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**SAFETY ASPECTS OF
ADVANCED FUELS IRRADIATIONS
IN EBR-II**

by

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TABLE OF CONTENTS

A.	INTRODUCTION.....	2
B.	SAFETY PHILOSOPHY.....	3
C.	TRANSIENT ANALYSES AND RESULTS.....	5
1.	Pin Models and Input Data.....	5
2.	Feedback Reactivity Effects of Advanced Fuels.....	7
3.	Response of Advanced Fuel to Postulated Transients..	8
4.	Effect of Shroud on Na Bonded Pin Performance.....	19
D.	LOSS OF Na BOND.....	21
E.	MOLTEN FUEL COOLANT INTERACTION.....	23
F.	FAILURE PROPAGATION.....	25
G.	CONCLUSIONS AND RECOMMENDATIONS.....	27
	REFERENCES.....	29
	APPENDIX "A"	

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1	Na BONDED CARBIDE S/A A RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT.....	9
2	He BONDED CARBIDE S/A B RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT.....	10
3.	He BONDED CARBIDE S/A C RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT.....	11
4	He BONDED NITRIDE S/A G RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT.....	12
5	Na BONDED NITRIDE S/A H RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT.....	13
6	COOLANT TEMPERATURE RESPONSE OF He BONDED CARBIDE ADVANCED FUEL PIN TO UNLIKELY EBR-II FLOW TRANSIENT.....	16
7	He BONDED CARBIDE S/A C RESPONSE TO ANTICIPATED LOSS-OF-FLOW TRANSIENT.....	17
8	He BONDED NITRIDE S/A G RESPONSE TO ANTICIPATED LOSS-OF-FLOW TRANSIENT.....	18
9	FUEL TEMPERATURE NECESSARY AT GIVEN SODIUM TEMPERATURE TO PRODUCE INTERFACE T = 2300°K BASED ON PROVISIONAL PROPERTIES..	26

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I	ADVANCED FUELS PHASE II - SAFETY ANALYSIS MATRIX.....	6
II	TIME RESPONSE OF ADVANCED PIN DESIGNS TO DESIGN BASIS REACTIVITY TRANSIENT.....	20

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ABSTRACT

Basic safety questions such as MFCL, loss-of-Na bond, pin behavior during design basis transients, failure propagation, etc., were evaluated as they pertain to advanced fuels in EBR-II. With the exception of pin response to the unlikely loss-of-flow transient, the study indicates that irradiation of significant numbers of advanced fueled subassemblies in EBR-II should pose no safety problems. The analysis predicts, however, that Na boiling may occur during the postulated design basis unlikely loss-of-flow transient in subassemblies containing He-bonded fuel pins with the larger fuel-clad gaps. The calculations indicate that coolant temperatures at top of core in the limiting S/A's, containing He bonded pins, would reach $\approx 1480^{\circ}\text{F}$ during the transient without application of uncertainty factors. Inclusion of uncertainties could result in temperature predictions which approach coolant boiling temperatures (1640°F). Further analysis of He-bonded pins is being done in this potential problem area, e.g., to apply best estimates of uncertainty factors and to determine the sensitivity of the preliminary results to gap conductance.

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A. Introduction

As part of the national Advanced Fuel Development Program, approximately 20 advanced fuel subassemblies are to be irradiated in EBR-II over the next several years. These subassemblies will contain both He and Na bonded mixed carbide and nitride fuel pins with varying pin parameters and are to be irradiated at a variety of linear power levels and maximum clad temperatures. The primary objectives of the fuel development program are: (1) to provide a data base on fuel element performance for future FTR irradiations, and (2) to provide specific information on materials, design, and fuel behavior for the development of a fuel type for commercial LMFBR's. The irradiation program is to be conducted in several phases. Details of the Phase I, IA, and II irradiations are given in Reference 1. Phase I irradiations are to begin in the third quarter of CY75 and the first of the Phase II subassemblies are scheduled for insertion in mid-76. It is conceivable that as many as eighteen advanced fuel subassemblies could be in the reactor at any one time.

