel rupture and its relation to the rupture of the pressure tube. However, it was found that an exact description of the time sequence of the fuel and pressure tube ruptures was not immediately apparent. Therefore, detailed postexaminations have been made of the fuel element, the pressure tube, and the thermal-hydraulic conditions which existed at the time of the incident. The results of these examinations are presented in this report along with an explanation of the events which are thought to be the most likely cause of the incident.

SUMMARY AND CONCLUSIONS

The mechanism that resulted in the rupture of an intentionally defected $\text{UO}_2\text{-PuO}_2$ fuel element operating under molten-core conditions and a Zircaloy pressure tube in the PRTR involves a complicated and not fully understood sequence of events. Analysis of observations and data generated as a result of postmortem examinations permits a plausible rupture mechanism to be hypothesized without serious contradictions. The authors chose to postulate a single rupture mechanism although it is recognized that variations in this basic mechanism may exist. The proposed fuel element rupture mechanism and sequence of events that ultimately resulted in the incident are presented as follows:

- (1) Preirradiated rod FR-4 was defected by drilling a hole through the cladding.
- (2) Water entered the rod during loop activation and operation
- (3) The $\text{UO}_2\text{-PuO}_2$ fuel became oxidized during irradiation, causing the O:U ratio to increase from 2.01 to 2.1. This caused a decrease in the thermal conductivity and possibly the melting temperature of the fuel. Very recent data indicate that in the presence of water vapor, UO_{2+x} might exist at higher temperatures than previously considered possible, and that its formation can be very rapid.
- (4) An unexpectedly large amount of fuel melting occurred because of the reduced thermal conductivity and lower melting point of the oxidized fuel.

- (5) The volume expansion of the unexpectedly large amount of molten fuel could not be accommodated by the void space within the fuel rod and resulted in high internal hydrostatic pressures.
- (6) The high internal hydrostatic pressures and higher than normal cladding temperatures caused the cladding to swell locally and split in a ductile manner in the vicinity of the defect.
- (7) High temperature fuel contacted the pressure tube and probably the coolant in the vicinity of the ductile split. (The pressure tube did not fail at this point, probably because of coolant accessibility to the point of contact.) This caused steam formation at this point and immediately downstream for approximately 24 sec before the pressure tube failed.

- (8) The localized swelling of the defected rod and steam formation caused a heat transfer disruption that permitted overheating of a larger area of downstream cladding during the 24 sec period.
- (9) As the region of the fuel rod immediately downstream of the swelled area overheated due to steam blanketing the adjacent Zircaloy pressure tube overheated and failed under pressure. It is postulated that the pressure tube overheated by radiant heat transfer from the overheated fuel rod or by direct contact with a skull of hot fuel and cladding from the ruptured fuel rod. There is no evidence of fuel contacting the pressure tube in the immediate vicinity of the rupture, and there was no evidence of fuel loss to the FERTF coolant prior to the failure of the pressure tube.
- (10) After the pressure tube failed, fuel (mostly molten) was forceably ejected from the rod under the pressure differential created between the inside of the rod and the decreased FERTF coolant pressure. Not all the fuel escaped through the hole in the pressure tube since an estimated 75% of the 700 g of fuel ejected from the rod remained in the rupture loop system.

It is interesting to note that the actual damage to the reactor was minor. Some equipment required maintenance due to being flooded, but no major components required replacement except the ruptured pressure tube. The cost of the incident arises primarily from the expense of replacing the degraded heavy water moderator, and from the time consuming task of decontaminating the reactor.

The extensive postmortem examination associated with the incident has provided a large amount of valuable information related to the irradiation behavior of vibrationally compacted mixed-oxide fuel rods operating under high power generating conditions in a water-cooled environment. Most noteworthy, perhaps, are the indicated behavioral differences between nondefected and defected mixed-oxide fuel rods and apparent differences in the physical behavior of the fuel in nondefected vibrationally compacted rods presumably operating under the same conditions.

Also, it is believed that the proposed rupture mechanism and the observed effects are not necessarily unique with vibrationally compacted $\text{UO}_2\text{-PuO}_2$ fuel and a similar sequence of events could have occurred with pellet-containing fuel rods. These results point out important areas of fuel behavior that will be intensively investigated during the High Power Density Program in PRTR.

BACKGROUND INFORMATION

Prior to the incident discussed in this report, the fuel testing program in PRTR had been expanded to include an evaluation of the irradiation performance of vibrationally compacted impacted mixed-oxide fuels operating at high specific powers.⁽³⁾ The High Power Density (HPD) Program also includes the Batch Core Experiment in PRTR to evaluate the physics parameters of a plutonium enriched fuel loading as a function of burnup under controlled conditions. One of the primary objectives of the fuels development part of the program is to determine if there are factors that limit plutonium-bearing fuels to operating conditions less than those attainable with equivalent uranium-only oxide fuels.

The fuel elements for the HPD program are 75.6 in. (192 cm) long, Zircaloy-clad 19-rod cluster fuel elements containing 58.5 in. of pneumatically impacted UO_2 -2 wt% PuO_2 high density fuel material vibrationally compacted to a bulk density 85-87% TD (Figure 1). The HPD fuel element was designed to include a plenum to accommodate all the gases potentially releasable from the fuel at rod powers as high as 30 kW/ft (fuel molten to an estimated 60% of the radius) at a burnup of 15,000 MWd/MT fuel (3.8×10^{20} fissions/cm³).

The 19-rod cluster fuel elements are centered in the Zircaloy-2 pressure tubes (3.250 in. ID), which have a 0.150 in. thick wall, by spacing gussets on the top and bottom end brackets (Figure 1). The diameter over the rods of the element is nominally 3.004 in. which provides a radial clearance of 0.123 in. between the fuel rods and the pressure tube. Six of the rods in the outer 12-rod ring are spirally wrapped with a 0.070 in. diameter spacing wire to prevent contact between rods and between the rods and the pressure tube.

Prior to the start of the Batch Core Experiment, HPD elements were irradiated in the PRTR core at nearly the maximum operating conditions expected during Batch Core Experiment, i. e., maximum rod power generations of 19-20 kW/ft with fuel temperatures as high as 2600-2800 °C. The maximum burnup of the 11 HPD elements currently in the reactor is 4950 MWd/MT fuel, average. Under these conditions, a

Corrosion Resistant
Cladding
and Hardware

Overall Length of
Element - 192.4 cm
Fuel Column
Length - 149.2 cm
Fuel Rod OD - 1.44 cm
Wall Thickness - 0.76 mm

Six Circumferential
Bands

Fission Gas
Plenum 18 cm
Long Supported
by an Inconel
Spring

Depleted UO_2
Pellet

Vibrationally
Compacted UO_2 -
4 wt% PuO_2
Impacted Fuel
(85 - 87% TD)

Spiral Wire
Wraps on a
10 in. Pitch

Depleted UO_2
Pellet

Top end cap

Plenum Support
Spring

1cm
1cm

Depleted UO_2
Pellet

UO_2 -1 wt% PuO_2
Powder

3cm

UO_2 -4 wt% PuO_2
Powder

1cm

UO_2 -1 wt% PuO_2
Powder

Depleted UO_2
Pellet

Bottom end cap

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FIGURE 1. PRTR High Power Density Fuel Element

central void forms over essentially the full length of the rod, high density columnar grains form, and homogenization of the mechanically mixed and impacted $\text{UO}_2\text{-PuO}_2$ fuel occurs near the thermal center. Nondefected and defected fuel element tests were conducted in the FERTF at maximum rod power generations of approximately 27 kW/ft with maximum fuel temperatures above melting.

In order to achieve the desired rod power generations during the pre-Batch Core test period, the enrichment of HPD test elements was increased from 2 wt% to 4 wt% PuO_2 and the six driver elements surrounding the FERTF were removed to create a flux trap in the central region of the PRTR core. Increasing the PuO_2 enrichment increases the neutron self-shielding and the degree of nonuniformity of radial power generation within rods. This effect results in slightly lower maximum fuel temperatures and smaller molten fuel volumes for comparable rod power generations. At 25 kW/ft, for instance, it was calculated that increasing the PuO_2 concentration from 2 to 4 wt%, decreases the calculated molten fuel zone by only 4% of the radius. It was, therefore, concluded that increasing the PuO_2 concentration in HPD test elements from 2 wt% to 4 wt% PuO_2 does not create a significant perturbation in the thermal conditions that exist in a 2 wt% PuO_2 element, and the test results were deemed valid for the 2 wt% PuO_2 elements.

The PRTR core loading through September 29, 1965 caused flux peaking in the ends of prototype HPD test elements. The degree of peaking will be reduced when the entire core is loaded with HPD elements at the start of the Batch Core Experiment. Prototype HPD test elements were loaded with graded enrichment in the ends of the rods to avoid high fuel temperatures adjacent to the end caps. The top end was loaded with a 1 cm thick depleted UO_2 pellet, followed by 1 cm of $\text{UO}_2\text{-1 wt% PuO}_2$ powder, followed by the $\text{UO}_2\text{-4 wt% PuO}_2$ fuel material. The bottom end, which initially experienced more severe flux peaking, was loaded with a 1 cm thick depleted UO_2 pellet, followed by 3 cm of $\text{UO}_2\text{-1 wt% PuO}_2$ powder fuel material.

This loading scheme alleviates the heat generation limitations in the end regions of the fuel rods and maximum fuel temperatures are generated near the midplane of the elements. Recent postirradiation examination results indicate that complete separation between the graded fuel materials in the end regions was not maintained.

The HPD test elements had three replaceable outer rods which permit the performance of more than one experiment with the same element by means of rod replacement. Some of the HPD elements tested had flux wires attached along their length to permit an accurate determination of the axial flux profile.

Vibrationally compacted UO_2 -4 wt% PuO_2 HPD element (FE-6504) was irradiated in the FERTF to a burnup of $920 \text{ MWd/MT}_{\text{fuel}}$ (31.8 MWd) at a maximum tube power of 1745 kW (average approximately 1650 kW) before being intentionally defected. This tube power produced a maximum peak rod power of 25 to 26 kW/ft for a short period of time. The maximum rod power was approximately 24 kW/ft for most of the irradiation period. Anomalous PRTR power fluctuations were first noted during this irradiation period. The element successfully withstood a 20% overpower transient which occurred in approximately 11 sec during intended steady-state operation at 1685 kW. Fuel rod positions within FE-6504 during this first irradiation period are indicated in Figure 2.

A UO_2 -2 wt% PuO_2 HPD element (FE-6505) was irradiated for a short time ($115 \text{ MWd/MT}_{\text{fuel}}$ or 4.2 MWd) during the next operating period in the FERTF at a maximum tube power near 1750 kW to study the nature of the reactor control instability phenomenon. The results of this test indicated the probable presence of nucleate boiling on the surface of the fuel rods. Reactor instability was traced to mechanical problems in the control system and was essentially eliminated after system modifications. Instability problems may have been aggravated by nucleate boiling.

Rod Position

(a)	FE-26	(k)	FB-83
(b)	FB-26	(l)	FB-55
(c)	FB-96	(m)	FB-72
(d)	FB-28	(n)	FR-4
(e)	FB-17	(o)	FB-74
(f)	FB-07	(p)	FB-29
(g)	FB-91	(q)	FB-67
(h)	FB-59	(r)	FR-3
(i)	FB-06	(s)	FB-41
(j)	FR-6		

FIGURE 2. Rod Positions in FE-6504 During the First Irradiation Period Before the Defect Test

Two replaceable rods (FR-3 and FR-6) were removed from FE 6504 for examination and were replaced by nonirradiated rods FR-10 and FR-11. A third preirradiated replaceable rod (FR-4) was deliberately defected with a 1/16 in. diameter hole near the midplane for continued irradiation in the FERTF. Fuel rod positions within FE-6504 during the second irradiation period are indicated in Figure 3. The hole was located 30 3/16 in. down from the square shoulder on the top end cap of the rod which placed it approximately 7 in. above the geometric center of fuel column and the maximum flux position.

A stepwise increase in fuel power was conducted with defected FE-6504 in the FERTF. The reactor achieved criticality at approximately 0549 on September 25, 1965, and the power level was held at 7 MW for about 30 hr due to fuel loading excess reactivity. Sufficient xenon buildup was achieved by 8-4 shift on September 26, 1965, and the power level was increased to 40 MW during that shift. The power level increases are summarized in Table I.

TABLE I. Power Level Increases

<u>Date</u>	<u>Shift</u>	<u>New Reactor Power Level, MW</u>	<u>Approximate FERTF Power, kW</u>	<u>Maximum Rod Power, kW/ft</u>
September 26	8-4	40	950	14.4
September 26	4-12	45	1070	16.2
September 27	12-8	49.3	1350	20.5
September 27	8-4	56.5	1550	23.5
September 28	12-8	60.9	1650	25
September 28	4-12	65.3	<1800	27.3

Radioactivity of the FERTF coolant began to increase as soon as reactor operation began due to the deliberate defect in the fuel element cladding. The coolant activity reached a near equilibrium value prior to each power level increase. A large activity burst occurred during the power level increase from 49.3 to 56.5 MW, probably contributed to by the onset of fuel melting. Small bursts of activity also occurred during other power level increases and during FERTF outlet temperature changes.

Rod Position			
(a)	FE-26	(k)	FB-83
(b)	FB-26	(l)	FB-55
(c)	FB-96	(m)	FB-72
(d)	FB-28	(n)	FR-10
(e)	FB-17	(o)	FB-74
(f)	FB-07	(p)	FB-29
(g)	FB-91	(q)	FB-67
(h)	FB-59	(r)	FR-11
(i)	FB-06	(s)	FB-41
(j)	FR-4 (Defected Rod)		

FIGURE 3. Rod Positions in FE-6504 During the Defect Test

In connection with the continued investigation of the reactor instability phenomenon, after each stepwise increase at 56.5, 60.9, and 65.3 MW, the reactor power level was held constant while the FERTF test section outlet temperature was reduced in 50 °F steps from 500 to 300 °F and returned to 500 °F in similar steps. Reactor control remained stable throughout these tests.

The maximum tube power during the second operating period was approximately 1795 kW with an associated maximum rod power of approximately 27 kW/ft. Plans were to irradiate the element for an extended period of time at the maximum power condition. The PRTR incident of September 29, 1965 occurred during the second operating period. The reactor scrammed due to a FERTF Test Section-Low Inlet Pressure trip approximately 5 min after the coolant outlet temperature reached 500 °F at the completion of the thermal cycle. The element accumulated an additional burnup of 120 MWd/MT (total of 1040 MWd/MT) during the second operating period prior to the incident.

Several of the pertinent reactor and FERTF variables which were being recorded at the time of the incident have been plotted with a common time scale in Figure 4. The variables plotted were the reactor power as indicated by neutron flux, the FERTF outlet coolant temperature, the pressure drop across the FERTF test section, and the FERTF coolant activity. All the data were obtained from low-speed (approximately 1 in./hr) instruments up to 24 sec before the reactor scram. The FERTF pressure drop, FERTF flow, and PRTR neutron flux data for the 28 sec preceding the reactor scram were obtained from high-speed (125 cm/hr) instruments (Figure 5).

