A COMPILATION AND REVIEW OF EXISTING SAFETY ANALYSES
FOR FUELED SUBASSEMBLIES IN EBR-II

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ABSTRACT

The results of studies dealing with the safety considerations of fueled subassemblies in EBR-II are summarized. Most of these studies are concerned with the response of oxide-fueled subassemblies to the abnormal incidents postulated and analyzed in the EBR-II safety documents; others deal with the overall behavior of the reactor or with the response of the plant protective system to these incidents. Safety studies on metal-fueled subassemblies are also presented so that a basis for comparison can be established.

I. INTRODUCTION

A knowledge of the operational behavior, under steady-state and transient conditions, of subassemblies fueled with mixed oxides is of primary importance to the LMFBR program. As a part of this program, an increasing number of experimental subassemblies fueled with mixed oxides are being irradiated in EBR-II. The analyses in the original EBR-II safety documents^{1,2} (referred to hereafter as ANL-5719 and ANL-5719A) were concerned with the operational characteristics of the uranium metal-5 wt. % fissium alloy fuel elements and the depleted-uranium blanket elements. It was necessary, therefore, to analyze the behavior of the mixed-oxide fuel under the same conditions to determine if the operation of the reactor with a significant oxide-fuel inventory is consistent with the intent of the safety documents. The results of these analyses have been compiled and reviewed and are summarized in this report. Although this report will not present all details of each analysis, the pertinent conclusions and salient discussion will be described.

Many of the pertinent safety questions have been addressed in Ref. 3, which is primarily in support of the operation of EBR-II as an irradiation facility at its design power level of 62.5 MWt. The report you're reading summarizes a class of pertinent safety-related analyses in a form suitable for easy cross-comparison of the analyses. Because Ref. 3 discusses many of these analyses in considerable detail, it will be referenced often in this report.

In the following sections, a number of incidents are hypothesized, and the consequences are determined. These incidents have been studied in order to examine the behavior of the fuel and the reactor under the most severe imaginable conditions and to determine the indigenous safety characteristics of the various fuels in the reactor. It should be recognized that for these incidents to occur would require either or both of the following conditions: gross mismanagement or a complete loss of administrative control; and a complete malfunctioning of all safety devices in the system.

II. NUCLEAR INCIDENTS

The first conditions considered in this report are those which would exist during and after various uncontrolled reactivity insertions. Six reactivity-insertion incidents (cases) were studied (except for Case 4, the reactor was under zero-flow conditions):

1. Insertion of safety rods at the normal speed of 2 in./min into a delayed critical reactor at zero power.*
2. Insertion of the central driver subassembly at the regular speed of 6 in./min over the last 24 in. of travel into a delayed critical reactor at zero power.
3. Insertion of a single control rod at 5 in./min into a delayed critical reactor at zero power.
4. Same as Case 3 except that the reactor is at full power and full flow.
5. Dropping of the central driver subassembly into a delayed critical reactor at zero power.
6. Same as Case 2 except that the subassembly is inserted at 72 in./min.

* At zero power, the reactor is at the source power level of approximately 8 W.

Under the assumption that the plant protective system (PPS) failed to respond, the above incidents were examined for the original, metal-fueled core in Appendix A of ANL-5719. The consequences reported there are summarized in the second column of Table I. Case 5 represents the most severe incident because fuel melting would be initiated in less than 0.2 sec. The reactor would be prompt critical at this time, and the subsequent course of the excursion was considered to be uncertain.

The last column of Table I lists the approximate times required to reach the reactor period trip setpoint* and the power-level trip setpoint for each of these incidents. The times to reach the period trip point were taken from a recent study⁴ of the PPS. The times to reach the power-level trip point were estimated from the appropriate curves of power versus time presented in a more up-to-date analysis of these reactivity-insertion incidents.⁵ Except for Case 5, the reactor could be scrammed by either the period trip or the power-level trip before fuel melting or eutectic formation would be initiated. For Case 5, fuel melting would be initiated even though the reactor scrammed almost immediately.

These same reactivity-insertion incidents were studied in Appendix F of ANL-5719A. This document takes into consideration all the differences between the original design of the reactor (the basis of ANL-5719) and its actual construction. As shown in Table I, under the column entitled ANL-5719A, the consequences were found to be about the same as in the original studies.