The purpose of this report is to outline the specific safety analyses that have been done to support the irradiations and to address any potential safety problems related to specific fuel type or number of subassemblies contained in-core. In some cases, a detailed safety

analysis cannot be done due to the lack of irradiation data and fuel performance information under steady state or transient conditions. In these instances, prior to operation, an assessment of operational risk must be conducted and a determination made whether this risk is acceptable in view of its safety and programmatic impact. Additionally, limits on proposed operations are suggested which are subject to revision pending additional analysis and/or fuels behavior data.

B. Safety Philosophy

The safety philosophy adopted for EBR-II operation requires that the consequences of an unlikely event be restricted to those of a minor incident, i.e., limited damage to replaceable components.⁽²⁾ Operation with standard fuel and experiments is restricted to preclude consequences in excess of the above. In the case of advanced fuels, similar restrictions apply. Since the consequences of pin rupture and accident progression following failure are unknown, pin design and operating conditions should be chosen to preclude significant failures for off-normal conditions associated with identified faults. The most likely initiator of local failure in advanced fuels irradiated in EBR-II, is coolant boiling ($\sim 1640^{\circ}\text{F}$). Therefore, consistent with the established safety philosophy, advanced fuel operating conditions should be chosen to preclude coolant boiling in the event of an unlikely fault.

Other failure mechanisms may be hypothesized. Some of these are: (1) loss of Na bond, (2) clad defects, (3) coolant blockage, (4) pin overpressure due to rapid internal fission gas release,

(5) fuel-clad mechanical interactions, and (6) fuel pin contact within subassemblies with resulting flow starvation. The above faults including coolant boiling may be hypothesized which in turn lead to molten-fuel-coolant-interactions (MFCI), pin-to-pin failure propagation, and possibly element-to-element failure propagation. Since the aim of the fuel development program is to provide information on failure mechanisms and consequences, it is impossible at this time to provide a detailed quantitative evaluation of the safety implications of all aspects of advanced fuel irradiation in EBR-II. Where the hypothesized failure mechanisms and/or consequences cannot be quantitatively evaluated, conservative limits to design and/or operation are suggested to ensure that the consequence of such events would have no serious impact detrimental to safety.

In the event of an anticipated fault which results in a power-to-flow ratio in excess of 1.31, criteria developed in the proposed Technical Specifications⁽³⁾ require that the performance of all subassemblies be evaluated to provide assurance of safety. In the case of the advanced fuel pin design, the criteria for subsequent safe operation are not fully developed; however, if temperatures are limited to at or near their steady state values during a transient, and are several hundred degrees below those reached during the unlikely design basis loss-of-flow transient, no safety problem need be anticipated.

Limiting anticipated and unlikely faults have been defined for EBR-II and provide the design basis transients for analysis of reactor

operations and experimental subassembly irradiations. These transients are:

1. Unlikely fault - Single pump seizure with simultaneous coastdown of the other pump. Shutdown reactivity 3.7\$, trip-rod drop time 0.45 sec and feedback reduced to proposed Technical Specifications limits.⁽³⁾
2. Unlikely fault - Insertion of two high-worth control rods at power. Feedback reduced to proposed Technical Specification limits.⁽³⁾
3. Anticipated fault - Loss of primary pumping power to 5% flow. Shutdown reactivity 3.7\$, trip-rod drop time 0.45 sec and feedback reduced to the Technical Specification limits.⁽³⁾

Analysis of the response of the proposed advanced fuel designs to these transients provides the basis for safety during transient over-power conditions and loss-of-flow faults.