FIGURE 4. Time Relationship of Loop and Reactor Data Prior to the Incident

FIGURE 5. High Speed Recording of FERTF Outlet Flow, FERTF Test Section Δp , and PRTR Neutron Flux Prior to and at the Time of the Incident

FUEL EVALUATION

GENERAL INFORMATION

Eight fuel rods (FR-3, FR-6, FR-4, FB-06, FB-83, FB-59, FB-55 and FR-10) were removed from the 12-rod ring of FE-6504 for postirradiation examination. Two rods (FR-3 and FR-6) were removed from the element for examination after the first irradiation period, and the remaining six were removed after the incident to provide information related to the cause of the rupture. The extent of postirradiation examination work performed on each rod varied, depending upon the purpose of the examination and what was found during the examination. In general, however, the work performed on most of the rods included diameter measurements, determination of the balance point, fission gas release, and examination of the fuel structures formed during irradiation including transverse sections along the length of the rods, as well as longitudinal sections at the top and bottom ends of the rods. The postirradiation examination information is summarized in Table II.

FE-6504 was discharged from the FERTF on October 27, 1965. Low flow coolant water was maintained in the FERTF from the time of the incident until discharge of the element. Examination of the element in the basin revealed that a section of cladding approximately 2.7 in. long and almost 0.5 in. wide at the widest point was missing from the ruptured rod. The missing area of the cladding extended from 28.3 in. to 31.0 in. from the square shoulder on the top end cap and included the 1/16 in. diameter hole originally drilled through the cladding to defect the rod (Figure 6).

TABLE II. Summary of Postirradiation Examination Results of Rods from FE-6504

	Rod FR-3 *	Rod FR-6*	Rod FR-10**	Rod FR-4†	Rod FB-06	Rod FB-83	Rod FB-59	Rod FB-55
Central Void, 20 in. from top, %r	--	--	--	67	32	23	38	37
Central Void, 30 in. from top, %r (Plane of defect)	--	--	--	73 60	26	76	8	17
Central Void, 20 in. from bottom, %r	--	--	--	57	4	6	0	0
Fuel Melting, 20 in. from top, %r	--	--	--	67	32	23	50	45
Fuel Melting, 30 in. from top, %r (Plane of defect)	--	--	--	73 75	50	46	64	62
Fuel Melting, 20 in. from bottom, %r	--	--	--	73	57	54	64	63
Localized overheating of the cladding in top end region	No	No	--	No	Yes	No	Yes	Yes
Axial upward movement of top depleted UO ₂ pellet, in.	--	--	--	0.15	0.6 to 0.8	0.0	0.5 to 0.6	0.54
Fission gas release, % Xe + Kr	81	75	--	--	82	79	88	88
Rod Swelling	No	No	No	No	No	No	Yes	No
Balance point change	No	No	No	--	--	--	Yes	--

* Irradiated during the first operating period of FE-6504. The fuel structure analysis data are not included for these two rods since they were not sectioned at the axial positions as indicated in the Table.

** Irradiated only during the second operating period of FE-6504.

† Ruptured rod.

FIGURE 6. Cladding Rupture in Rod FR-4 from HPD FE-6504. Rupture Occurred in the Area of the Rod Previously Defected with a 1/16 in. diam Hole (1 X)

Crevice corrosion spots formed on the surface of the fuel rods at the contact points between the circumferential strip bands and fuel rods during the first cycle of irradiation of FE-6504 before defecting. Corrosion spots from one of the original bands were located 28.5 in. down from the square shoulder on the top end cap near the rupture area. The original circumferential bands were removed from the element to facilitate the rod manipulations and defecting. The remotely replaced bands were not located in the same positions as the original bands, and hence, none were in the immediate vicinity of the rupture. The nearest replaced band was located approximately 3.5 in. below or upstream from the defect. (FERTF coolant flows up.)

Four rods (FB-06, FB-59, FB-29, and FB-55) in the outer ring of FE-6504 showed evidence of localized overheating on the outside surface of the cladding approximately 8 1/4 in. down from the square shoulder on the top end cap after the second irradiation period (Figure 7). The overheated cladding areas have the appearance of having experienced incipient boiling burnout and are found only on the outside of the rods in the outer ring of the element. The size of the overheated regions varies slightly; however, the cladding has deformed inward over an axial distance of approximately 0.2 in. and in one instance, extends approximately 1/3 around the circumference of the rod. The overheated area of the cladding is located slightly below or upstream (approximately 0.2 to 0.4 in.) from the original and replaced circumferential bands and slightly above or downstream (approximately 0.2 in.) from the preirradiated pellet-fuel interface. (Due to inaccuracies associated with positioning the circumferential bands remotely, the replaced top band could have been located directly over and possibly associated with the areas of localized overheating.) The fuel column has expanded upward 0.5 to 0.8 in. in those rods which show evidence of localized overheating in the top end regions. This will be described in more detail in the discussions related to the individual fuel rods.

D-5795

4.6 X

FIGURE 7. Overheated Cladding
Near the Top of Rod FB-55 in
FE-6504

The Zr-Co flux monitor wire was attached along the full length of FE-6504 during its first irradiation cycle in the FERTF before being intentionally defected. A Co^{60} activation traverse of the wire yielded a peculiarly shaped axial flux profile (Figure 8). The reason for the unusually shaped flux profile is unknown; however, it is possible that the monitor wire was not located at the same radial position within the element along its entire length.

EXAMINATION OF RODS FR-3 AND FR-6

Two replaceable rods (FR-3 and FR-6) from FE-6504 were destructively examined after the initial irradiation period in PRTR to a burnup of 920 MWd/MT_{fuel} at a maximum tube power of 1745 kW. This tube power produced a maximum rod power estimated to be 25 to 26 kW/ft; however, the maximum rod power during most of the irradiation was approximately 24 kW/ft.

Rod FR-3 was sampled for fission gas in the fuel region, and rod FR-6 was sampled in the plenum region. Each rod yielded approximately the same amount of total gas - 218 and 208 ml (STP) - respectively, indicating good communication between the fuel and plenum. Mass spectrographic analysis of the gas collected indicated that 81% of the Ze+Kr formed in rod FR-3 was released, and 75% of the Xe+Kr was released from rod FR-6. There were no measurable changes in diameter or balance point in the two rods indicating that no swelling or gross axial fuel movement occurred under these operating conditions.

EXAMINATION OF ROD FR-3

Examination of the bottom end region of rod FR-3 showed that the depleted UO_2 pellet was essentially unaffected (only minor cracking) as a result of irradiation (Figure 9). The 3 cm long 1 wt% PuO_2 fuel region had undergone sintering with a maximum fuel centerline temperature estimated to have been 1700 °C indicated. Limited fuel melting occurred in the 4 wt% PuO_2 region with the formation of a very uniform central void over the 8.5 cm distance examined (Figure 10).

FIGURE 8. Co^{60} Activation Traverse of FE-6504 During the Irradiation Period Before the Defect Test

16674
Photomicrograph

667
 β - γ Autoradiograph

FIGURE 9. Longitudinal Section of the Bottom End Region of Rod FR-3 Showing the Depleted UO₂ Pellet Next to the Zircaloy End Cap and the UO₂-1 wt% PuO₂ Graded Enrichment Powder Fuel (4X)

16675
Photomicrograph

668
 β - γ Autoradiograph

FIGURE 10. Longitudinal Section Near the Bottom End Region of Rod FR-3 Showing the Interface Between the UO_2 -1 wt% PuO_2 Graded Enrichment and the 4 wt% PuO_2 Fuel Material (4 X)

Typical fuel structures were formed during irradiation. The high activity indicated in some cracks by the β - γ autoradiograph* shows possible evidence of the flow of liquid fuel into radial cracks that presumably existed during irradiation. Other cracks, which probably formed during cooldown, show no evidence of penetration by molten fuel. This effect has only been observed under marginal melting conditions.

Two transverse cross sections from the central region of rod FR-3 were examined. Examination of the fuel structures formed during irradiation indicates that fuel was molten to as much as 30% of the radius during operation (Figure 11). This is less than the calculated value of 45% for a linear rod power of 24 kW/ft. The amount of melting and the size of the central void are greater at the plane of the fuel rod above the midpoint than at the midpoint. (In general, previous experience has shown that the size of the central void above the midplane in long fuel rods that have undergone some degree of fuel melting is larger than it is at or below the midplane, thus suggesting some degree of axial fuel relocation).

Fuel structures in the top region of rod FR-3, however, indicated higher fuel temperatures than at the bottom of the rod as a result of flux peaking (Figure 12). This situation was probably aggravated by inadvertent mixing of the UO_2 -1 wt% PuO_2 graded enrichment fuel with the 4 wt% PuO_2 fuel in the top end region. There was no evidence of reaction between the Inconel-X plenum spring and the depleted UO_2 pellet. However, the UO_2 pellet at the top was more severely cracked than was the one at the bottom, indicating higher temperatures in the top pellet.

* The β - γ autoradiographs shown in this report were made by exposing the irradiated fuel specimens to high-resolution film plate and indicate the presence of fission products. Alpha autoradiographs (4) were made by exposing the irradiated specimens to cellulose acetate and indicate the presence of plutonium or an isotope associated with the presence of plutonium. Dark areas on the autoradiographs indicate regions of high activity. A uniform concentration of spots is present in some of the α -autoradiographs caused by artifacts in the cellulose acetate.

Limited diffusion of plutonium-containing fuel into the depleted UO_2 pellet occurred. In previous irradiations, higher fuel temperatures occurred in the bottom end regions than at the top. Even though high fuel temperatures and localized melting occurred in the top end region of the rod, no condition that would compromise fuel element integrity during operation was inferred.

The fuel was sampled by microdrilling in two areas of the longitudinal specimen from the top of rod FR-3 (Figure 12). Radiochemical analysis of the samples showed that they contain 3.8 and 3.4 wt% PuO_2 . Although the nominal PuO_2 concentration is 4 wt%, the analytical results are biased low. The results confirm that the UO_2 -1 wt% PuO_2 graded enrichment became mixed with the UO_2 -4 wt% PuO_2 fuel material. Evidence of mixing is also indicated on the β - γ autoradiograph.

EXAMINATION OF ROD FR-6

Examination of three transverse cross sections from rod FR-6 indicates that fuel temperature conditions in the central region of this rod during irradiation were similar to those that existed in rod FR-3. Fuel structures formed during irradiation indicate that fuel was molten to as much as 29% of the radius in the plane above the midpoint of the fuel rod. The size of the central void above the midplane is larger than the central void at, and below, the midplane. Fuel was molten to approximately 22% of the radius at the midplane of rod FR-6, as compared to 17% at the midplane of rod FR-3. This is not considered to be a significant difference.

Indicated fuel temperatures in the top end region of rod FR-6 were lower than those in the top of rod FR-3, even though the irradiation conditions were identical and autoradiographs show that the 1 wt% PuO_2 and 4 wt% fuels were mixed to the same degree (Figure 13). The lower average fuel temperatures in rod FR-6 are also reflected by the slightly lower fission gas release, i. e., 75% Xe+Kr release for rod FR-6 vs. 81% Xe+Kr release for rod FR-3. There is no evidence of reaction between the Inconel-X plenum support spring and the depleted UO_2 pellet in the top region of this rod. The top depleted UO_2 pellet, however,

D-1776
Photomicrograph

670
 β - γ Autoradiograph

FIGURE 11. Transverse Section at the Maximum Flux Position of Rod FR-3. Melting Indicated to 30% of the Radius (4.8 X)

16716
Photomicrograph

682
 β - γ Autoradiograph

FIGURE 12. Longitudinal Section of the Top End Region of Rod FR-3.
Holes are Microdrilled Chemical Analysis Samples (4 X)

16717
Photomacrograph

719
 β - γ Autoradiograph

718
 α Autoradiograph

FIGURE 13. Longitudinal Section of the Top End Region of Rod FR-6.
 β - γ and α -Autoradiographs Show Mixing of the UO_2 -1 wt% PuO_2 Graded
Enrichment with the UO_2 -4 wt% PuO_2 Fuel Material (4 X)

cracked rather extensively due to thermal stresses. Limited diffusion occurred between the UO_2 pellet and the UO_2 - PuO_2 fuel material in the top end region. There is no evidence of fuel melting in the top end region, and fuel structures are as anticipated and indicate a normal condition.

EXAMINATION OF RUPTURED ROD FR-4

The ruptured rod FR-4 was examined more extensively than any of the other rods from FE-6504. Three transverse sections of the Zircaloy cladding were examined in the vicinity of the rupture. Four transverse sections including the fuel and cladding from along the length of the rod were examined. A longitudinal section from immediately above and below the rupture area, a longitudinal section of the bottom end region, and four longitudinal sections of the top plenum region of the rod were examined. The results of these examinations are described below.

Cladding Examination in the Vicinity of the Rupture

A section of cladding approximately 2.7 in. long and almost 1/2 in. wide at the widest point was missing from the ruptured rod (Figure 6). The missing area includes the 1/16 in. diameter hole originally drilled through the cladding to defect the rod after its first period of irradiation in the FERTF. A discolored band surrounds most of the rupture area indicating that the cladding was overheated.

Except for the area in the immediate proximity of the ruptured region, there were no measurable diameter changes along the length of the rod. The ruptured region appears to consist of two separate areas (Figure 6), one large area of missing cladding without appreciable associated distortion and another area, separated from the larger one by a nearly complete isthmus of cladding, which has the appearance of an opened-up longitudinal split with considerable associated cladding deformation.

Three transverse sections were made through the rupture area (Figure 14). The specimens were autoradiographed to determine the presence and location of adhered fuel material and examined metallographically to determine the cladding structure, the extent of oxide layer formation, and the degree of hydriding in the rupture region.

Autoradiographs confirmed the presence of once-molten $\text{UO}_2\text{-PuO}_2$ fuel material adhered to the inner and outer surfaces of some fracture areas and, in some cases, the fuel material apparently alloyed with once-molten (as evidenced by a porous cast structure) Zircaloy cladding at the edge of the break (Figure 15).

Metallographic examination of the Zircaloy cladding in the vicinity of the rupture shows no appreciable increased hydride concentration. Hydride platelets appear only in the undisturbed regions of the cladding, and in most areas, the concentration is approximately the same as it was prior to irradiation. No localized massive hydride concentrations have been observed.