Additional analyses⁵ of these incidents were made in which the prompt feedbacks due to fuel expansion and coolant-density changes were taken into account. These analyses assumed the reactor to be under full-flow conditions in all cases. The results of these analyses are summarized in Ref. 3 and are shown in Table I under the heading of Subsequent Analyses. These results agree with those reported in the earlier documents. The maximum reactivity inserted into the core for the Case-5 incident would be 1.08\$ (i.e., the reactor would be prompt critical), but because of the feedback effects mentioned above, would be reduced to 0.99\$ at the onset of fuel melting.

*The reactor period is the time required for the flux level to increase or decrease by a factor of e. EBR-II is equipped with a scram mechanism that will activate if the period becomes less than 25 sec. The period trip times in the table are the actual times that would elapse between the initiation of the incidents and the scrambling of the reactor by this mechanism.

TABLE I. Consequences of Nuclear Incidents: All-metal-fueled Core

Case No.	Effect on Fuel		Subsequent Analyses	Time to Reach Trip Setpoints
	ANL-5719	ANL-5719A		
1	Fuel melting after 120 sec	Fuel melting after 120 sec	Fuel melting after 121 sec	Period trip at ~25 sec; power-level trip at ~90 sec
2	Fuel melting after 55 sec	Fuel melting after 55 sec	Fuel melting after 49.8 sec	Period trip at ~9 sec; power-level trip at ~40 sec
3	Probably no fuel melting	Possible fuel melting after 230 sec, but eutectic temperature will be exceeded	Fuel melting after 220 sec; eutectic formation after ~180 sec	Period trip at ~32 sec; power-level trip at ~130 sec
4	Fuel melting before insertion is completed	Eutectic formation beginning after 20-50 sec	Fuel melting after 94.5 sec	No period trip; power-level trip at ~10 sec
5	Fuel melting within 0.17 sec	Fuel melting after <0.13 sec	Fuel melting within ~0.1 sec	Period and power-level trips immediately activate; response delays may be too long to prevent melting
6	Fuel melting after 5.3 sec	Fuel melting after ~6 sec	Fuel melting after 5.5 sec; eutectic formation after ~5.4 sec	Period trip at <.1 sec; power-level trip at ~4 sec

Other subsequent analyses have been made for the all-metal-fueled reactor under reduced-flow conditions and for more-recent irradiation cores under reduced- and full-flow conditions. (For all the analyses, the reduced flow was 5.5% of nominal full flow. This flow approximates that due to the auxiliary pump plus thermal convection.) Table II summarizes the results.

As under full-flow conditions, Case 5 again represents the worst case for the all-metal-fueled reactor and would lead to fuel melting after about 0.1 sec. In each of the other cases, however, formation of the uranium alloy-stainless steel eutectic would be initiated at the times shown in Table II, and, in most cases, coolant boiling and fuel melting would follow. As a general statement, under reduced-flow conditions the loss of fuel-element integrity in all but the most severe case would be due to cladding failure rather than centerline fuel melting.

The reactor period is dependent on various feedback effects including those of the fuel and the coolant. Because the above analyses were made under reduced-flow conditions, the feedback from the coolant was different than it would be under full-flow conditions. A more detailed study of the feedback effects, which will include an evaluation of the time required to reach the 25-sec period trip setpoint under reduced-flow conditions, will be made in the future. Comparison of Tables I and II shows that the times required to reach the power-level trip point are approximately the same under reduced-flow and full-flow conditions. It should be noted, however, that these times are only approximate values taken from curves intended to show overall trends rather than absolute values.

Cases 1, 2, and 4 were analyzed for more-recent irradiation core loadings under full-flow conditions. Each of these loadings contained several subassemblies with mixed-oxide-fuel elements and some subassemblies of experimental structural material. It was found that fuel melting would be initiated in the oxide-fuel elements at the times shown in Table II, but that the temperature in the metal-fuel elements would rise so slowly that neither fuel melting nor eutectic formation would occur. The overall conclusion of the study of these cores at full-flow conditions is that, for the reactivity-insertion incidents considered, the initiation of fuel melting would be in the oxide-fuel elements rather than in the metal-fuel elements.