C. Transient Analyses and Results

1. Pin Models and Input Data

Calculations of the response of the proposed advanced fuel designs included in the Phase II Test Matrix⁽¹⁾ to the unlikely and anticipated design basis transients have been completed. The pins chosen for analysis were those whose operating conditions and designs were most limiting from a thermal hydraulics standpoint. These pins are listed in Table 1 and are contained within subassemblies designated as A, B, C, G, and H in Reference 1. The calculations were done with

TABLE I. Advanced Fuels Phase II - Safety Analysis Matrix

<u>Subassembly No.</u>	<u>kW/ft</u>	<u>Max Clad Temp - ID</u>	<u>Type</u>	<u>Density</u>	<u>Radial Gap</u>	<u>Pin Dimensions - In.</u>		
						<u>Wall</u>	<u>Pellet</u>	<u>Pin</u>
A	30	1200 ⁰ F	Na-C	98	0.018	0.015	0.304	0.37
B	30	1200	He-C	81	0.0055	0.020	0.319	0.37
C	19	1240	He-C	81	0.0055	0.020	0.319	0.37
G	30	1265	He-N	87	0.0055	0.020	0.319	0.37
H	30	1210	Na-N	95	0.012	0.015	0.316	0.37

the EROS/EROTIC code⁽⁴⁾ which is used and accepted for dynamic simulation of EBR-II. EROS solves the space-independent reactor kinetics equations and computes the temperature induced feedback reactivity to define the resulting transient. This transient is then used to predict the thermal behavior of selected pin designs in EROTIC. EROTIC allows up to 80 nodes to define the pin geometry and composition.

Fuel properties (thermal conductivities, specific heats, etc.) were the most recent recommended values as reported in Reference 5 and 6.

2. Feedback Reactivity Effects of Advanced Fuels

Safe operation of EBR-II with metal fuel depends on negative feedback reactivity primarily due to expansion of the sodium and fuel.

Operation with significant numbers of advanced fueled subassemblies, may result in reduced feedback due to the potential absence of the fuel expansion component in carbides and nitrides. Calculations⁽⁷⁾ of the sodium component of the EBR-II prompt power coefficient indicate that approximately 70% of the total is due to sodium expansion and the remainder is due to fuel expansion. These findings were supported by a special reactor test, EX-77,⁽⁸⁾ in which feedback reactivities were measured at constant and varying power-to-flow ratios. This was a test in which the full effect of sodium expansion was retained and the fuel component systematically reduced. From these data, it is then possible to determine the relative magnitudes of the fuel and sodium components.⁽⁹⁾

For the transient analysis reported here, the feedback was reduced to the proposed Technical Specification limit.⁽³⁾ This value is $-0.0017\$/\text{MWt}$ ($.49\text{lh}/\text{MWt}$) and is less than the magnitude of the sodium expansion effect alone. Therefore, demonstration of safe operation at the Technical Specification limit precludes any concern regarding the reduced feedback effects of the advanced fuels in EBR-II. If feedback were in fact to drop below the above limiting value, operation of EBR-II would not be allowed.

3. Response of Advanced Fuel to Postulated Transients

a. Loss of Flow Transient - Unlikely Fault

The results of the calculations of the temperature response of the S/A's listed in Table 1 to the EBR-II unlikely flow transient are shown in Figures 1 through 5. The figures show four curves with the uppermost being the fuel centerline temperature, next the fuel surface temperature, followed by the average clad temperature and the average bulk coolant temperature. Temperatures shown are at the top of the core where the clad and coolant temperatures are maximum values. The calculations indicate that the He bonded pins operating with high clad temperatures will be limiting, and that operation of subassemblies C and G at the stated conditions result in the highest coolant temperatures at the core top for the unlikely flow transient. Fuel temperatures show a nominal increase prior to reactor scram; however, they do not approach melting temperatures.

During the transient, Na bonded fuel centerline temperatures also increase initially and then rapidly decrease following scram and at no time do fuel temperatures approach melting.