Photomicrographs of the fracture areas of sample B-7 (Figure 16) show once-molten fuel material adhered to the outer surface of the cladding. The Zircaloy cladding has distorted appreciably near the edge of the fracture. Beta-phase Zircaloy, indicative of temperatures in excess of 1000°C , formed near the edge of the fracture. The ZrO_2 layer is thickest near the edge of the break and tapers off rapidly away from the fracture. A metallic-appearing oxygen-rich diffusion layer which has the same contour as the ZrO_2 layer is present under the ZrO_2 layer. In some instances, as illustrated in Figure 16 B-7B, the ZrO_2 layer and the oxygen-rich layer extend around the end of the break, possibly suggesting that the fracture existed at this point for a finite period of time before gross rupture occurred although the reaction could have occurred in a very short period of time at the temperatures indicated. Section B-7 (Figure 16) is unusual in that one of the fracture surfaces (B-7B) appears

FIGURE 14. Ruptured Cladding Area of Rod FR-4 Showing Location of Metallographic Specimens (1X)

D-2896

795

FIGURE 15. Transverse Section B-6 Through the Ruptured Region of Rod FR-4. High Activity as Indicated on the β - γ Autoradiograph Corresponds to Areas Where Molten Fuel Material has Adhered to and/or Alloyed with the Cladding (4 X)

FIGURE 16. Fracture Areas of Cladding Specimen B-7 (75X)

as described above, whereas there is no evidence of adhered fuel material or a ZrO_2 layer on the other fracture surface (B-7A).

Autoradiographs of both fracture surfaces of Section B-6 (Figure 17) show evidence of fuel material alloying with the Zircaloy cladding at the edge of the fracture. There is no evidence of fuel material adhering to the cladding surfaces in the fracture regions of Section B-6. The cladding fracture surface shown in Section B-6A distorted considerably as a result of melting. The once-molten Zircaloy has a porous cast structure. Again, the ZrO_2 layer extends around the end of fracture surface B-6A similar to Section B-7B. The oxygen-rich layer, however, does not appear in those regions of the fracture that have alloyed with fuel material.

The structure of Section B-6B is similar to Section B-6A, except the cladding has not distorted appreciably, and the ZrO_2 layer does not extend around the end of the fracture surface.

Photomacrographs and autoradiographs of Section B-4 (Figure 18) show that once-molten fuel has adhered to the inside surface of Section B-4A, and no fuel has alloyed with the cladding in either Section B-4A or B-4B. The ZrO_2 layer and associated oxygen-rich layer do not extend around the end of the fracture surfaces of either Section B-4A or B-4B. Section B-4A has undergone some distortion near the edge of the fracture. The oxygen-rich layer shown in this section continues on both the inside and outside surfaces of the cladding further away from the fracture than shown in other sections. A thin layer of transformed Zircaloy, associated with the oxygen-rich layer, is also present on the inside and outside surfaces of the cladding for some distance away from the fracture in Section B-4A.

Fuel Examination

Seven longitudinal and four transverse specimens from the ruptured rod were examined to evaluate the condition of the fuel material remaining in the rod.

FIGURE 17. Fracture Areas of Cladding Specimen B-6 (75 X)

D-3667

B-4A

D-3066

B-4B

FIGURE 18. Fracture Areas of Cladding Specimen B-4 (75X)

As determined by weight difference, more than 700 g of $\text{UO}_2\text{-PuO}_2$ fuel material, or 40% of the total amount originally present, was expelled from the rod. Approximately 5 in. of fuel are missing completely in the immediate vicinity of the rupture. It is estimated that approximately 75% of the released fuel remained in the rupture loop system, and the other 25% escaped through the hole in the pressure tube.

The longitudinal section of the bottom end region of the ruptured rod shows that the depleted UO_2 pellet and approximately 3 cm of $\text{UO}_2\text{-1 wt% PuO}_2$ fuel material were effective in limiting fuel temperatures near the end cap (Figure 19). The small amount of $\text{UO}_2\text{-PuO}_2$ fines that settled between the pellet and the Zircaloy end cap during fabrication had no apparent effect on the operation of the fuel rod. Minor cracking occurred in the depleted UO_2 pellet as a result of thermal stresses during irradiation. Fuel melting occurred in the $\text{UO}_2\text{-1 wt% PuO}_2$ graded fuel to within 0.11 in. of the bottom end cap. A greater degree of melting occurred in the $\text{UO}_2\text{-4 wt% PuO}_2$ fuel region, as expected.

An abnormally large central void formed in the fuel in the lower section of the ruptured rod. The central void below the rupture is unobstructed for a distance greater than the 24 in. examined. Examination of a transverse section of the ruptured rod at a plane approximately 20 in. from the bottom of the rod or 16 in. below the end of the rupture shows an abnormally large void (57% of the radius) and melting to approximately 73% of the radius (using the porous fuel structure as being indicative of the once-molten region) (Figure 20). A 0.067 in. thick layer of the porous fuel structure or once-molten region lines the inside surface of the central void (Figure 21). A thin uniform reaction layer approximately 0.001 in. thick is present on the inside surface of the cladding (Figure 22).

FIGURE 19. Longitudinal Section at the Bottom End of Ruptured Rod FR-4
(4 X)

D-2813
Photomicrograph

769
 β - γ Autoradiograph

770
 α Autoradiograph

FIGURE 20. Transverse Section 20 in. from the Bottom of Ruptured Rod FR-4. Melting Indicated to 73% of the Radius (4.8 X)

FIGURE 21. Transverse Section 20 in. from the Bottom of Ruptured Rod FR-4 (21 X)

FIGURE 22. Transverse Section 20 in. from the Bottom of Ruptured Rod FR-4 (250 X)

The longitudinal section of the rod immediately below the rupture shows a thin layer or skull of once-molten fuel on the inner surface of the central void, which covers 60% of the radius at this point (Figure 23). Melting is indicated to 75% of the radius in the fuel remaining closest to the rupture. The fuel ends abruptly adjacent to the rupture and, in general, has the same structural characteristics as the fuel further down the ruptured rod.

In that portion of the rod above the rupture, the size of the central void is even larger than it is below the rupture. Structurally, the fuel above and below the rupture is similar, except that in the region above the rupture, the fuel is missing completely to the outer edge of the porous or once-molten region. Also, the large central void extends uninterrupted over the entire section of the rod above the rupture.

The longitudinal section of the rod immediately above the rupture shows that the remaining fuel ends abruptly near the rupture (Figure 24). There is some evidence of a thin layer or skull of once-molten fuel on the inner surface of the central void; however, the amount of skull is not so extensive as it was in the region immediately below the rupture. The size of the central void, which in this case is the same as the melt zone, extends to 73% of the radius at the lower end of the fuel column near the rupture.

Examination of a transverse section approximately 20 in. from the top of the rod or 8 in. above the end of the rupture shows that the central void, which consumes 67% of the radius, extends to the outer edge of the porous fuel region (Figures 25 and 26). A thin reaction layer formed on the inside surface of the cladding in the upper portion of the ruptured rod similar to that which formed below the rupture.

As mentioned previously, the central void extends uninterrupted along the full length of the fuel column in that portion of the rod above the rupture. In fact, molten fuel eroded completely through the depleted UO_2 pellet at the top of the fuel column and entered into the gas plenum region (Figure 27).

16757
Photomicrograph

767
 β - γ Autoradiograph

768
 α Autoradiograph

FIGURE 23. Longitudinal Section of Rod FR-4 Immediately Below
the Rupture (4 X)

FIGURE 24. Longitudinal Section of Rod FR-4 Immediately Above the Rupture (4 X)

D-2812
Photomicrograph

763
 β - γ Autoradiograph

764
 α Autoradiograph

FIGURE 25. Transverse Section 20 in. from the Top of Ruptured Rod FR-4. Melting and Central Void Indicated to 67% of the Radius (4.8 X)

FIGURE 26. Transverse Section 20 in. from the Top of Ruptured Rod FR-4 (21 X)

Fuel Extended into
the Plenum to
This Level

Photomacrograph

β - γ Autoradiograph

α Autoradiograph

Inconel-X Spring

Depleted UO_2
Pellet

066088

FIGURE 27. Longitudinal Section of the Fission Gas Plenum Region in the Top of Ruptured Rod FR-4 (~ 1X)

There is evidence of melting on the inside surface of the depleted pellet and solidified molten fuel on the Inconel-X spring (Figures 28 and 29). There was reaction between the molten fuel and the Inconel-X spring, and in one region near the top of the molten fuel column, two coils of the spring were almost completely consumed by reaction with the fuel. Measurements indicate that the top depleted UO_2 pellet has moved upward approximately 0.15 in. against the downward force of the plenum spring. The molten fuel column extended approximately 4.2 in. above the depleted UO_2 pellet or 60% of the plenum length. Particles of fuel, which apparently came from the nonmolten region above the rupture, are embedded in the mounting material in the top longitudinal section near the end cap. These pieces probably broke loose during handling and shipping of the ruptured rod prior to impregnation and sectioning.

The postirradiation O:U ratio was determined for two samples of UO_2 -4 wt% PuO_2 fuel material remaining in rod FR-4 near the rupture by coulometric titration. Results of the analyses show that the O:U ratio of fuel remaining in the ruptured rod increased from nominally 2.01 to 2.1 during irradiation. A stoichiometry increase of this magnitude could have profound effect upon the operating conditions of the fuel by decreasing the effective thermal conductivity^(5, 6) and lowering the melting point.⁽⁷⁾

EXAMINATION OF ROD FR-10

A minimum examination was performed on nondefected replaceable rod FR-10 that was irradiated in FE-6504 during its second irradiation cycle in the FERTF (115 MWd/MT).

There were no changes in the diameter of the rod and no measurable change in the balance point, indicating that the operating conditions did not cause swelling or gross axial fuel movement. The rod was not sampled for fission gas or sectioned for metallographic examination.

FIGURE 28. Longitudinal Section Through the Depleted UO_2 Pellet at the Top of Ruptured Rod FR-4. A Thin Layer of Once-Molten Fuel Lines the Inner Surface of the Eroded Pellet (250 X)

FIGURE 29. Longitudinal Section in the Top Region of Ruptured Rod FR-4 (~32X)

EXAMINATION OF ROD FB-06

Rod FB-06, which was immediately adjacent to and had the same irradiation history as the ruptured rod in FE-6504, was examined to compare the irradiation performance of both rods. Three transverse sections from along the length of the rod and a longitudinal section from each end were examined.

There were no significant diameter changes in rod FB-06 as a result of irradiation, indicating that the operating conditions were not conducive to rod swelling. Mass spectrographic analysis of the 217 ml of gas collected from the rod indicated that 82% of the Xe+Kr formed during irradiation was released from the fuel.

Examination of the longitudinal section of the bottom end region of rod FB-06 shows that graded enrichment was effective in limiting fuel temperatures near the end cap (Figure 30). The depleted UO_2 pellet cracked due to thermal stresses. A thin layer (approximately 0.012 in. thick) of UO_2 - PuO_2 fuel that deposited between the depleted UO_2 pellet and the Zircaloy end cap during fabrication had no effect on the irradiation performance of the rod. Fuel melting occurred in the UO_2 -1 wt% PuO_2 fuel region to within 0.25 in. of the depleted UO_2 pellet at the bottom of the rod.

Fuel melting occurred to 57% of the radius in a transverse section taken at a plane 20 in. from the bottom of rod FB-06 (Figure 31). This section is characterized by a small central void, which is typical of structures formed in the lower portion of rods that operated with molten fuel.

A transverse section at a plane 30 in. down from the top of the rod, which is the same plane as the rupture in adjacent rod FR-4, indicates that fuel melting occurred to 50% of the radius (Figure 32). The central void in this specimen extends to 26% of the radius.

FIGURE 30. Longitudinal Section of the Bottom of Rod FB-06 (4 X)

D-4360
Photomicrograph

810
 β - γ Autoradiograph

809
 α Autoradiograph

FIGURE 31. Transverse Section 20 in. from the Bottom of Rod FB-06.
Melting Indicated to 57% of the Radius (4.8 X)

D-4359
Photomicrograph

808
 $\beta - \gamma$ Autoradiograph

807
 α Autoradiograph

FIGURE 32. Transverse Section 30 in. Down from the Top of Rod FB-06.
(This is the same plane as the defect in adjacent Rod FR-4.) Melting
Indicated to 50% of the Radius (4.8 X)

The transverse section of rod FB-06 taken at a plane 20 in. from the top end of the rod has a center void that extends to 32% of the radius (Figure 33). Melting occurred to the edge of the center void in this specimen since the inner surface of the void is lined with a thin layer of fuel which has the characteristic porous once-molten structure. Apparently, the molten fuel in the upper portion of the rod slumped downward. A postirradiation balance point was not obtained on this rod.

Examination of the top region of rod FB-06 showed that the 1 cm of UO_2 -1 wt% PuO_2 graded fuel material became mixed with the UO_2 -4 wt% PuO_2 fuel material during fabrication or subsequent handling (Figure 34). A large inverted teardrop center void is present in the top end region near the pellet. Measurements indicate that the fuel column moved the depleted UO_2 pellet upward 0.6 to 0.8 in. against the downward force of the spring. The depleted UO_2 pellet was affected by the adjacent UO_2 - PuO_2 fuel; however, molten fuel did not penetrate the pellet as it did in the defected rod. Because of interdiffusion, it is difficult to discern the original pellet-fuel interface. Sintering and homogenizing of the mechanically mixed UO_2 - PuO_2 fuel material occurred to the outside diameter of the fuel over a distance of 0.7 in. down from the pellet. Local overheating of the Zircaloy cladding occurred over a distance of approximately 0.2 in. Overheating of the cladding in the region of high fuel temperatures allowed partial collapse of the cladding in the overheated region. The maximum diameter decrease is estimated to be 0.02 in. or 4%. Metallographic examination of the cladding shows the presence of beta-phase zirconium, indicative of temperatures in excess of 1000 °C. At these temperatures, the inward plastic deformation of the cladding was probably caused by the coolant pressure of 1050 psi.

D-4358
Photomacrograph

806
 β - γ Autoradiograph

805
 α Autoradiograph

FIGURE 33. Transverse Section 20 in. Down from the Top of Rod FB-06.
The Center Void and Once-Molten Fuel Region Extend to 32% of the Radius
(4.8 X)

16847

Photomicrograph

804

 β - γ Autoradiograph

803

 α Autoradiograph

FIGURE 34. Longitudinal Section from the Top End of Rod FB-06. Holes are Microdrilled Chemical Analysis Samples (4X)

Radiochemical analysis of two microdrilled samples of fuel in the overheated region near the cladding distortion show that the fuel contains nominally 4 wt% PuO_2 . These analytical results confirm the interpretation of the autoradiographs that the UO_2 -1 wt% PuO_2 graded fuel material became mixed with the UO_2 -4 wt% PuO_2 fuel.

Axial flux profile monitored exposed during the irradiation of UO_2 -2 wt% PuO_2 FE-6505 just prior to the irradiation of defected FE-6504 provide an estimate of the power distribution expected in the end regions due to flux peaking. Although these data indicate an increase in the flux near the end, the increased power generation associated with the degree of peaking shown cannot account for the temperatures indicated by fuel structures, even if one assumes no fuel grading in the rod ends.