TABLE II. Consequences of Nuclear Incidents at Reduced Flow
and for Other Types of Cores

<u>Case No.</u>	<u>Effect on Fuel</u>	<u>Power-level Trip, sec</u>
<u>All-metal-fueled Core:</u>		
<u>Reduced Flow (5.5% of Full Flow)</u>		
1	Eutectic formation after 115 sec	~100
2	Eutectic formation after 45 sec	~40
3	Eutectic formation after 165 sec	~130
5	Fuel melting after 0.1 sec; eutectic formation after 0.15 sec	~0.08
6	Eutectic formation after 5.2 sec	~4
<u>Irradiation Cores: Full Flow</u>		
1	Oxide melting after 190 sec; metal fuel doesn't exceed 1000°F	~145
2	Oxide melting after 62 sec; metal-fuel temperature rises slowly	~50
4	Oxide melting after 65 sec; little change in peak metal-fuel temperature	
<u>Irradiation Cores:</u>		
<u>Reduced Flow (5.5% of Full Flow)</u>		
1	Eutectic formation at 115 sec; peak oxide temperature is 1740°F at this time	~100
2	Eutectic formation at 46 sec; peak oxide temperature is 1680°F at this time	~40

The times required to reach the power-level trip setpoint for Cases 1 and 2 with the recent irradiation core loadings are shown in Table II. As before, these are only approximate values, but they do show that the time required to reach the trip point at full flow is longer for the irradiation cores than for the original all-metal-fueled core.

Analysis of Cases 1 and 2 for irradiation cores at reduced flow showed that the order of events under these conditions is greatly different than under full-flow conditions. As shown in Table II, each of these two cases would result in the initiation of formation of the uranium alloy-stainless steel eutectic in the hottest metal fuel element before melting would be initiated in the hottest oxide-fuel element. These events would occur in the opposite order in which they occur under full-flow conditions, because the coolant temperature would increase much more rapidly during the transient. This temperature rise would result in a strong, negative reactivity feedback that would limit the magnitude of the power burst. The oxide-fuel elements, because of their greater heat capacity and significantly lower thermal conductivity, would thus be protected while the metal-fuel elements could undergo eutectic formation with the stainless steel cladding.

It can be seen from Table II that the times at which the power-level trip point would be reached are somewhat shorter for irradiation cores under reduced-flow conditions than under full-flow conditions. The trip times under reduced-flow conditions are, in fact, about the same as those for the all-metal-fuel core. This fact and the above finding that the effects of the transients of Cases 1 and 2 are greatest on the metal-fuel elements show that, under the initial conditions of zero power and reduced flow, the dynamic behavior of the reactor is more dependent upon the behavior of the metal-fuel pins than on that of the oxide-fuel pins. On the other hand, the behavior of the oxide-fuel pins tends to dominate under full-flow conditions.

As pointed out in Ref. 3, measurements in the critical-assembly mockups in ZPR-3, the EBR-II wet critical, and during EBR-II run 37 show that the reactivity-insertion rates used in all the above analyses were unrealistically high. If the more realistic rates were used, there would

generally be more time available before undesirable consequences (e.g., eutectic formation and/or fuel melting) are initiated. There would be, therefore, a greater possibility of protective response and/or administrative action that could terminate the incident and thereby avert any serious consequences.

The results presented so far have all been obtained under the assumption that the plant protective system (PPS) failed to respond. To evaluate the response of the PPS, including the delay times in the circuitry, the six cases of reactivity insertion have been reanalyzed under the assumptions of full or partial response of the PPS. This work has been documented in Ref. 4 and is summarized in Ref. 3. The various conditions studied and the assumptions made in Ref. 4 are too numerous to reiterate in the current report. The overall conclusion made in Ref. 4, however, is that there is responsive and redundant protection for all the hypothetical component malfunctions considered in the safety documents.

It is pointed out in Ref. 4 that period-trip response times of less than 50-100 msec would be required to protect the reactor in Case 5, the inadvertent dropping of a central subassembly into a just-critical reactor. The measured response times, however, are on the order of 300 msec, so this shutdown mechanism would not prevent the consequences noted earlier. It is further pointed out that a variable power-level trip point, having a response time of 30-50 msec, would protect the reactor against any serious consequences resulting from this incident.