NA BONDED CARBIDE S/A A
RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT

Figure 1

HE BONDED CARBIDE S/A B
RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT
Figure 2

HE BONDED CARBIDE S/A C
RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT

Figure 3

HE BONDED NITRIDE S/A G
RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT

Figure 4

NA BONDED NITRIDE S/A H
RESPONSE TO UNLIKELY LOSS-OF-FLOW TRANSIENT
Figure 5

The results reported here are based on calculations of the nominal temperature rise during the transient and do not include any uncertainty factors. Inclusion of uncertainties will result in higher values of fuel pin and coolant temperatures and may result in temperature predictions which approach a boiling temperature (1640°F). A detailed evaluation of the uncertainties for advanced fuel subassemblies is susceptible to great error at this time, because of the early stage of development of such fuel. Applying *uncertainties used in evaluation of EBR-II metal driver fuel performance*, however, would indicate peak coolant temperatures about 120°F higher than nominal, implying the possibility of boiling in advanced fuel subassemblies with their larger values of uncertainty, particularly in that of gap conductance.

These analyses were done for the beginning-of-life fuel pins and do not take into account the effects of fuel swelling and cracking. It is expected that susceptibility to boiling during the transient would occur for a short period at the start of the irradiation and that at some point sufficient swelling would occur, improving gap conductance, and thus eliminating the possibility of boiling. Recent data⁽¹⁰⁾ suggest this may occur within the first several hundred hours of the irradiation period.

It should be emphasized that the susceptibility of any advanced fueled subassembly to boiling during a transient is a strong function of the design and operating conditions. For example, the gap conductance has a strong influence on the peak temperatures attained under transient conditions. Calculations have been done which show

that coolant boiling would occur during the unlikely loss-of-flow transient for He-bonded pins with radial gap widths of twice the values used in this study. Inclusion of uncertainties would predict boiling to occur earlier in the transient and involve more of the subassembly.

A series of calculations were also done to determine the effect of different operating conditions on the peak temperatures attained during the unlikely flow transient. The results are shown in Figure 6 as a function of the nominal power-to-flow ratio. The pins studied are typical of He bonded carbide advanced fuel designs and are identical to subassembly C of Table 1. Also shown on the figure, are the results obtained for S/A's B and G. The higher peak temperature reached in S/A G at the same power-to-flow ratio is attributed to different fuel type (N vs C) and pin parameters. As shown, the peak coolant temperature reached during a transient is strictly a function of the power-to-flow ratio for any pin type. These results suggest that the subassembly operating parameters, i.e., power-to-flow ratio could be controlled to preclude coolant boiling during the design basis transient.

b. Loss of Flow Transient - Anticipated Fault

The response of the most limiting pin designs (He bonded pins S/A's C and G) to the anticipated loss-of-flow event is shown in Figures 7 and 8. For this transient, the peak coolant temperatures are well below boiling and more than 250°F below those reached in the unlikely loss-of-flow event.

FIGURE 6
COOLANT TEMPERATURE RESPONSE OF HE-BONDED CARBIDE ADVANCED
FUFL PIN TO UNLIKELY EBR-11 FLOW TRANSIENT

HE BONDED CARBIDE S/A C
RESPONSE TO ANTICIPATED LOSS-OF-FLOW TRANSIENT

Figure 7

HE BONDED NITRIDE S/A G

RESPONSE TO ANTICIPATED LOSS-OF-FLOW TRANSIENT

Figure 8

c. Reactivity Insertion

The response of the subassemblies selected for the safety analysis matrix to the unlikely reactivity insertion event of two high worth control rods was calculated to determine the time required to approach coolant boiling. These times are then compared to the PPS response time to determine if the transient is turned over prior to reaching conditions where damage may occur. The results are shown in Table II. The most limiting pin from the standpoint of coolant boiling is again the He bonded carbide S/A - C running at the highest power-to-flow ratio. Considering time to fuel melting, S/A - B is limiting because of the higher (30 kw/ft) linear power density. There is no indication that bond boiling or vaporization would occur in the Na bonded pins prior to termination.