EXAMINATION OF ROD FB-83

After an examination of the nondefected rod (FB-06) adjacent to the ruptured rod revealed that higher than expected fuel and cladding temperatures existed in the top end region, the rod (FB-83) on the opposite side of the ruptured rod was examined.

There were no measurable changes in the outside diameter of the rod, and no postirradiation balance point was obtained on this rod. A total of 203 ml of gas was collected from the rod, equivalent to a calculated fission gas release of 79%. The total amount of gas collected and the percent Xe + Kr released agree well with previous quantities obtained from other rods from this element.

A small central void extends to 6% of the radius in the transverse section of rod FB-83 at a plane 20 in. from the bottom (Figure 35). Fuel melting extended to 54% of the radius at this axial position. The fuel structure in this cross section is typical of those formed in the lower portions of long fuel rods that have operated with central melting.

The transverse section of rod FB-83 at a plane 30 in. from the top of the rod, which is the same plane as the defect in rod FR-4, shows that the center void extends to 26% of the radius and fuel melting extends to 46% of the radius (Figure 36).

D-5583
Photomicrograph

871
 β - γ Autoradiograph

872
 α Autoradiograph

FIGURE 35. Transverse Section 20 in. from the Bottom of Rod FB-83.
Melting Indicated to 54% of the Radius (4.8 X)

FIGURE 36. Transverse Section 30 in. Down From the Top of Rod FB-83. This is at the Same Plane as the Defect in Adjacent Rod FR-4. Melting Indicated to 46% of the Radius (4.8X)

The transverse section of rod FB-83 at a plane 20 in. from the top has a central void that extends to 23% of the radius (Figure 37). Although no large amount of porous or once-molten fuel is apparent in this section, melting occurred to the edge of the center void since microstructural examination reveals a thin layer of porous fuel material on the inner surface of the void. Apparently the molten fuel from this region slumped downward.

Although autoradiographs indicate that the UO_2 -1 wt% PuO_2 graded enrichment in rod FB-83 became mixed with the UO_2 -4 wt% PuO_2 fuel material, as observed in other rods, there is no evidence of unexpectedly high fuel temperatures or localized overheating of the cladding as observed in the top end region of rod FB-06 (Figure 38).

There is no evidence of interaction between the depleted UO_2 pellet and the Inconel-X plenum spring. Even though limited pellet-powder fuel interdiffusion occurred during irradiation, pellet cracking is minimal. A somewhat smaller inverted teardrop central void than the one observed in rod FB-06 is present in the top region of this rod. Since the inside surface of the central void is lined with a thin layer of porous or once-molten fuel, the majority of the molten fuel apparently slumped downward during irradiation or at the time of shutdown. Radiochemical analyses of two microdrilled fuel samples taken at the same place as the overheated fuel and cladding region in rod FB-06 show that the fuel composition is nominally 4 wt% PuO_2 . These results confirm that the UO_2 -1 wt% PuO_2 graded enrichment became mixed with the UO_2 -4 wt% PuO_2 fuel, as indicated by the autoradiographs.

Measurements show that the position of the depleted UO_2 pellet in the top end of rod FB-83 did not change as a result of irradiation as it did in rod FB-06.

FIGURE 37. Transverse Section 20 in. from the Top of Rod FB-83.
Melting Indicated to 23% of the Radius (4.8 X)

16806
Photomicrograph

821
 β - γ Autoradiograph

822
 α Autoradiograph

FIGURE 38. Longitudinal Section of the Top End Region of Rod FB-83. β - γ and α autoradiographs show mixing of the UO_2 -1 wt% PuO_2 graded enrichment and the UO_2 -4 wt% PuO_2 fuel material. Holes are micro-drilled chemical analysis samples. (4X)

EXAMINATION OF ROD FB-59

Fuel rod FB-59, which showed external evidence of localized overheating in the top end region, was examined to better understand the overheating phenomenon.

For the first time, postirradiation measurements indicated a diameter increase in a nondefected HPD fuel rod. The diameter of rod FB-59 increased from nominally 0.565 in. at the ends to a maximum of 0.568 - 0.569 in. over an axial distance of approximately 20 in. in the lower portion of the rod. This represents a maximum diameter increase of 0.7%. Also, there are indications that the balance point shifted in the downward direction suggesting that fuel relocation occurred. The fuel rod swelling probably resulted from the volume expansion of molten fuel in the lower portion of the rod which had a higher bulk fuel density due to relocation of fuel in the downward direction.

A total of 206 ml of gas was collected from this rod which corresponds to a Xe+Kr release value of 38%. The high gas release value is probably the result of the higher volumetric average fuel temperature that existed in this fuel rod during irradiation.

The transverse section 20 in. from the bottom of the fuel rod shows essentially no central void with fuel melting to as much as 64% of the radius during irradiation (Figure 39).

Examination of the transverse section 30 in. down from the top of the fuel rod, which is at the plane of the defect in the ruptured rod, shows that the fuel was molten to as much as 64% of the radius during irradiation (Figure 40). There is essentially no center void at this axial position. This section is from a region in the rod that indicated a diameter increase of approximately 0.5%.

The transverse section at a plane 20 in. from the top of the rod has a central void that extends to 38% of the radius (Figure 41). Fuel melting extended to a maximum of 50% of the radius at this plane.

D-5584
Photomicrograph

863
 β - γ Autoradiograph

864
 α Autoradiograph

FIGURE 39. Transverse Section 20 in. from the Bottom of Rod FB-59.
Melting Indicated to 64% of the Radius (4.8 X)

FIGURE 40. Transverse Section 30 in. from the Top of Rod FB-59.
Melting Indicated to 64% of the Radius (4.8 X)

D-5425
Photomicrograph

834
 β - γ Autoradiograph

833
 α Autoradiograph

FIGURE 41. Transverse Section 20 in. from the Top of Rod FB-59.
Melting Indicated to 50% of the Radius (4.8 X)

Examination of the longitudinal section at the top of rod FB-59 shows that the UO_2 -1 wt% PuO_2 graded enrichment became mixed with the UO_2 -4 PuO_2 fuel material (Figure 42). Diffusion between the depleted UO_2 pellet and fuel has made the interface between the two indistinguishable on the photomicrograph. The interface is discernible, however, on the autoradiograph, and indicates that the central void and fuel melting occurred to within 0.05 in. of the pellet. Fuel melting, in an inverted teardrop region, occurred near the top of the fuel column.

Unexpectedly high fuel temperatures at the top of the column caused sintering and grain growth to the outer surface of the fuel over an axial distance of 0.58 in. below the depleted UO_2 pellet. Molten fuel flowed outwardly along radial cracks and possibly contacted the cladding without causing any noticeable core-cladding reaction. Localized overheating and collapse of the cladding occurred adjacent to the overheated fuel region in the same axial plane as observed in other rods from this element. The cladding collapsed a maximum of approximately 0.01 in. (2% of the diameter) over an axial distance of 0.32 in. Metallographic examination of the cladding in the deformed region shows the presence of beta-transformed zirconium grains which is indicative of temperatures above 1000°C .

Measurements indicate that the fuel column has moved the depleted UO_2 pellet upward 0.5 to 0.6 in. against the downward force of the plenum spring.

EXAMINATION OF ROD FB-55

Fuel rod FB-55, which showed the most severe external evidence of localized overheating in the top end region of any of the four overheated rods, was examined to more thoroughly understand this phenomenon (Figure 7).

Due to the manner in which rod FB-55 was removed from FE-6504, it was not possible to determine the postirradiation balance point. Measurements along the length of the rod, however, show that the diameter has not changed as a result of irradiation.

D-5424
Photomicrograph

837
 β - γ Autoradiograph

838
 α Autoradiograph

70

BNWL-272

FIGURE 42. Longitudinal Section at the Top End of Rod FB-59. Autoradiographs show mixing of the UO_2 -1 wt% PuO_2 graded enrichment and the UO_2 -4 wt% PuO_2 fuel material. Molten fuel flowed out radial cracks to possibly contact the cladding. (5 X)

A total of 211 ml of gas was collected from this rod which yielded a Xe+Kr release value of 88%. This value is the same as for rod FB-59, and indicates that the volumetric average fuel temperatures were nearly the same for the two rods. This was also verified by a comparison of the fuel structures formed in the two rods during irradiation.

The transverse section taken 20 in. from the bottom of the fuel rod shows essentially no central void and fuel melting to as much as 63% of the radius during irradiation (Figure 43).

Examination of the transverse section taken 30 in. from the top of the rod, which is in the same plane as the defect in the ruptured rod, shows that fuel melting occurred to as much as 62% of the radius during irradiation (Figure 44). There is a small central void at this axial position that extends to 17% of the radius.

The transverse section at a plane 20 in. from the top of the rod has a central void that extends to 37% of the radius (Figure 45). Fuel melting occurred to 46% of the radius at this plane.

Examination of the fuel in the longitudinal section from the top of rod FB-55 shows that the UO_2 -1 wt% PuO_2 graded enrichment has become mixed with the UO_2 -4 wt% PuO_2 fuel material as observed in other rods (Figure 46). The depleted UO_2 pellet is relatively unaffected although interdiffusion has occurred at the pellet-powder fuel interface. Fuel melting occurred to within 0.1 in. of the depleted UO_2 pellet. There is no evidence of interaction between the UO_2 pellet and Inconel-X spring. Although no central void has apparently formed in the top end region of the rod, fuel melting has occurred to within 0.1 in. of the pellet.

Unexpectedly high fuel temperatures in the top end region resulted in sintering and grain growth to the outer surface of the fuel. Molten fuel flowed outwardly along radial cracks to possibly contact the cladding without causing any noticeable core-clad reaction. Localized overheating

D-5794
Photomicrograph

921
 β - γ Autoradiograph

887
 α Autoradiograph

FIGURE 43. Transverse Section 20 in. from the Bottom of Rod FB-55.
Melting Indicated to 63% of the Radius (4.8 X)

D-5793
Photomicrograph

920
 β - γ Autoradiograph

886
 α Autoradiograph

FIGURE 44. Transverse Section 30 in. from the Top of Rod FB-55.
Melting Indicated to 62% of the Radius (4.8 X)

FIGURE 45. Transverse Section 20 in. from the Top of Rod FB-55.
Melting Indicated to 46% of the Radius (4.8 X)

16882

Photomicrograph

883

 β - γ Autoradiograph

884

 α Autoradiograph

FIGURE 46. Longitudinal Section of the Top End of Rod FB-55. Autoradiographs Show Mixing of the UO_2 -1 wt% PuO_2 Graded Enrichment and the UO_2 -4 wt% PuO_2 Material. The outward flow of molten fuel along radial cracks to possibly contact the cladding is also indicated. (4X)

and collapse of the cladding occurred adjacent to the overheated fuel region and in the same axial plane as was observed in other rods from this element. The cladding collapsed a maximum of approximately 0.01 in. (2% of the diameter) over an axial distance of 0.3 in. Metallographic examination of the cladding in the affected region shows the presence of beta-transformed grains indicative of cladding temperatures above 1000°C.

Measurements indicate that the fuel column has moved the depleted UO_2 pellet upward approximately 0.54 in. against the downward force of the plenum spring.

RELATED INFORMATION

Following the rupture of FE-6504 in the FERTF, experiments and evaluations to help elucidate the rupture mechanism were initiated. Other experiments related to the irradiation behavior of purposely defected, vibrationally compacted HPD fuel rods had been performed prior to the incident. These included exposure of irradiated fuel specimens in an out-of-reactor test loop under control conditions to investigate the oxidation behavior of the fuel; calculations to determine the amount of molten fuel that could be accommodated in a vibrationally compacted PRTR fuel rod without stressing the cladding; and irradiation of purposely defected vibrationally compacted UO_2 fuel rods operating under molten-core conditions in the ETR water-cooled loop before and after the PRTR incident. Also, a purposely defected UO_2 - PuO_2 fuel element was successfully irradiated under molten-core conditions in the FERTF prior to the incident. A discussion of these items is given below:

O:U Analytical Results

The stoichiometry of the fuel material contained in various rods of FE-6504 was determined by coulometric titration (Table III). O:U ratios were determined for two irradiated fuel samples from nondefected rods FB-83 and FR-6. The results show an O:U ratio of 2.017 for

fuel from rod FB-83 and $2.00\bar{3}$ for fuel from rod FR-6. These values agree well with preirradiated O:U ratios for this type of fuel material.

TABLE III. Stoichiometry of Fuel Material From FE-6504.

<u>Rods from FE-6504</u>	<u>Postirradiation O:U</u>
Ruptured Rod FR-4	$2.10\bar{1}$ $2.07\bar{0}$
Nondefected Rod FB-83	$2.01\bar{7}$
Nondefected Rod FR-6	$2.00\bar{3}$
Nondefected Rod FR-6 After Exposure in Out-of-reactor loop	$2.00\bar{7}$

Another irradiated fuel specimen from rod FR-6 was exposed in an out-of-reactor loop under conditions that approximated the FERTF coolant chemistry, temperature, and pressure during irradiation of defected FE-6504 prior to the rupture.⁽⁸⁾ The test conditions consisted of exposure to flowing lithiated water (pH 10.5) at 280 °C for approximately 60 hr and included two thermal cycles. The O:U ratio of the fuel material following this treatment was $2.00\bar{7}$. Loop exposure under these conditions apparently had little effect upon the stoichiometry of fuel material.

Analyses of two fuel samples from the vicinity of the rupture in rod FR-4, however, showed that the O:U ratio increased to approximately 2.1 during irradiation in the FERTF. The samples were complete cross sections of the rod, hence any radial variations in stoichiometry that may exist within the fuel would be averaged by the sampling technique. The analytical results, however, are indicative of the stoichiometry of the fuel in the cooler peripheral region of the ruptured rod since the central or once-molten fuel material is missing. Such an increase in stoichiometry could have a profound effect upon the operating conditions by decreasing the effective thermal conductivity^(5, 6) and lowering the melting point of the fuel.⁽⁷⁾

UO₂ Performance Limits

Analyses have been made of UO₂ fuel rod performance limits based upon cladding strains introduced by the volume expansion of the fuel.⁽⁹⁾ In these analyses, the surface heat flux limit was considered to be that value which causes an increase in UO₂ volume equal to the free space within the rod, and the only limiting factor considered was thermal expansion of the UO₂. Other factors which are heat flux or temperature dependent (such as fuel relocation, boiling burnout, or gas release), were assumed to be nonlimiting. The calculated limits were based on an instantaneous reactor startup condition, and did not consider any possible fuel relocation effects resulting from a programmed startup. The results indicate that the maximum permissible heat rating for a UO₂ fuel element is sharply defined by fuel diameter and fuel bulk density. The calculations indicate, for example, that an 85% TD fuel rod could operate safely with approximately 60% of the radius molten.