III. LOSS OF PRIMARY-SYSTEM FLOW

A. Analyses in Safety Documents ANL-5719 and ANL-5719A

Another possible source of a major accident is loss of power to the primary pumps. Three major cases were analyzed in the safety documents: (I) loss of all pumping power followed immediately by reactor scram; (II) reactor scram followed seconds later by a loss of all pumping power; and (III) loss of all pumping power, but with all control rods remaining fixed in their operating positions. In each case, the reactor was initially at full power and full flow.

Case I was studied under several assumptions of total control-rod worth, the degree of initial rod insertion, and the scram delay time. Table III shows the major consequence of Case I under the most-pessimistic set of assumptions (i.e., Case 1.1 in ANL-5719). The peak fuel temperature would be approximately 1300°F for this incident and would occur about 10 sec after the loss of pumping power. It is not believed that this case presents a serious safety problem.

Case II was also studied under the assumptions described above. Table III shows the results under the most pessimistic assumptions. Again, there appear to be no serious safety problems for this incident.

As would be expected, the most severe loss-of-flow (LOF) incident would be Case III. Under these conditions, fuel melting would begin in the hottest element within about 3 sec.

These same LOF incidents were reconsidered in ANL-5719A. Some modifications had been made in the plant layout, and these were reflected in the analysis of the incidents. For example, mechanical centrifugal pumps had been installed instead of electromagnetic pumps, and, as a result, the flow coastdown is somewhat slower than for the original system. Where applicable, the control-rod acceleration during a scram was changed to 1.5 g from the original value of 3 g. As shown in Table III, these changes had little effect on the original analysis. The only significant change is that, in Case III, there is now a 30-sec delay before fuel melting would be initiated.

These incidents were studied later in a manner that took into account the prompt feedback due to fuel expansion and coolant-density changes. As stated in Sec. VIII of Ref. 3, the results were found to be essentially the same as in the original safety documents.

B. Studies of Effectiveness of Plant Protective System

The set of LOF studies discussed in Sec. A, above, has inherent assumptions regarding the response of the PPS. If a loss of pumping power should actually occur while the reactor is at power, the primary flow rate would decrease and the coolant outlet temperature would, therefore, increase. Both of these parameters have trip points and should activate the control

TABLE III. Fuel Temperatures upon Loss of Primary-system Flow under
Most-pessimistic Assumptions: All-metal-fueled Core

<u>Case No.</u>	<u>ANL-5719</u>	<u>ANL-5719A</u>	<u>Subsequent Analyses</u>
I	Peak fuel temperature is ~1300°F	Peak fuel temperature is ~1280°F	Essentially the same
II	Peak fuel temperature is ~1330°F	Peak fuel temperature is ~1280°F	Peak fuel temperature is ~1300°F
III	Fuel melting in about 3 sec	Fuel melting in about 30 sec	Essentially the same

rods. Studies of the effectiveness of these trips were included in the previously mentioned analysis of the PPS.⁴

The first study of the effectiveness of the coolant outlet-temperature trip was made by setting the trip point at 940°F.* Various time delays between the loss of pumping power and trip were assumed. It was found that the uranium alloy-stainless steel eutectic would be formed if the delay is greater than 15-20 sec. Measured response times on the order of 1-2 sec have been observed.

The effectiveness of the low-flow trip was examined by setting the trip point at 94% of full flow and varying the delay time. It was found that delay times of up to about 6 sec could be allowed without leading to eutectic formation. The measured response time of the low-flow trips has been found to be approximately 3-3/4 sec, which would be sufficient to prevent eutectic formation. A further mode of protection would be supplied by the rate-of-flow trip, which would signal a trip when the flow is changing at a rate of 3%/sec or more. This mechanism has a measured response time of about 2-1/2 sec and would activate when the flow is approximately 97% of full flow.

It was concluded from these studies that there are sufficient and redundant protective levels to prevent reactor parameters from exceeding the protective restraints on the temperatures of coolant, cladding, and fuel.