It is expected that, during this transient, fuel slumping and/or melting may occur for several of the pin designs prior to termination. The behavior of these pins during the transient and during subsequent irradiation is unknown, although the fuel condition and expected behavior is very much like that which occurs during a loss-of-Na bond event and the same consequences are expected. The loss-of-bond fault is discussed in Section D.

4. Effect of Shroud on Na Bonded Pin Performance

Calculations of the effects of the shroud on Na bonded fuel pin performance during the design basis transient were also done. The shroud was assumed to be 3 mils thick with 50% of its surface being perforated. The heat transfer properties are then similar to a 50-50 mix

TABLE II
Time Response of Advanced Pin Designs
to
Design Basis Reactivity Transient

<u>Subassembly No.</u>	<u>Type</u>	<u>Time to Na Boiling</u>	<u>Time to Fuel Melting* (Fuel Surface at hot spot)</u>	<u>Time for PPS Action</u>
A	Na-C	27.5 sec	>30 sec	5.4 sec
B	He-C	>30.0 sec	18.0 sec	5.4 sec
C	He-C	27.0 sec	25.0 sec	5.4 sec
G	He-N	28.0 sec	Dissociation Temp = 2780°C 24.0 sec	5.4 sec
H	Na-N	27.5 sec	>30.0 sec	5.4 sec

*Carbide melting temperature was assumed to be 4125°F (Solidus Temperature)

of stainless and sodium.

The calculations show a slightly ($\sim 5^{\circ}\text{F}$) higher temperature rise for the shrouded pin and consequently the thermal performance can be considered unchanged for purposes of safety analysis.

5. End of Life Conditions

End of life conditions for He bonded pins (S/A - C) were modeled by assuming that the fuel expanded into the available space inside the clad and closed the gap. Thermal conductivity values were reduced to account for the increased porosity by⁽¹¹⁾

$$k = k_0 \frac{1 - P}{1 + \beta P}$$

where P is the porosity and $\beta = 0.5$.

The calculations of the nominal peak temperature during the EBR-II unlikely design basis transient are about 250°F lower than for the beginning of life fuel pin which has the lower gap conductance.

D. Loss of Na Bond

Loss of Na bond in the advanced fuel pin designs can be either partial or complete depending on the nature of the initiating mechanism. Partial bond defects may be inherent, e.g., due to manufacturing defects, or may develop during the course of the irradiation. Both analysis and experiments indicate that small bond defects will not cause any safety concern.⁽¹²⁾

Gross bond voiding may be hypothesized to result from bond expulsion due to rapid fission gas release, bond vaporization during

overpower, or manufacturing defects which may develop during irradiation. Large bond defects may or may not result in fuel melting or slumping. Should the fuel melt and/or slump and contact the clad, it is expected to solidify and cool upon contact with no adverse consequences.⁽¹²⁾ The same reasoning would apply to the case where fuel melting would occur during a steady state or transient overpower condition.

Calculations of the time to onset of fuel melting following loss of Na bond have also been done as part of this investigation. The pin studied was a Na bonded carbide pin operating at 30 kw/ft with a clad surface temperature of 1300°F. The Na bond was modeled as a coolant region in an EROS calculation, and bond loss was simulated by a rapid reduction (to 10%) in the heat transfer coefficient at the fuel-bond interface. The results indicate that fuel melting would be at 4.4 seconds following loss of bond. This value is consistent with results reported in Reference (12) for very similar fuel pins. There the analysis showed the fuel to reach melting at 4.75 sec. Since both analyses and experiments indicate that loss of bond will not result in further clad deterioration, irradiation of advanced fuels with bond defects in EBR-II should not cause any safety problems.

The continued irradiation behavior of sodium bonded pins with partial or gross bond defects is unknown at this time. Questions remain as to the existence of fuel-clad interactions and the general behavior of advanced fuels under these conditions. It is expected that interim examination will provide sufficient data and the safety assurance to continue the irradiations to higher burnups.