Calculations of the volume expansion of vibrationally compacted UO₂ fuel in Zircaloy-clad PRTR fuel rods also show that melt-radii up to approximately 60% of the fuel radius can be accommodated by the void volume within the rod without stressing the cladding.⁽¹⁰⁾ Figure 47 shows the calculated relationship between the incremental fuel volume and the amount of molten fuel. The calculations are based on the following assumptions:

- The melting temperature of UO₂-4 wt% of PuO₂ fuel is 2800 °C.
- The preirradiated bulk fuel density is 85% TD.
- The fuel transverse section has three distinct regions consisting of:
 - i. Molten fuel
 - i'. Solid fuel above 1800 °C
 - i''. Fuel below 1800 °C

FIGURE 47. Calculated Relationship Between the Incremental Fuel Volume and the Amount of Molten Fuel in a Vibrationally Compacted $\text{UO}_2\text{-PuO}_2$ PRTR Fuel Rod.

- The $\Delta V/V_0$ (11, 12) (based upon the volume weighted average temperature in each region) for regions
 - i. = 0.25
 - i'. = 0.09
 - i''. = 0.03.
- All fuel above 1800 °C is assumed to be 100% TD or 97% TD.
- The temperature profiles within the fuel were determined from calculated flux depression data for UO_2 -4 wt% PuO_2 fuel rods.
- The thermal conductivity, K, for vibrationally compacted UO_2 -4 wt% PuO_2 fuel is as follows:

$$K = \frac{45.1}{135 + T} + C T^3 \quad (13)$$

where

K = thermal conductivity for 100% dense
 UO_2 , watts/cm- °C
 T = temperature (°K).

Density correction for K:

$$K' = K (1 - \beta \varphi)$$

where K = thermal conductivity for UO_2 with porosity
 equal to φ

$$\beta = 2.5 \pm 1.5$$

φ = porosity of sample = (1-density).

ETR Defect Tests

Two intentional defect tests have been performed in the ETR. The first test (GEH-12-38) conducted in the water-cooled P-7 loop in the ETR was designed to investigate the irradiation behavior of an intentionally defected vibrationally compacted UO_2 fuel rod operating under molten-core conditions. The test element was a Zircaloy-clad 7-rod cluster, six rods of which contained UO_2 -1 wt% PuO_2 fuel and the defected rod contained UO_2 enriched to 1.8% U^{235} and vibrationally compacted to 86% TD.

The PRTR-size rods were 0.565 in. OD and 30 in. long. A longitudinal split 1.0 in. long and 0.020 in. wide penetrated the cladding at the maximum flux position. The fuel element, which had two circumferential strip bands to hold it together as an integral unit, was contained in a stainless steel basket during irradiation in the loop. A coolant temperature of 420 °F at 1600 psi was maintained in the loop during the 4-week irradiation period.

During reactor startup, the delayed neutron activity in the loop increased nearly linearly with reactor power. The loop gamma activity peaked during power level changes and decreased to a steady-state value during continued operation at constant power level. The gamma activity in the P-7 loop piping cubicle was 7 R/hr during full power steady-state conditions. During reactor scrams and quick recoveries, bursts of gamma activity in the cubicle reached 42, 35, 25, and 18 R/hr on successive recoveries. After each recovery, the activity returned to the normal level within one hour of steady-state operation.

An unexpected decrease in the neutron flux in the loop position produced a lower than anticipated power generation in the defected rod. Postirradiation examination revealed maximum fuel temperatures near melting at the plane of the defect (estimated maximum power generation, ~ 19 kW/ft). There was no evidence of rod swelling in the region of the defect, fuel washout, waterlogging, or increased hydride formation as a result of irradiation.

The second test (GEH-12-39) irradiated in the P-7 loop of ETR was designed to simulate the irradiation conditions during the testing of FE-6504 in PRTR. The test element was a Zircaloy-clad, 7-rod cluster; six rods contained UO_2 -3 wt% PuO_2 fuel, and the defected rod contained UO_2 enriched to 5.0% U^{235} . These PRTR-sized rods, which had a 3.5 in. long fission gas plenum in the top end region, were 0.565 in. OD and 33.8 in. long. A 0.06 in. diameter hole penetrated the

cladding at the maximum flux position 20.7 in. below the top of the fuel column. The fuel rods were held together by three circumferential bands. The element was contained in a stainless steel basket for irradiation in the loop. The coolant temperature was 480 °F at 1600 psi during the 35-day irradiation period, and the test element was subjected to 13 reactor scrams from full power.

The steady-state gamma activity in the loop piping cubicle was 12 R/hr during the first portion of the irradiation period and decreased to 10 R/hr during the latter portion. During reactor power recoveries after scrams, bursts of gamma activity in the piping cubicle were greater than 100 R/hr. The activity returned to the steady-state level after each reactor power recovery.

Postirradiation examination of the defected rod revealed a loss of approximately 2 g of fuel, no dimensional changes in the rod, and no rod or defect distortion. Examination of a transverse cross section at the plane of the defect indicated a maximum fuel temperature above 2800 °C for the estimated maximum linear rod power of 20 -21 kW/ft.

FERTF Molten-Core Defect Test Prior to the Incident(3, 15)

A preirradiated vibrationally compacted UO_2 -2 wt% PuO_2 HPD fuel element (FE-6004), intentionally defected with a 1/16 in. diameter hole through the cladding, was successfully irradiated under molten-core conditions in the FERTF prior to the incident. This HPD element was different from the one involved in the incident in that the rods did not have a fission gas plenum in the top end region. Also, the experimental conditions during this test differed from the test conditions at the time of the incident in that the maximum rod power at the plane of the defect was lower (24 to 25 kW/ft) and the coolant temperature was not cycled. The defected element did, however, endure several shutdowns and power cycles.

The element operated successfully with melting at the plane of the defect for approximately 60 hours when the test was terminated. The defected test had been in the FERTF for a total of 287 operating hours. Prior irradiation of this element at a maximum linear rod power of approximately 19 kW/ft to over 3500 MWd/MT_{fuel} was undoubtedly sufficient to cause considerable in-reactor sintering of the fuel. Burst type activity releases, characteristic of leaker defects, occurred during incremental power increases at the lower power levels. Incremental power increases at the higher levels produced less severe activity releases, but in all cases during continued operation, the loop activity approached a steady-state value in an exponential manner. The steady-state gamma activity in the loop during irradiation of this element was approximately 15 times greater than for any defected elements previously tested. The rod power, however, was approximately 3 times greater than for any previous experiment. Activity bursts occurred when the reactor was shut down and also when the loop was depressurized.

There was essentially no change in the physical appearance of the defect, no dimensional changes in the rod, no significant fuel washout, and no metallographic evidence of increased hydride concentration in the cladding after irradiation in the FERTF. An observable effect of irradiation was a grey discoloration of the cladding downstream from the defect due to the evolution of material and/or gases from the defect. The alpha and beta autoradiographs indicate that the symmetry of the fuel structure and fission product distribution patterns were affected by the presence of the defect. The concentration distribution patterns of the alpha and beta-gamma emitters, as indicated on the autoradiographs, do not coincide except in the once-molten fuel region.

Examination of the fuel structure formed during irradiation at the plane of the defect indicates a greater degree of melting than anticipated from the maximum rod power generations calculated. Although the appearance of the fuel structure in an adjacent nondefected rod differed in some respects from the fuel structure in the defected rod at the same plane, approximately the same amount of fuel melting was indicated. There was less porosity associated with the once-molten fuel region in the nondefected rod and the fission product distribution patterns (particularly as indicated on the alpha autoradiograph) were different for the defected and nondefected rods. Sixty-four percent of the fission gases were released from the fuel under these operating conditions.

PRESSURE TUBE EVALUATION

GENERAL

The FERTF Zircaloy pressure tube was discharged and destructively examined in sufficient detail to evaluate the relationships between the fuel failure and the pressure tube failure (Figures 48 and 49). This tube, No. 6095, was a standard Zircaloy-2 pressure tube, 3.250 in. ID by 0.150 in. wall. The aluminum shroud or calandria tube surrounding the FERTF pressure tube is 6 in. ID, whereas all the other PRTR shroud tubes are 4 in. ID. A helium-filled gas annulus exists between the two tubes. There was no evidence of the pressure tube having been defective; therefore, only that portion of the tube damaged by the rupture was examined in detail. As a result of this examination, the relative position of the rupture in the fuel rod with respect to the blowout in the pressure tube was accurately determined. Examination of the inside surface of the Zircaloy pressure tube in the region of the rupture shows that fuel rod cladding and hot fuel contacted the tube in a region approximately 1-1/14 in. below the blowout.

There was evidence of local overheating of the pressure tube at the rupture. Damage to the surrounding aluminum shroud tube was negligible. There was a small pitted area on the shroud tube which had a sandblasted appearance opposite the hole in the pressure tube. The pitted area on the shroud tube was caused by impingement of fuel from the rod at the time of the rupture.

LOCATION OF RUPTURE

By superimposing photographs of the inside of the pressure tube and the ruptured fuel rod FR-4, it is possible to position the location of the rupture in the fuel rod relative to the location of the rupture in the pressure tube. The blowout in the pressure tube was a nearly circular penetration approximately 1/2 in. diameter (Figures 48 and 49). Figure 49 is a photograph of the inside of the pressure tube after it had been

16752

FIGURE 48. Outside View of Zircaloy Pressure Tube Rupture (2X)

16796

FIGURE 49. Inside Surface of the Ruptured Zircaloy Pressure Tube. Light Colored Area may be the Result of Fuel Impingement at the Time of the Pressure Tube Failure or Possibly from a Blast of ZrO_2 Particles and Gas from a Localized Zirconium-Water Reaction. ($\sim 0.8X$)

split longitudinally down the center. Figure 50 is a transparent reproduction of an alpha autoradiograph of the inside surface of the tube slightly upstream of the pressure tube penetration.

By superpositioning the alpha autoradiograph reproduction on the photographs of the pressure tube, the location of fuel particles embedded on the inside surface of the Zircaloy pressure tube becomes obvious. In addition, by superimposing the transparent photograph of the fuel element failure (Figure 51) on the reproductions of the alpha autoradiograph (Figure 50) and the inside surface of the pressure tube (Figure 52), the relative positions of each at the time of the rupture becomes clear. One of the photographs, in this case the pressure tube (Figure 52), had to be printed as a mirror image to make this superposition possible. The superpositioning of photographs shows that the fuel cladding and fuel material contacted the Zircaloy pressure tube in the area of alpha activity without causing failure of the tube. The overheating and failure occurred slightly downstream from the point of fuel contact. The cause or identity of the light colored area radiating from the location of fuel contact with the pressure tube has not been positively determined. This area may be the result of the impingement of fuel from the rod at the time of the pressure tube failure or possibly from a blast of gas and ZrO_2 particles resulting from a localized zirconium-water reaction.

TEMPERATURE OF PRESSURE TUBE

Excessive temperatures in the rupture region of the Zircaloy pressure tube are evidenced by a thick ZrO_2 layer (Figure 48) and beta-transformed Zircaloy-2 (Figure 53). A metallographic section of the ruptured area sectioned as indicated in Figure 48 shows the normal cold worked grain structure slightly removed from the blowout (Figure 53). Retention of the cold worked grain structure indicates that temperatures did not exceed 600°C in this region. The Zircaloy-2 immediately surrounding the hole consists of a beta-transformed structure with evidence of grain growth in the beta phase indicative of temperatures in excess of 1000°C .

FIGURE 50. Transparent Representation of Alpha Autoradiograph
of the Inside of the Pressure Tube (~2.1 X)

BNWL-272

FIGURE 51. Transparent Representation of Ruptured Area in Fuel Rod
(~2.1X)

FIGURE 52. Inside Surface of Hot Spot Blowout in Zircaloy Pressure Tube 6095. (~2.1X)

0660332

FIGURE 53. Section A-A of Figure 48. Section Through Blowout Crater in Zircaloy Pressure Tube Shows Transition from Transformed Beta Grain Structure on Left (>1000 °C) to Cold-Worked Grain Structure on Right (<550 °C).

The transition from cold worked structure to recrystallized structure extends as far from the rupture on the outside of the tube as it does on the inside. This is not consistent with point source heating from the inside of the tube. The isotherms indicated are more representative of a heat source located in the rupture such as would be caused by hot fuel passing through the hole after the rupture.

TIME FOR PRESSURE TUBE RUPTURE

It is difficult to estimate the time required for the pressure tube to rupture at its operating stress of 11,000 psi when a localized spot on the tube is heated to 1000 °C or higher. D. H. Stone⁽¹⁶⁾ has reported 700 °C stress rupture data for Zircaloy-2 tubing. Larson-Miller and other methods of extrapolating the 700 °C data to 1000 °C are not completely valid because of the phase change in Zircaloy at 900 °C to 1000 °C. The data indicate, however, that failure of the pressure tube would be almost instantaneous under these conditions, and that the tube had to be heated suddenly to achieve the beta-phase temperature before it failed.

SHROUD TUBE DAMAGE

Extent of damage to the aluminum shroud tube caused by the pressure tube blowout was evaluated by visual examination in situ with a borescope and by replicating the inside surface with a rubber casting (Figure 54). The replica reveals an affected area approximately 2 in. in diameter that has a sandblasted appearance apparently caused by the impingement of steam and fuel particles emerging from the fuel rod when the pressure tube failed. The azimuth of the affected area of the shroud corresponds to the position of the tube rupture and its axial position indicates that the jet was directed slightly downward. The damage to the shroud tube is so slight that it would have an insignificant effect on the mechanical properties.

4652298

FIGURE 54. Cast Rubber Replica of the Inside Surface of the Aluminum Shroud Tube (3 X)

Calculations based on published ⁽¹⁷⁾ and unpublished Chalk River data were made to determine the effect of the pressure tube rupture on the surrounding shroud tube. When using ultraconservative assumptions, the calculations predict that the shroud tube would burst, but with more reasonable assumptions, the calculations predict that the shroud would remain undamaged. The fuel and fission product contaminated coolant entered the heavy water moderator system via penetrations through the top of the calandria which was not damaged as a result of the incident.

CAUSE OF PRESSURE TUBE FAILURE

The evidence indicates that blowout or rupture of the pressure tube was caused by localized overheating. Extremely high ductility in the Zircaloy-2 pressure tube is indicated by the reduction in wall thickness at the point of failure. The high ductility of Zircaloy-2 indicated is in contrast with the low ductility behavior of stainless steel under similar circumstances. A failure in the form of a hot spot blowout of a stainless steel pressure tube in the CG 6x9 position of the ETR exhibited low ductility separation of heat sensitized grain boundaries. ⁽¹⁸⁾ Both the stainless steel and the Zircaloy-2 pressure tubes failed safely, however, in that the failure did not propagate.

THERMAL HYDRAULIC EVALUATION

THERMAL HYDRAULIC CONDITIONS AT THE TIME OF THE INCIDENT

Coolant Temperature

The measured coolant inlet and outlet temperatures in the FERTF at the time of the failure were 408 °F (209 °C) and 489 °F (254 °C) and the corresponding subcoolings (degrees below saturation temperature) were 142 °F (79 °C) and 61 °F (34 °C), respectively. The average coolant temperature at the point of failure was calculated to have been about 455 °F (235 °C) with a corresponding subcooling of 95 °F (53 °C).