C. Analyses of Irradiation Loadings

Analyses of various LOF incidents in cores more typical of recent irradiation loadings have also been made. These analyses accounted for the feedback effects in the same manner as the previously mentioned studies. The results are reported in Sec. VIII of Ref. 3. A wide variety of incidents were considered. Among them were: normal coastdown after loss of pumping power; instantaneous flow stoppage; auxiliary pump on; auxiliary pump off; one pump on and the other off; with trip protection; without trip protection. In each case, the reactor was assumed to be under full-power and full-flow

*The nominal trip point is 896°F. The time to reach that temperature would be significantly less than the times assumed for this study.

conditions at the beginning of the incident. A detailed analysis of each incident is presented in Ref. 3 and need not be reiterated in this report. However, one major conclusion of this study must be emphasized. The behavior of the oxide-fueled experimental subassemblies was found to be greatly different than that of the metal-fueled drivers. In each case, the peak centerline temperature of the oxide-fuel elements began an immediate decrease due to the inherent properties of oxide fuel. The initial behavior of the peak temperature in the metal-fueled drivers, however, was more dependent upon the assumed flow and trip-point conditions. The amount of auxiliary flow and the time delay between cessation of pumping power and reactor trip were especially important parameters in determining the behavior of the metal-fuel elements.

IV. FAILURE OF FUEL-ELEMENT CLADDING

A. Analyses of Effects of Flow Blockage in ANL-5719A

The next major category of safety considerations is that of failure of fuel-element cladding. Studies of flow blockage were originally made in Ref. 2 under the following assumptions:

1. Reactor is at steady-state, full-power operation.
2. Maximum centerline and surface temperatures of the fuel before flow blockage are 1197°F and 1060°F, respectively.
3. The blockage material has the thermal conductivity of Type 304 stainless steel.
4. Except for Case D, below, the blockage extends the entire axial length of the fuel element.

Five cases of flow blockage were considered, all based on the above assumptions:

- A. Blockage of one coolant-channel trifiute.
- B. Blockage around one element to a thickness equal to the diameter of the wire wrap.
- C. Blockage around part of one element to a thickness equal to the diameter of the wire wrap, but expanding to fill an entire trifiute channel for another part of the same element.

D. Blockage in the fuel-element grid, resulting in restriction of flow to the core section of the subassembly.

E. Same as Case A except that there are a number of blocked channels in the subassembly; no one fuel element, however, is in contact with more than one blocked channel.

The results of this study are shown in Table IV and are discussed below.

For Case A, the maximum fuel temperature would increase to 1213°F while the maximum fuel-surface temperature would rise to 1165°F. Neither of the temperatures presents any safety problems.

For Case B, angles of coverage of up to 150° yielded no temperatures of serious consequence. The maximum temperatures would be about the same as those for Case A.

Case C also would result in about the same maximum temperatures. Maximum fuel-centerline temperature would be 1261°F; maximum fuel-surface temperature would be 1173°F.

It was shown in Case D that with 50% of the flow area blocked (i.e., with about 85% of nominal flow rate), the fuel temperatures would increase less than 50°F. However, if the blockage should increase beyond this point, the uranium alloy-stainless steel eutectic may be formed.

The analysis of Case E showed that the fuel temperatures would increase very slowly as the number of blocked flow channels increased beyond one. Neither fuel melting nor eutectic formation would result from this incident.

In summary, this series of flow-blockage studies showed that no unduly high temperatures would be reached in the metal-fuel elements as long as each blockage was independent and no one element was in contact with more than one blocked channel.

B. Later Analyses Related to Failures of Fuel-element Cladding

A great deal of additional work has been done to determine the effect of flow blockage on failure of fuel-element cladding in EBR-II subassemblies.

TABLE IV. Effect of Flow Blockage on Fuel Temperatures,
as Analyzed in ANL-5719A

<u>Case No.</u>	<u>Effect</u>
A	Peak fuel temperature is 1213°F; peak fuel-surface temperature is 1165°F
B	Temperatures about same as those for Case A
C	Peak fuel temperature is 1261°F; maximum fuel-surface temperature is 1173°F
D	Fuel temperatures rise less than 50°F with 50% of the flow area plugged
E	Fuel temperatures only slightly higher than those for Case A

This work has been reported in the monthly Reactor Development Program Progress Reports and, in some cases, in various sections of Ref. 3. Most of the analyses reported below are concerned with oxide-fueled subassemblies, but flow blockages in driver and blanket subassemblies are also discussed, because they represent revisions or extensions of the original flow-blockage studies.

The behavior of Mark-IA, uranium-5 wt. % fission fuel elements subjected to flow blockages of various degrees and to one of two nuclear excursions has been studied.⁶ These excursions, described in Revision 3 of the Irradiations Guide,⁷ are: (1) insertion of one control rod into a delayed critical reactor at zero power; (2) insertion of one control rod into a reactor operating at full power. The blockage material was assumed to be fuel that had collected around the fuel element to a thickness equal to the diameter of the spacer wire (i.e., the same as Case B in the original studies, but with fuel as the blockage material). The angle of flow blockage was varied from 10 to 180°.

It was found that, for all angles of blockage considered, neither Excursion (1) nor Excursion (2) would cause the cladding temperature to reach the melting point. The highest cladding temperature for Excursion (1) was 1934°F, which would be reached 225 sec after the excursion began. The maximum cladding temperature for Excursion (2) was only 1336°F.

Flow-blockage Cases A, B, and D have been studied under the assumption that oxide fuel is in the flow channels.⁸ Power and flow conditions typical of those experienced by EBR-II experimental oxide-fueled subassemblies were assumed. For Case A, the peak cladding temperature would increase to ~2170°F, and some fuel melting would occur. For Case B, the melting point (~2600°F) of the Type 304 stainless steel cladding would be exceeded if the angle of flow blockage is greater than 100°. Case D was studied to determine the minimum coolant flow required for preventing coolant boiling. It was found that the flow rate could be reduced to ~30% of its nominal value before coolant boiling would be initiated.

Cases A and B resulted in cladding temperatures that, although they may or may not be above the melting point of the cladding, are nonetheless much higher than the nominal values. At these elevated temperatures, the tensile strength of the cladding is significantly less than at normal

operating temperatures. For example, the short-term tensile strength of unirradiated Type 304 stainless steel cladding at 2170°F is about 2000 psi. With this strength, a typical oxide-fuel element could contain an internal pressure of ~300 psi. If the element is at a sufficient burnup, the internal pressure due to fission-product gases could easily be 300 psi or more, and the cladding could burst under these conditions. The short-term tensile strength of the cladding under the conditions reached in Case D is ~6000 psi. An internal pressure of ~900 psi could, therefore, be contained in these elements. Reference 8 points out that the cladding of oxide-fuel elements could fail because of buildup of internal pressure at temperatures below the melting point of the cladding. An accurate analysis of this phenomenon should, however, account for the increase in yield strength due to irradiation of the cladding. This factor would allow greater pressures within the fuel element (i.e., the element could be taken to a higher burnup) before failure would be anticipated.

Analyses were done for a flow blockage in a blanket subassembly in row 7, 8, or 9.⁹ Both depleted uranium and stainless steel were considered as blocking materials. This study showed that very large angles of blockage (on the order of 240° or more) would be required to increase the cladding temperature above the uranium-stainless steel eutectic point.

Work has also been done¹⁰ on analyzing flow blockage resulting from a fission-gas release. It was found that under conditions typical for an oxide-fueled experimental subassembly, a blockage over an angle of 100° would be required before the melting point of the cladding would be reached. Calculations of the rate of cladding-temperature increase showed that the cladding temperature would rise to 2000°F in less than 2 sec and to an equilibrium temperature of ~2500°F in less than 20 sec. A high-burnup fuel element could be insulated with fission gas for 20 sec. Therefore, if the gas insulates a 100° angle of the cladding surface, the cladding could fail. As noted before, the decrease in cladding strength at the elevated temperatures, rather than the melting temperature, could be the cause of ultimate cladding failure.

Calculations were made^{11,12} that considered the case of a control rod being inserted into a reactor at full power, full flow, and containing an oxide-fuel element blanketed by fission gas. It was found that cladding

melting would not occur for angles of blanketing up to 80°. With an angle of 100°, however, cladding melting would be initiated after ~12 sec for a 0.290-in.-OD element and after ~30 sec for a 0.250-in.-OD element. Some fuel melting would take place in each case.

An experimental program¹³ is currently under way in which a number of oxide-fueled subassemblies will be irradiated until a cladding failure occurs in each. These tests will yield information on the method and consequences of failure of the cladding of an unencapsulated fuel element. The possibility of failure propagation to another fuel element is of principal interest in these tests.