Flow Rate

The flow rate in the FERTF was set and controlled at a 91% reading on the flow-controlling venturi. The flow rate varies with temperature, but at the time of the failure, the flow rate was 170 gpm, (67,800 lb/hr, or 2,760,000 lb/hr-ft²).

Tube Power and Heat Flux

The FERTF tube power at the time of the failure was 1765 kW. Due to neutron flux gradients and hence, power generation gradients, within the element and fuel rod, the heat fluxes had wide spatial variations. The average heat flux around the failed rod at its axial peak heat generation point (slightly upstream of its vertical center) is calculated to have been 633,000 Btu/hr-ft². However, because of the flux gradient existing across the rod, the peak circumferential heat flux on the outer surface of the rod farthest from the center of the fuel element is calculated to have been 737,000 Btu/hr-ft². The axial position of the intentional defect and subsequent rupture was sufficiently close to the point of axial peak flux that the calculated heat fluxes were essentially the same.

BOILING BURNOUT HEAT FLUXES

Boiling burnout is a severe breakdown of the heat transfer mechanism and generally occurs at high heat fluxes or low flow rates where the nucleate type of boiling can no longer be supported. During boiling burnout, the temperature of the heat transfer surface increases and sometimes melting occurs. There is a large body of laboratory boiling burnout data available for rod bundle fuel elements with geometries and operating conditions similar to FE-6504.^(19, 20, 21) If these data are extrapolated to the conditions of FE-6504 in a normal position and without surface bulges, a boiling burnout heat flux of about 1,500,000 Btu/hr-ft² is predicted. This is greater than twice the peak surface heat flux of the fuel rod at the point of failure.

TEMPERATURE CALCULATIONS FOR THE CASE OF CONTACT BETWEEN THE FUEL ROD AND THE PRESSURE TUBE

Calculations have been made to determine the fuel rod and pressure tube temperatures which would have existed in the event of a single fuel rod of FE-6504 bowing outward and contacting the Zircaloy pressure tube. Results of these calculations are shown in Figures 55 and 56. The calculations do not indicate with certainty that the fuel cladding would have melted; however, the calculated temperatures would have been sufficiently high to cause failure of the pressure tube. The calculations do not account for the high temperatures ($>1000^{\circ}\text{C}$) indicated by structure changes in the rupture region of the pressure tube. The possibility of a rod touching the pressure tube by bowing cannot be ruled out even though there is no supporting evidence. Spiral wire wraps on the fuel rods keep the failed rod a minimum of 0.070 in. from the pressure tube. The failed rod was not wire wrapped; however, it was bound with circumferential bands to the wire-wrapped rods at about 14 in. intervals. The failed rod would have had to bow at least 0.070 in. in the 14 in. span to touch the pressure tube. This could possibly happen; however, there is no evidence of permanent deformation in either the failed rod or any of the other outer 11 rods in the element

FIGURE 55. Calculated Temperature Map of a UO -4 wt% PuO₂ PRTR Fuel Rod Operating at 24 kW/ft in Contact with the Zircaloy Pressure Tube

FIGURE 56. Calculated Isotherms for a UO_2 -4 wt% PuO_2 PRTR Fuel Rod Operating at 24 kW/ft in Contact with the Zircaloy Pressure Tube

which might be expected to occur if a bowing condition did take place. If bowing occurred between bands, the failure would be expected to be nearly midway between bands unless a band were loose. No loose bands were found, and the failure was not centered between bands.

PRESSURE DROP INFORMATION

At the time of the incident, a temperature cycle of the FERTF coolant had just been completed and a high-speed recording instrument had been turned on in preparation to taking data once steady-state conditions were established. Included on the recording was a pressure drop trace (Figure 5). This trace showed an increase in the pressure drop of 3 or 4% from about 8.2 psid to 8.5 psid approximately 24 sec before the reactor scram. At about the time of the scram, a further increase occurred. The latter increase was sufficient to cause the ΔP recording to go off scale, which would be at least another 0.4 psid.

Calculations were made to determine the size of steam bubble which would be required to cause the approximately 3% increase in pressure drop. The calculations show that a steam bubble about 4 in. long and occupying 10% of the cross section of the flow area would increase the pressure drop about 0.3 psi or about 3.7% of the normal 8.2 psi pressure drop. The flow area bounded by the failed rod in the element and the pressure tube wall is about 10% of the flow area. In other words, a steam bubble extending the length of the rupture in the failed rod and occupying that fraction of the flow area associated with the failed rod could be expected to cause an increase in the pressure drop across the test section similar to the increase that occurred approximately 24 sec prior to the pressure tube failure.

HOT SPOTS AT THE ENDS OF THE RODS

Calculations were made to determine the length of time it might take for the hot spots to develop at the downstream ends of four fuel rods. The calculations assumed that heat transfer from the jacket to coolant was suddenly interrupted by boiling burnout. The calculations then added

the weight-heat capacity product of the Zircaloy jacket and the UO_2 fuel. This, when divided into the volumetric heat generation rate, gave a temperature increase rate of $334^\circ\text{F}/\text{sec}$. The Zircaloy cladding temperature would have had to increase from 750°F to approximately 1600°F to cause the metallographic changes observed. Hence, the cladding temperatures observed would have been attained in approximately 2.5 sec.

This simplified calculational model is not quite correct. First, boiling burnout does not completely stop heat transfer, but rather, decreases it significantly. If the small but finite heat transfer after boiling burnout were considered, somewhat longer times would be calculated. Second, the model does not account for the temperature variations around the circumference of the rod. If these were accounted for, somewhat shorter times would be calculated. These two errors tend to offset each other; however, it is believed that the second effect is more dominant in the early stages of the temperature rise. Therefore, the times calculated by this simplified model are probably reasonably accurate, but perhaps slightly too long. This analysis indicates that the hot spots that occurred in the ends of some rods in the element and the associated cladding temperatures ($>1000^\circ\text{C}$) as indicated by metallographic examination could be expected to occur about 2 sec after disruption of heat transfer.

If it is assumed that the increase in the FERTF pressure drop that occurred at about the time of the reactor scram was due to the formation of a large volume of steam which extended to the end of the fuel rod to cause boiling burnout, the Zircaloy cladding could reach a temperature of 1600°F (870°C) during a 2 sec period.

CALCULATED TEMPERATURES IN A ZIRCALOY PRESSURE TUBE CONTACTED WITH MOLTEN UO_2

Transient heat transfer calculations were performed to determine the temperatures resulting in a Zircaloy pressure tube contacted with molten UO_2 . The model chosen for the calculations was a 1/2 in. diam cylinder of molten UO_2 at 5500°F (3040°C) resting against the pressure

tube. Heat was assumed to flow from the circumference of the UO_2 cylinder to the coolant at a rate of $250 \text{ Btu/hr-ft}^2\text{-}^\circ\text{F}$. The heat transfer contact between the UO_2 and the pressure tube was assumed to be perfect. Heat was assumed to flow from the UO_2 into the tube wall and then within the tube wall to the coolant. The calculations show that the tube wall directly adjacent to the UO_2 would reach an average temperature of about 1850°F (1010°C) about 3 sec after contact. The calculations also show that the average temperature in the pressure tube would then level off to between 1800°F and 1850°F . At these temperatures, a Zircaloy pressure tube would be expected to fail almost instantaneously. The calculations also indicate a very steep temperature gradient in the Zircaloy pressure tube and that 3/16 in. away from the region of UO_2 contact, the temperatures are within a few tens of degrees of the coolant temperature.

CALCULATED TEMPERATURES OF THE PRESSURE TUBE DUE TO RADIANT HEATING FROM AN OVERHEATED FUEL ROD

Heat transfer calculations were performed to determine what temperatures could be expected in the pressure tube due to radiant heat transfer from an overheated fuel rod. It was assumed that the heat transferred to the tube from a 1/2 in. strip of a fuel rod would be conducted within the wall of the pressure tube and then to the coolant. The maximum temperature of the pressure tube would depend upon the temperature of the overheated fuel rod, the emissivities of the fuel rod and pressure tube surfaces, and the heat transfer coefficient from the pressure tube to the coolant in the immediate vicinity. None of these variables are well known and thus, the pressure tube temperatures cannot be calculated accurately.

The results of the calculations indicate that very high fuel surface temperatures and low heat transfer coefficients from the pressure tube to the coolant would be necessary to have caused the magnitude of tube temperatures indicated by metallographic examination ($> 1000^\circ\text{C}$). While it is thought that such values for the fuel surface temperature and the pressure tube to coolant heat transfer coefficient were not likely, the calculations do show that tube wall temperatures high enough to cause deformation and failure ($\sim 600^\circ\text{C}$) could have been possible.

DISCUSSION

OBSERVATIONS WHICH COULD BE PERTINENT TO THE CAUSE OF THE INCIDENT

Many observations and pieces of evidence concerning the incident have been accumulated during the course of the incident investigation. In addition, many calculations have been made bearing on various aspects of the incident. Some of these are embodied in this report, others are reported elsewhere, and still others have not as yet been reported. Those observations which could possibly have a bearing on the cause of the incident are summarized below:

Fuel Element

- a. There is no evidence that any of the rods in the fuel element were bowed out toward the pressure tube.
- b. Corrosion spots were observed on the surfaces of the rods at the location of a circumferential band at the top end of the rupture region. These corrosion spots were caused by a band which was not on the element at the time of the rupture.
- c. A circumferential band was located around the fuel element about 3.5 in. below the plane of the defect.
- d. All five circumferential bands around the fuel element were intact and apparently in their original positions after the rupture.
- e. The defect hole was drilled 30-3/16 in. below the square shoulder on the top end cap. The hole was located within the area of what is now the upper portion of the two pieces of missing cladding in the rupture area.
- f. There is localized fuel rod swelling associated with the lower portion of the cladding rupture.
- g. External evidence of hot spots or boiling burnout was observed on four rods 8-1/4 in. down from the square shoulder on the top end cap. This would be located at the fuel-pellet interface in the

preirradiated condition. The three worse affected rods are located 120 degrees apart in the element and next to support gussets in the end bracket. A band could have been located over these localized hot spots during irradiation.

Ruptured Rod FR-4

a. Fuel

- (1) An unusually large unobstructed central void extended essentially the full length of the ruptured rod.
- (2) Over 700 g of fuel were missing from the rod.
- (3) An unexpectedly large amount of fuel melting occurred in the ruptured rod during irradiation. Up to 75% of the radius was molten in the vicinity of the defect at the time of the rupture. This is more than was found in the examination of any other rod from the element. Calculations show that the volume expansion resulting from more than a 60% molten radius could be expected to stress the cladding.⁽¹⁰⁾
- (4) Analyses show that the O:U ratio of the UO_2 -4 wt% PuO_2 fuel in the ruptured rod increased from 2.01 to 2.10. There is laboratory evidence that this would decrease the effective thermal conductivity^(5, 6) and the melting temperature.⁽⁷⁾
- (5) Fuel is missing to the edge of porosity (defined as the melt boundary) in the top portion of the rod.
- (6) Some porous material (thought to be once-molten fuel) remained in lower portion of the ruptured rod.
- (7) Fuel is missing completely for approximately 5 in. in the vicinity of the rupture.
- (8) Graded enrichment in the bottom end of the rod remained intact and prevented excessive fuel temperatures adjacent to the end cap.

- (9) The UO_2 -1 wt% PuO_2 graded fuel material and the UO_2 -4 wt% PuO_2 fuel became mixed in the top end region of the rod.
- (10) Molten fuel eroded through the top depleted UO_2 pellet.
- (11) Molten fuel extended 4.2 in. (60%) into the gas plenum.
- (12) There was a reaction between the molten fuel and the Inconel-X plenum support spring.
- (13) The inner portion of the fuel adjacent to the central void in the plenum region has lost the characteristic once-molten appearance.
- (14) The top depleted UO_2 pellet moved up approximately 0.15 in.

b. Cladding

- (1) Some areas of the break have a burned or melted appearance and a heat-discolored band surrounds the upper or downstream portion of the cladding rupture.
- (2) The lower or upstream portion of the rupture has the appearance of a longitudinal split. Considerable deformation is associated with the split region which does not appear to have been overheated to the same degree as the upper or downstream region of the rupture.
- (3) There is evidence of beta-transformed and once-molten Zircaloy cladding in the rupture region.
- (4) There is no metallographic evidence of massive hydride concentrations in the cladding. Some areas show a general increase in hydride concentration.
- (5) A thick ZrO_2 layer associated with an oxygen-rich diffusion layer has formed on the inner and outer surfaces of the cladding near the fracture surfaces.

- (6) Once-molten fuel has adhered to the inner and outer surfaces of the cladding over the ZrO_2 layer near the fracture surfaces in some areas of the rupture.
- (7) Fuel and cladding have alloyed together at the fracture surfaces in some areas of the rupture.
- (8) A zirconium-oxide layer has formed over the fuel-clad alloy in some areas of the ruptured cladding.
- (9) It has been estimated that the ZrO_2 layers on the cladding in the vicinity of the rupture could have formed in 20 sec or less at the temperatures indicated. (22, 23)

Nondefected Rods

a. Fuel

- (1) The depleted UO_2 pellet is intact and the UO_2 -1 wt% PuO_2 graded enrichment region is separated from the UO_2 -4 wt% PuO_2 fuel material as designed in the bottom end region of the rods.
- (2) The degree of fuel melting varies from rod to rod. For instance, the indicated degree of melting ranges from 46 to 65% of the radius at a plane 30 in. down from the square shoulder on the top end cap which is in the plane of the defect.
- (3) Structures typical of those formed in oxide fuels were formed in the nondefected rods. In general, the size of the central void is larger in the upper portions of the rods than in the lower portions, a possible indication of axial fuel relocation.
- (4) Mixing of the UO_2 -1 wt% PuO_2 graded enrichment and the UO_2 -4 wt% PuO_2 fuel occurred in the top end regions of all rods examined.
- (5) Fuel melting occurred in the top ends of some rods and not in others.

- (6) Central voids formed up to the pellet-powder fuel interface.
- (7) There is evidence of molten fuel flowing out through radial cracks to possibly contact the cladding in the top end region of some rods that experienced localized overheating.
- (8) The top depleted UO_2 pellet has moved upward from 0.5 to 0.8 in. in those rods which displayed localized overheating in the top end regions.

Hot Spots

- (1) External evidence of localized overheating was present on the outside surface of four rods approximately 8-1/4 in. down from the square shoulder on the top end cap. Three of the affected rods are located 120 degrees apart in the element and positioned adjacent to the support gussets in the top end bracket.
- (2) Transformed beta-phase structures are present in the overheated areas of the cladding indicating temperatures in excess of 1000°C .
- (3) There is no evidence of corrosion or increased hydride concentrations associated with the overheated regions.
- (4) There is no evidence of fuel-clad reaction, even in the overheated regions.
- (5) Sintering and grain growth has occurred to the outside edge of the fuel in the overheated regions.
- (6) It is possible that a circumferential strip band was located in the region of the hot spots during irradiation.
- (7) Chemical analysis of the fuel in the overheated region of one rod (FB-06) indicated an average plutonium concentration of approximately 4.3 wt% PuO_2 for two samples.