A task force studying the deformation of a subassembly hexagonal can as a result of fission-gas release from an oxide-fuel element has recommended a 1000-psi limit on the plenum pressure of unencapsulated oxide-fuel elements in a subassembly without pressure-pulse protection when the subassembly is not adjacent to a safety or control rod and a limit of 450 psi when it is adjacent to a safety or control rod. Those limitations should remain in effect until experimental and/or analytical evidence is available to warrant an increase in these pressures.

V. OPERATION OF EBR-II WITH SIGNIFICANT OXIDE-FUEL CONTENT

Most of the oxide-fuel analyses reported in this document so far have been concerned with the safety considerations of a single oxide-fueled subassembly. Many studies have also been made dealing with the overall behavior of reactor loadings containing significant amounts of oxide fuel. Much of this work is discussed in Sec. VIII of Ref. 3.

Studies of mixed cores¹⁴ (i.e., cores containing both metal- and oxide-fueled subassemblies) showed that if the reactor, when at full flow and zero power, is subjected to an uncontrolled reactivity insertion of less than ~0.10\$/sec, oxide fuel would begin to melt before the metal fuel. For insertion rates higher than 0.10\$/sec, however, the metal fuel would begin to melt first. This behavior is primarily due to the greatly different thermal conductivities of the two types of fuels. As noted in Sec. II of this report, metal fuel would begin to melt first for all reactivity-insertion rates under reduced-flow conditions.

The effects of a control-rod insertion into a core containing a 50/50 mixture of oxide and metal fuel were also reported in Ref. 14. A parametric study of the feedback due to fuel expansion and coolant-density changes was made under the assumption that the reactor was at full-power and full-flow conditions at the beginning of the insertion. It was determined that fuel would begin to melt in the average oxide-fuel element after about 55 sec if coolant-density changes are the only source of reactivity feedback. If the feedback from both sources is considered, the time to the start of fuel melting is increased to about 70 sec. Either of these two time intervals should be sufficiently long for PPS action or administrative control to terminate the incident.

A parametric study using different values for the Doppler reactivity coefficient was also made for this core and reported in Ref. 14. This study clearly showed the importance of this parameter in allowing additional time before the initiation of fuel melting following a reactivity insertion. A similar study assuming the initial conditions of zero power and full flow yielded the same conclusion.

In the parametric studies cited in the preceding document, the Doppler coefficient reached a value of -0.001 , and in some of the studies cited below, it reached the same or a higher value. The values in typical EBR-II irradiation core loadings, however, are only on the order of -0.0002 . Reference 15 showed that the Doppler coefficient in EBR-II could be enhanced by the addition of sufficient amounts of beryllium oxide. For example, a Doppler coefficient of -0.00164 could be achieved in an oxide core that contained 16.69 vol. % of BeO. However, the neutron spectrum in such a core would be considerably softer than that in the same core without the BeO. For example, the median flux energy for the core without the BeO would be 395 keV whereas that for the core with the BeO would be only 155 keV. Such a softening of the spectrum may be undesirable for the LMFBR experimental-irradiation program.

The high-speed insertion of the central subassembly into a just-critical core with a 50/50 mixture of oxide and metal fuel and at reduced flow (3.3% of full flow) was also analyzed in Ref. 14. These conditions

are similar to those of Case 6 in the safety documents except that the core is under reduced-flow conditions (see Table II for these results). A Doppler reactivity coefficient of -0.0005 and a feedback equal to that from half a metal-fueled core (i.e., neglecting the feedback from the oxide-fuel elements) were assumed. It was found that the reactivity would reach a peak value of 96¢ before the prompt feedbacks would cause it to decrease. The peak metal-fuel temperature would be 1800°F , and the peak temperature of the metal-fuel cladding would be 1758°F after about 10 sec. The effects of the initial reduced-flow conditions are again seen (i.e., the peak oxide-fuel temperature would be only 2250°F , well below the melting point of the fuel). The behavior of the power level and the metal fuel in this core loading is very similar to that in the original, all-metal core. The study showed that the power peaked at 5.0 sec and that the peak reactivity was $0.0070 \Delta k/k$. These values compare favorably to the values of 4.5 sec and $0.0075 \Delta k/k$ obtained in the original study. Fuel melting would be initiated at 5.3 sec in the all-metal-fueled core as compared to about 10 sec in the mixed core.