This is compared to an average of 3.9 wt% PuO_2 for two samples from the same region of a nonaffected rod (FB-83). (The accuracy of the chemical analysis is $\pm 10\%$. Also, the analysis of the small fuel areas sampled by the microdrilling technique may be affected by localized variations and produce misleading results.)

c. Fission Gas Release

- (1) The fission gas release data for the rods sampled is remarkably coherent. The values range from 75 to over 88% Xe+Kr release.
- (2) The amount of gas released is related to the volumetric average fuel temperature which is reflected by the degree of fuel melting. The largest amount of fission gas was released from the rods that displayed the greatest degree melting (FB-59 and FB-55).

Pressure Tube

- a. The fuel clad was hot longer than or was hotter than the pressure tube.
- b. A thick oxide formed on the inside of the pressure tube before it ruptured.
- c. The hole in the pressure tube was opposite the large area of missing cladding in the rod.
- d. The pressure tube had to be heated suddenly to achieve beta-phase temperatures before it failed.
- e. There is some evidence of a localized cladding-water reaction where the cladding first contacted the pressure tube.

Thermal Hydraulics

- a. ΔP of the test section increased by $\sim 4\%$ (0.3 psi) about 24 sec before the neutron flux and flow decreased. It is calculated that a steam bubble 4 in. long and occupying 10% of the flow area could cause a pressure drop of about this magnitude.
- b. The reactor was scrammed by a FERTF Test Section - Low Inlet Pressure trip.
- c. At about the time of the scram, the ΔP of the test section increased off scale ($\sim 5\%$), decreased off scale, and then increased off scale, where it remained.
- d. The maximum heat flux at the point of rupture was 737,000 Btu/-hr-ft² (average circumferential heat flux of 633,000 Btu/-hr-ft²). The boiling burnout heat flux at this location is estimated to be 1,500,000 Btu/hr-ft², provided the fuel element was centered in the pressure tube.
- e. The saturation pressure in the FERTF at the time of the rupture was 650 psia corresponding to an outlet water temperature of 495 °F.
- f. The pressure in the FERTF was 1100 psia prior to the time of the rupture, and about 650 psia immediately after the scram.
- g. Calculations of the temperatures in the FERTF Zircaloy pressure tube for the case of a fuel rod bowing out to touch the pressure tube show a maximum temperature of 1090 °F. This is considerably less than temperatures estimated by metallurgical examination (> 1650 °F).
- h. Calculation of temperatures for the case of a cylinder of molten UO₂ (1/2 in. in diameter) protruding from a fuel rod

and touching the pressure tube indicate the pressure tube would reach 1850° F (1010° C) about 3 sec after contact.

- i. Calculations indicate that heat transfer by radiation probably would not have caused pressure tube temperatures of > 1000° C found by metallographic examination but might have been sufficient to cause pressure tube temperatures of 600 °C.

General Observations

- a. No RAM alarm was received before scram. The RAM unit measures the loop coolant activity 3 to 5 sec after the coolant has passed the fuel element and sounds an alarm as a result of excessive activity indications.
- b. Evaluation of gross gamma (GM) and delayed neutron (DN) loop coolant activity monitors indicates that before the rupture, there was no increased FERTF activity and the levels were no higher than have been experienced in previous FERTF tests with defected elements.
- c. Following the rupture, the amount of noble gas measured in the PRTR containment vessel was equal to 40% of that calculated to have been contained in the ruptured rod.
- d. Measurements in the Plutonium Recycle Critical Facility (PRCF) show that the effect of coolant voiding in the FERTF was to decrease reactivity.
- e. PRCF measurements show that the effect of bowing one rod out 1/16 in. (to within one wire wrap thickness of the pressure tube wall) was to increase reactivity. The local flux increase was measured in the PRCF to be 9%. Shifting a whole fuel element over until it touches the pressure tube wall increases the local flux by the same amount.

- f. Loss of all fission gases in the defected rod would increase rod power a maximum of 8%. (24)
- g. Approximately 500 g of the 700 g of fuel missing from the ruptured rod were estimated to have remained in the FERTF system.

PROPOSED RUPTURE MECHANISM

The accumulation of observations and results of calculations were examined and a mechanism of failure was outlined. The proposed mechanism may not be entirely correct in some of its details, but it is consistent with the evidence, and it is believed to closely describe the sequence of events leading to the failure. The step-by-step mechanism is given as follows:

1. Preirradiated fuel rod FR-4 was defected by drilling a hole through the cladding.
 - a. Some basin water probably entered the rod.
 - b. Loop coolant water was forced into the rod during cold pressurization of the FERTF. The gas plenum may have permitted more than the normal amount of water to enter the rod.
2. The UO_2 - PuO_2 fuel reacted with the water inside the fuel rod to oxidize the fuel during irradiation at operating temperatures. (Very recent data indicate that in the presence of water vapor, UO_{2+x} might exist at higher temperatures than previously considered possible, and that its formation can be very rapid.)
 - a. The O:U ratio increased from 2.01 to 2.10.
 - b. Increasing the O:U ratio has been reported to decrease the effective thermal conductivity of UO_2 fuel by as much as 40%. (5, 6)
 - c. Limited laboratory experiments show that increasing the O:U ratio to approximately 2.15 decreases the melting temperature of UO_2 fuel by as much as 400 °C. (7)

3. The melting transition occurred in the fuel during the PRTR power increase from 49.3 to 56.5 MW.
 - a. Increased fuel temperatures were evidenced by an increase in the FERTF coolant DN activity of approximately 100 times which occurred during this power rise.
 - b. The FERTF tube power and fuel element peak specific rod power increased from 1330 to 1530 kW and 20.6 to 23.3 kW/ft, respectively, during this reactor power increase.
4. The reactor power was then increased to 60.9 MW and subsequently to 65.3 MW. The corresponding FERTF tube powers were 1650 and 1770 kW and peak specific rod powers were 25.4 and 27.3 kW/ft at these power steps. At each reactor power level step of 56.5, 60.9, and 65.3 MW, the coolant temperature was cycled so that the coolant outlet temperature was decreased from 500 °F to 300 °F and back to 500 °F. Coolant very likely re-entered the defect hole during the cooling portions of the temperature cycle.
5. An unexpected large amount of fuel melting occurred within the defected rod because of the reduced thermal conductivity^(5, 6) and/or lower melting point⁽⁷⁾ of the oxidized fuel. Unexpectedly high fuel temperatures along the full length of the defected rod were evidenced by the postirradiation examination.
 - a. A maximum of 75% of the fuel radius was molten near the defect. This may be compared with molten fuel radii of 46 to 64% found at the same plane in other nondefected rods from the fuel element.
 - b. The defected fuel rod may have operated at a slightly higher power generation than did the other rods of FE-6504 due to the expulsion of fission product gases. A power increase of approximately 8% has been calculated for complete escape of the fission gas poisons.⁽²⁴⁾

6. The volume expansion caused by the unexpectedly large amount of molten fuel could not be accommodated by the void space within the fuel and high internal hydrostatic pressures resulted.
 - a. The void space in the 86% theoretical density fuel has been calculated to be able to accommodate the volume expansion of fuel molten to 60% of the radius.⁽¹⁰⁾
 - b. The specific volume expansion of UO_2 from room temperature up to the melting point is reported to be approximately 15%.⁽¹¹⁾ The specific volume expansion in passing through the melting transition is 10%.⁽¹²⁾ Therefore, from room temperature through the melting transition, the specific volume expansion of UO_2 is approximately 25%.
7. The cladding temperatures were higher than normal in the vicinity of the defect.
 - a. The largest amount of fuel melting occurred in this area since the intentional defect was placed near the peak flux region.
 - b. The sintering and molten fuel boundaries were displaced closer to the cladding in the direction of the defect due to a localized reduction in the fuel-clad contact coefficient because of an increased gap or the presence of water and/or steam in the immediate vicinity of the defect.
 - c. It is also possible that the fuel was more severely oxidized in the immediate vicinity of the defect to further reduce the effective thermal conductivity and melting point in this region.
8. High internal hydrostatic pressures and higher than normal cladding temperatures caused the cladding to swell locally and split in a ductile manner in the vicinity of the defect.
9. High temperature fuel contacted the coolant and pressure tube at the ductile split. The exposure of high temperature fuel to the coolant and blockage of the flow path with heat generating material caused steam formation to occur at this point and for 3 to 4 in. downstream about 24 sec before the pressure tube failed.

- a. There was no appreciable loss of fuel to the coolant when the cladding split occurred. A significant fuel loss to the coolant would be expected to cause an alarm on the RAM units.
 - b. The pressure tube did not overheat and burst at this point because of liquid coolant availability.
 - c. The ΔP increase across the FERTF measured 24 sec before the pressure tube burst can be calculated to be consistent with steam formation of this type.
10. Steam formed downstream of the ductile split, blanketed the heat transfer surface, and prevented adequate heat removal during the next 20 to 24 sec.
 - a. The cladding overheated rapidly in the presence of the steam blanket.
 - b. The overheated cladding rapidly oxidized. Oxide layers as thick as those found on the remaining cladding would have formed under these conditions in times on the order of 20 sec. (22)
 - c. High temperatures, oxidation, and local hydriding severely weakened the cladding.
11. As the region of the fuel rod immediately downstream of the swelled area overheated due to steam blanketing, the adjacent Zircaloy pressure tube became overheated and failed under pressure. Two possible mechanisms to explain the pressure tube overheating are as follows:
 - a. One mechanism that may have caused the overheating of the pressure tube is radiant heat transfer from the overheated fuel rod.
 - b. The cladding may have melted or swelled allowing a skull of hot fuel and cladding to contact the pressure tube causing the tube to rapidly overheat and rupture.
12. After the pressure tube failed, fuel (mostly molten) was forceably ejected from the rod under the pressure differential

created between inside of the rod and the decreased FERTF coolant pressure. Approximately 25% of the fuel escaped through the hole in the pressure tube with the coolant discharge.

13. Loss of hot fuel (mostly molten) from the rod caused a large amount of steam formation within the pressure tube. The movement of this large steam volume downstream caused a momentary boiling burnout condition in the power peaked end region of some outer rods.
 - a. The burnout condition at the downstream ends of some of the outer rods did not persist long since the cladding would not have been physically able to endure at the high temperatures. There is no evidence of an increased thickness ZrO_2 layer, and there is no metallographic evidence of increased hydriding.
 - b. The cladding temperatures reached at the hot spots have been calculated to occur within about 2 sec after the occurrence of boiling burnout.
 - c. The large amount of steam formation caused the further increase in FERTF pressure drop noted at the time of the pressure tube failure.
14. An additional event may have occurred which possibly affected the rupture mechanism. The high hydrostatic pressures in the fuel rod may have caused molten fuel to break through the eroded top depleted UO_2 pellet and enter into the gas plenum. There is no evidence to determine when this occurred during the irradiation. The subsequent heating of the plenum gas may have caused a pressure increase within the rod which could have affected the initial cladding split. This is speculative and its occurrence is not an essential part of the mechanism.

SUMMARY AND DISCUSSION OF FUEL EVALUATION DATA

Fuel Structures Formed During Irradiation

The fuel structures formed in rods from FE-6504 during irradiation are typical of those formed in molten UO_2 fuel specimens. In general, in-reactor sintering and densification of the fuel cause the formation of a central void in the thermal center of fuel rods. A central void may also result from the shrinkage of molten UO_2 when it solidifies. The central void is surrounded by a well defined porous fuel region which represents the extent of fuel melting during irradiation. Outside the well defined porous fuel region is a band of large pore-free, high density grains. Beyond the high density region composed of large grains, is a band of columnar grains and then a narrow region of equiaxed grain growth. The equiaxed grain growth region is surrounded by an annulus of structurally unaffected lower temperature fuel which extends to the inside surface of the cladding. Variations of these basic structural characteristics occur along the axis of long continuous fuel elements like those irradiated in PRTR. In certain regions of the fuel rod, particularly the upper portions, evidence of the porous or once-molten region is generally present only on the inner edge of the central void. In other regions (primarily the lower portions) of a long fuel rod which operated with fuel melting, there is no center void at all.

FE-6504 was irradiated in the FERTF positioned in the center of the PRTR core. The FERTF was surrounded by a flux trap and a ring of similar elements which had nearly the same exposure. There were no shim rod movements or other events in the proximity of FE-6504 during its irradiation in the PRTR and, hence, there should have been no perturbations to cause power generation variations from rod to rod within the element. This ideal geometrical situation permits a comparison of the irradiation behavior of fuel rods which experienced the same irradiation history.

Comparison of fuel structures in rods from the 12-rod ring of FE-6504 shows a wide variation of structural characteristics for rods irradiated under the same conditions, as summarized in Table II. Cross sections of four rods (excluding the ruptured rod FR-4) from the 12-rod ring were examined at three different planes along their length. The results show that the size of the central void among the four nondefected rods in a plane 20 in. down from the top ranges from 32 to 38% of the radius. At a plane 30 in. down from the top of the rods, the size of the central void ranges from 8 to 26% of the radius and at a plane 20 in. from the bottom, the size of the central void ranges from 0 to 6% of the radius. Also, at a plane 20 in. down from the top of the four nondefected fuel rods, the region of fuel melting ranged from 23 to 50 % of the radius; 30 in. down from the top of the rods, the fuel melting region ranged from 46 to 64% of the radius; and at a plane 20 in. from the bottom of the rods, the melt region varied from 54 to 65% of the radius. These observations indicate that the physical behavior of vibrationally compacted fuel operating in a long fuel rod under high power generating conditions varies noticeably, even in like rods exposed to the same irradiation conditions.

Postirradiation examination of nondefected rods shows that the size of the central void in the upper portion of the rod is generally larger than it is at, and below the midplane of the fuel column. This indicates that axial fuel relocation occurs in vibrationally compacted fuel rods which operate with central melting, although in most cases there was no measurable change in balance point. This would indicate that the balance point technique for determining axial fuel relocation is insensitive or the difficulties experienced in making such measurements in remote facilities preclude obtaining the required accuracy.

Fuel rod swelling was evidenced by diameter increases in the lower portion of rod FB-59. Examination of fuel structures formed during irradiation indicate that as much as 64% of the radius was molten in the swelled region of the rod. Calculations indicate that fuel rod swelling can be expected under these conditions since there is not sufficient void volume within the fuel (excluding the plenum) to accommodate the volume expansion of this amount of molten material.^(9, 10) The specific volume expansion of UO_2 from room temperature up to the melting point is reported to be approximately 15%.⁽¹¹⁾ The specific volume expansion in passing through the melting transition is approximately 10%.⁽¹²⁾ Therefore, from room temperature through the melting transition, the specific volume expansion of UO_2 is approximately 25%. This expansion must be accommodated by the void volume within the fuel rod to avoid swelling. It is not known why other rods in FE-6504 operating under conditions similar to rod FB-59 did not show evidence of swelling.

In spite of the fuel structural variations in nondefected rods irradiated under the same conditions, postirradiation examination results indicate that a greater degree of fuel melting existed in ruptured rod FR-4 than in nondefected rods from FE-6504, even though irradiation histories were the same. In the ruptured rod FR-4, as much as 67% of the radius was molten at a plane 20 in. from the top of the rod, 73 to 75% of the radius was molten at the plane of the rupture 30 in. down from the top of the rod, and 73% of the radius was molten at a plane 20 in. up from the bottom. These values are compared to 23 to 50% of the radius molten at a plane 20 in. from the top, 46 to 64% of the radius at a plane 30 in. down, and 54 to 64% at a plane 20 in. from the bottom for nondefected rods from FE-6504. The unexpected large amount of fuel melting in the ruptured rod could have been caused by a decrease in the effective thermal conductivity of the fuel, a decrease in the melting temperature of the fuel, a slightly higher power generation due to the venting of fission gases, or a combination of these possible effects. Variations in the fuel-to-jacket thermal contact would also affect the amount of fuel melting.

Increasing the O:U ratio from 2.00 to 2.15 has been reported to decrease the effective thermal conductivity of UO_2 by as much as 40%. (5, 6) Calculations of melt radius in a vibrationally compacted UO_2 -4 wt% PuO_2 PRTR fuel rods as a function of linear rod power for different values of effective thermal conductivity show that the melt radii indicated in the ruptured rod are reasonable (Figure 57). (These calculations assumed a constant fuel-clad contact coefficient of $2000 \text{ Btu/hr-ft}^2 \cdot ^\circ\text{F}$.) A possible reduction in the melting temperature of the fuel would further increase the melt radius. It has been reported that increasing the O:U ratio from 2.00 to 2.15 decreases the melt temperature of UO_2 by as much as 400°C . (7)

Postirradiation fuel structures in ruptured rod FR-4 were unique in other respects. Most notable was the abnormally large, unobstructed, central void that extended over essentially the full length of the rod. The large central void resulted from the large amount of fuel melting and the fact that 700 g or 40% of the fuel was expelled from the rod at the time of the pressure tube rupture. Postirradiation examination indicated that it was primarily molten fuel that was lost from the rod. The molten fuel would have been quickly and forcibly ejected from the rod under the pressure differential created between the inside of the rod and the decreased FERTF coolant pressure when the pressure tube failed. There is evidence of a small amount of once-molten fuel remaining near the bottom end of ruptured rod FR-4 and a thin skull of once-molten fuel on the inner surface of the central void in the lower portion of the rod. In the upper portion of the rod, however, there is no evidence of once-molten fuel remaining.

There was sufficient molten fuel in the ruptured rod to form a continuous molten fuel column from one end of the rod to the other. The continuous molten fuel column eroded through the depleted UO_2 pellet at the top of the rod which permitted molten fuel to pass into the plenum region for a distance of 4.2 in. or 60% of the plenum length. It is not

FIGURE 57. Calculated Fuel Radius Above 2800°C Versus Linear Rod Power Generation for a Vibrationally Compacted UO₂-4wt% PuO₂ Fuel Rod Operating in PRTR.

known when this occurred during the irradiation. Molten fuel could have passed into the plenum region as a result of the relief of the internal pressure caused by the expansion of the large molten fuel volume in the rod or by vapor phase transport. The structure of the peripheral fuel in the plenum region indicates that it was once-molten fuel; however, the fuel adjacent to the central void has lost its once-molten appearance. The upward movement of the depleted UO_2 pellet of approximately 0.15 in. against the downward force of the plenum spring is further evidence that the volume expansion of the molten fuel exerted some internal pressure.

Examination of Cladding From Ruptured Rod FR-4

The proposed rupture mechanism postulates that the high internal hydrostatic pressure exerted within the defected rod by the expansion of the molten fuel caused the cladding to split initially in a ductile manner in the vicinity of the defect. The largest amount of fuel melting within the rod occurred in this area since the intentional defect was positioned near the peak flux region. Also, it is possible that the fuel was more severely oxidized in the immediate vicinity of the defect to more drastically reduce the effective thermal conductivity and melting point in this region. In addition, examination of a previous molten-core defect test shows that the symmetry of the fuel structures formed in the vicinity of the defect was affected by the presence of the defect.^(3, 15) The sintering and molten fuel boundaries were displaced closer to the cladding in the direction of the defect. This effect probably resulted from a localized reduction in the fuel-cladding contact coefficient because of an increased gap or the presence of water and/or steam in the immediate vicinity of the defect. It is reasonable to suspect that similar phenomena occurred in the intentionally defected rod prior to the rupture.

Swelling associated with the initial split permitted the cladding and hot fuel to contact the Zircaloy pressure tube which did not burst at this location because of coolant availability. This occurred without the detectable loss of fuel to the coolant. In fact, the evidence indicates that no appreciable amount of fuel escaped from the ruptured rod until the pressure tube burst. Similar split-type failures of packed particle UO_2 fuel rods operating under molten-core conditions have been experienced by other investigators without the loss of fuel to the coolant system. (25, 26, 27) In the case of FE-6504, however, the localized swelling and contact with the pressure tube caused disruption of the coolant flow and created a steam blanket and localized overheating of the cladding immediately downstream of the intentional defect. The overheating situation may have been further aggravated by the close proximity of the high temperature fuel to the cladding in the immediate vicinity of the defect.

As the region of the fuel rod immediately downstream of the swelled area overheated due to steam blanketing, the adjacent Zircaloy pressure tube became overheated and failed under pressure. The mechanism by which the Zircaloy pressure tube became overheated is not known, but two possible mechanisms to explain the overheating are as follows:

- a. One possible mechanism that may have caused overheating of the pressure tube is radiant heat transfer from the overheated fuel rod. Radiant heat transfer calculations indicate that the tube wall could not have reached the temperature indicated by metallographic examination except under conditions which are thought to be doubtful. However, pressure tube wall temperatures sufficient to cause deformation and failure under pressure ($\sim 600^\circ\text{C}$) are more likely. Heating of the pressure tube to the metallographically indicated temperatures ($> 1000^\circ\text{C}$) at the edge of the rupture could possibly have been caused during the passage of hot fuel through the hole after failure. While it is

felt that sufficient heat transfer by this manner was not likely, it would account for the unusual isotherms indicated by the metallographic structures formed in the vicinity of the pressure tube rupture.

- b. Another possibility is that the cladding melted or swelled allowing a skull of hot fuel and cladding to contact the pressure tube causing the tube to rapidly overheat and rupture. If this occurred, the calculations show that the pressure tube would overheat and fail within a matter of a few seconds. While this might be possible, there is no evidence of fuel contacting the pressure tube in the immediate vicinity of the rupture, and there was no evidence of fuel loss to the FERTF coolant prior to the failure of the pressure tube.

After the pressure tube failed, the molten fuel in the rod was quickly ejected under the pressure differential created between the inside of the fuel rod and the reduced coolant pressure. Loss of hot fuel from the rod caused a large amount of steam formation within the pressure tube. The movement of this large steam volume downstream caused a momentary boiling burnout condition in the power peaked end region of some outer rods.

Localized overheating of the Zircaloy cladding in the rupture region is evidenced by the presence of recrystallized and beta-transformed structures indicative of maximum temperatures well over 1000 °C. In fact, some of the Zircaloy cladding has a porous cast structure indicative of melting. Thick zirconium oxide layers formed on the outside and inside surfaces of the cladding near the rupture.

Examination of the Zircaloy cladding in the ruptured region of rod FR-4 shows evidence of once-molten fuel material adhering to the inner and outer surfaces near the edge of the break as well as fuel alloying with melted cladding. Zirconium oxide layers as thick as 100 microns

were measured at the edge of the break. In some cases, the oxide layer extends around the edge of the fracture. Thick oxide layers are present on the surface of grossly distorted cladding indicating that the cladding distortion existed for a sufficiently long period of time to allow the layer to form before the pressure tube ruptured. Formation of thick zirconium oxide layers around the edge of the fracture and over fuel-clad alloy also suggests that the cladding failed and came in contact with molten fuel sometime before the pressure tube ruptured and the reactor scrambled.

Examination of the ZrO_2 layers, the oxygen-rich diffusion layers, and the structure changes that occurred in the Zircaloy cladding in the vicinity of the rupture indicates that the metal corroded at high temperature and that it was not an accelerated low-temperature corrosion phenomenon.⁽²²⁾ One of the thickest ZrO_2 layers observed at the edge of the fracture is 4.3 mils thick (Figure 18). The thickest oxides occur in areas of the cladding where the wall thickness appears to suddenly increase. If this sudden cladding thickness change represents a melting or softening boundary, it would have been at a temperature of approximately 1800 °C. An extrapolation of zirconium corrosion data in high-temperature water⁽²³⁾ indicates that ZrO_2 thickness of this magnitude could be expected to form in approximately 6 sec under these conditions. Another interpretation of the data indicates that ZrO_2 of this thickness (approximately 4 mils) would form in approximately 22 sec at a temperature of 1700 °C. Thus, the available data indicates that a piece of Zircaloy heated for 22 sec in steam at temperatures near the melting point (1800 °C) would be expected to form oxide thicknesses similar to, or thicker than those observed on the cladding in the region of the rupture.

Examination of the Top End Region of Nondefected Rods

Four rods in the 12-rod ring of FE-6504 developed "hot" spots on the outside surface of the cladding approximately 8 1/4 in. down from the square shoulder of the top end cap during the second period of irradiation in the FERTF. The occurrence of the overheated rods is sporadic since only four rods in the 12-rod ring exhibited this phenomenon. Three of the affected rods were positioned 120 degrees apart. The Zircaloy cladding has overheated locally beyond the transformation temperature indicative of temperatures in excess of 1000°C. There is no metallographic evidence of increased hydride formation associated with the overheated regions. The axial position of the overheated areas is such that a circumferential band could have been located in this region after the rebanding operation in the basin. The presence of the band could have contributed to the overheating condition.

The fuel material in the region of the overheated cladding also shows evidence of excessive temperatures. Sintering and grain growth occurred to the surface of the fuel adjacent to the cladding. In some instances, there is evidence that molten fuel flowed radially out through cracks to possibly contact the cladding, although there is no evidence of core-clad reaction. In fact, there is no evidence of core-clad reaction in any of the locally overheated regions. These observations, plus the fact that the overheated cladding remained intact, there was no metallographic evidence of increased hydriding, and no thick zirconium oxide layer on the cladding; suggest that the overheating condition in the ends did not persist for any appreciable time.

Autoradiographs of the top end regions of affected and nonaffected rods indicate that the UO_2 -1 wt% PuO_2 graded enrichment fuel mixed with the UO_2 -4 wt% PuO_2 fuel in each to approximately the same degree. This was confirmed by chemical analysis which shows that the fuel composition near the top ends of affected and nonaffected rods was nominally 4 wt% PuO_2 .

The flux wire scan on FE-6505, which was irradiated in the FERTF prior to defected FE-6504, provides an indication of the power generation distribution in the end regions of these rods. These data indicate that even with no graded enrichment in the ends of the fuel rods, power generations of the magnitude evidenced by fuel structures and temperatures in nominally UO_2 -4 wt% PuO_2 fuel, should not have been attained and that the maximum power generation should have been generated near the center of the rod. It is, therefore, indicated that the high fuel temperatures observed in the top end regions of some rods were the result of localized cladding overheating and not the cause of the overheating. It is postulated that the localized burnout condition was caused by momentary steam blanketing at the time of the rupture.

Fission Gas Release

The percent Xe+Kr release for the 6 rods sampled from the 12-rod ring of FE-6504 ranges from 75 to 88%. This degree of scatter is considered small for this type of data. In addition, the percent release values reflect the small differences in volumetric average fuel temperatures for these rods as determined by postirradiation examination of fuel structures formed during irradiation.

GENERAL DISCUSSION

This report has outlined in some detail, a step-by-step sequence of events through the course of the fuel rod and pressure tube ruptures. Other mechanisms could be postulated. The proposed mechanism is presented in the belief that it best accounts for most of the evidence accumulated concerning the event. Proof of the accuracy of the mechanism is not possible, and variations of the details cannot be ruled out. Although the effect of several unique circumstances associated with this experiment upon the rupture mechanism is unknown, it is felt they are not an essential part of the mechanism as proposed.

It should be emphasized that the actual damage to the reactor was minor. Some equipment required maintenance due to being flooded, but no major components required replacement except the ruptured pressure tube. The cost of the incident arises primarily from the expense of replacing the degraded heavy water moderator and from the time consuming task of decontaminating the reactor.

The extensive postmortem examination associated with the incident has provided a large amount of valuable information related to the irradiation behavior of vibrationally compacted mixed-oxide fuel rods operating under high power generating conditions in a water-cooled environment. Most noteworthy, perhaps, are the indicated behavioral differences associated with nondefected and defected mixed-oxide fuel rods and apparent differences in fuel temperatures between nondefected rods presumably operating under the same conditions.

The irradiation behavior of intentionally defected vibrationally compacted $\text{UO}_2\text{-PuO}_2$ fuel rods operating under high performance conditions does not seem to be completely consistent since one defected rod behaved satisfactorily while operating under molten core conditions whereas, another defect test ruptured. A direct comparison of the irradiation results of these two experiments may be invalidated by the fact that the irradiation conditions were not precisely the same. For instance, the rod that behaved satisfactorily did not have a fission gas plenum and operated at a maximum power generation of 24 to 25 kW/ft. The ruptured rod contained a fission gas plenum; it operated at a maximum power generation of 27 kW/ft; and the coolant temperature was cycled during irradiation.

Fuel temperatures in the defected rod that behaved satisfactorily were approximately as expected whereas, indicated fuel temperatures were unexpectedly high in the ruptured rod. The high fuel temperatures in the ruptured rod, which is believed to be the underlying cause of

failure, are attributed to oxidation of the fuel by the water coolant and the consequent reduction in effective thermal conductivity and/or melting temperature. This behavior is not believed to be unique with vibrationally compacted pneumatically impacted $\text{UO}_2\text{-PuO}_2$ fuel and the reason for the apparent difference in the oxidation behavior of the fuel in the two experiments is unknown.

It is possible that a "rupture threshold" condition has been experienced. In any event, the cause of the rupture is more directly related to the defect in the rod than to operation with molten vibrationally compacted mixed-oxide fuel per se. These results point out important areas of fuel behavior that will be intensively investigated during the High Power Density Program in PRTR.

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- W. K. Winegardner - Out-of-reactor loop experiments on irradiated fuel.
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- M. C. Fraser and P. J. Pankaskie - Pressure tube examination.
- B. Griggs - Analysis of the corrosion aspects related to the incident.
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