The effects of dropping the central subassembly into a core with a 50/50 fuel mixture under the initial conditions of zero power and 3.3% of full flow were reported in Ref. 14. A Doppler coefficient of -0.002 and a fuel-expansion feedback equal to half that of an all-metal-fueled core were assumed. It was found that the reactor would be prompt critical for a short period of time before the reactivity excursion is reversed. It was also noted that, because of the rapid heating rate of the oxide fuel, the Doppler coefficient offered a greater degree of protection than axial expansion of metal fuel in this case.

The major conclusions from these studies are:

1. An increased loading of oxide-fueled subassemblies will not compromise the inherent safety over the operating range of EBR-II.
2. The point of initial fuel melting in a core containing both metal and oxide fuel is dependent on the reactivity-insertion rate, coolant-flow rate, and inherent physical properties of the fuel.
3. In a loading that is predominantly oxide fuel, the combination of a Doppler coefficient of -0.001 and the normal coolant-density changes would produce an overall dynamic behavior similar to that of a present loading of predominantly metal fuel.

4. The difference between the transient behaviors of oxide- and metal-fuel elements is due principally to their greatly different thermal conductivities.

Extensive analysis¹⁶ has been done in which not only the feedback due to fuel expansion but also that due to the collapse of molten fuel were taken into account. The feedback due to sodium-density changes or voiding was not included. It was shown that for unbounded reactivity ramps of about 1\$/sec, the melting and subsequent slumping of the metal fuel in a mixed core would protect the reactor from a major energy release. Additional work on a typical irradiation-core configuration showed that for unbounded ramps of 5-20\$/sec, the consequences are less severe than they would be in an all-metal-fueled core. Thus, the metal fuel in a mixed core serves as a protective device against major energy releases.

The steady-state thermal-hydraulics of several mixed core configurations have been studied.¹⁵ Operational limits and constraints that reflect the current thinking with regard to fuel melting, maximum cladding temperatures, maximum allowable coolant velocities, etc. were imposed on the oxide-fueled subassemblies in the core. Typical uncertainty factors for oxide-fuel elements were also used. It was found that when uncertainty factors were included, the maximum fuel temperature in a fully enriched oxide-fuel element would exceed the melting point of the fuel in several of the loadings considered. These loading patterns of fully enriched oxide-fueled subassemblies were considered infeasible until a technical basis is established for operating unencapsulated oxide-fuel elements with some centerline melting.

VI. MISCELLANEOUS ANALYSES

An analysis has been made¹⁷ of the temperature distribution within a fuel pin fabricated by infiltrating a matrix of UO_2 with PuO_2 spheres of a small diameter. This study was conducted to determine if, because nearly all the negative Doppler feedback is due to ^{238}U and nearly all the heat generation is in the plutonium, there is a significant lag between the power production and the Doppler feedback in a mixed-oxide fuel element when a control rod is inserted into a core operating under full-power and

full-flow conditions. The volume-weighted average temperatures of the UO_2 and the PuO_2 spheres were found to be the same; therefore, there was no significant delay in the Doppler feedback. It is believed that this conclusion applies to all mixed-oxide fuel elements that may be subjected to a nuclear excursion of similar magnitude.

Comparisons¹⁸ have been made between the calculated and measured temperatures in instrumented subassembly XX01. Good agreement was obtained between both the temperature of the coolant and the centerline temperatures of the fuel elements, thereby giving assurance that the computational methods used are applicable and accurate. Calculations also were made¹⁹ to determine the gap conductance of the UO_2 fuel elements in XX01. The conductance was found to be a function of linear heat rating and initial gap size. There was also evidence that the presence of xenon in the gap would decrease the gap conductance.

VII. CONCLUSIONS

The overall conclusion to be drawn from the many analyses reported here is that the performance and safety of the reactor with a core having a significant oxide-fuel inventory is not compromised relative to that of the original all-metal-fueled core considered in the safety documents. In fact, there is some evidence that the safety of a mixed core is somewhat improved over that of the original core loading.

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