E. Molten Fuel Coolant Interaction

The potential for severe vapor explosions (MFCI) in liquid metal systems has been hypothesized, and considerable work has been done to define the potential for the event and its consequences, particularly on oxide fueled systems. Because of the higher thermal conductivity and density coupled with the potential for attaining higher temperatures during offset conditions, advanced fuels exhibit a greater potential for MFCI's than do the oxides.^(13, 14) It has been suggested and verified in controlled laboratory experiments that a necessary but not sufficient condition for occurrence of an MFCI is that the interface temperature must exceed the spontaneous nucleation temperature ($\sim 2300^{\circ}\text{K}$) of the cold fluid. The relationship between the hot fluid and cold fluid in terms of the system physical properties and the interface temperature is given by

$$\frac{T_h - T_i}{T_i - T_c} = \sqrt{\frac{(\rho K C_p)_c}{(\rho K C_p)_h}}$$

where T is the absolute temperature

ρ is the density

K is the thermal conductivity

C_p is the specific heat

and the subscripts refer to the hot and cold fluids and the interface temperature.

c (subscript) indicates coolant

h (subscript) indicates fuel

The relationship predicted by the above equation between the fuel and coolant temperatures for an interface temperature of 2300°K is shown in Figure 9 taken from Reference (15). The results were derived by fitting linear functions of temperature to values of $\sqrt{\rho K C_p}$ and solving for fuel temperatures required to initiate vapor explosions at the stated coolant conditions. The most limiting advanced fuel type is the 100% dense carbide and at Na boiling temperatures, the fuel temperature would have to be $>200^{\circ}\text{F}$ above its melting point to initiate an MFCI assuming other conditions are favorable. This coupled with the fact that molten fuel would probably not penetrate the clad makes the occurrence of an MFCI extremely unlikely in EBR-II utilizing advanced fuels.

It is considered important, however, to limit the possibility of significant clad failure under conditions when the possibility also exists to melt fuel, i.e., operating conditions should be chosen to preclude boiling in the event of the occurrence of the unlikely loss-of-flow accident.

In order to achieve conditions conducive to a violent vapor explosion in EBR-II with advanced fuels, accidents more severe than the unlikely events considered here would have to occur. Events with severity levels comparable to the DBA would be required to generate fuel and/or coolant temperatures in the region where vapor explosions could occur, according to the above criteria. At this point, the consequences of other aspects of the accident would be more severe, e.g. thermal energy released and coolant flashing, and the MFCI would assume a secondary role for consideration.

F. Failure Propagation

Pin-to-pin and the more severe S/A-to-S/A failure propagation faults for oxide fuels has been the subject of intense research and analysis. However, very little work has been directed to the advanced fuels. Proposed failure mechanisms, such as blanketing due to fission gas release and fuel melting and slumping are affected by gas release rates, thermal conductivities and design characteristics, which all vary depending on the fuel type.

Work to date shows that gas release rates for advanced fuels are less than those for oxide fuels. An unknown is the release rate during transient conditions, although test UL-4⁽¹⁶⁾ suggests that this should pose no problems. The Guide for Irradiation Experiments in EBR-II places restrictions on the location of experiments depending on the internal pressure. It also requires that this pressure be calculated at operating temperatures with the assumption that rates of fission gas release be equivalent to those of oxides up to 17 kW/ft, and above this a 100% release rate is to be assumed. Plenum pressures calculated in this manner should bracket the expected peak pressures to be attained in advanced fuels and provide an adequate safety margin.

Other aspects of failure propagation are still under study and complete assurance of safety depends on the accumulation and analysis of the irradiation and fuel behavior data.

The irradiation vehicle for the advanced fuel pins is a sub-assembly which is very similar to present designs incorporating by-pass flow, except there is no provision for the insulating gas. The proposed

design and thermal analysis is discussed and contained in Appendix A. The design is only preliminary and final analysis of the steady state and off-normal conditions is to be done.

G. Conclusions and Recommendations

Analysis of the behavior of the advanced fuel pin designs at the proposed operating conditions has shown that coolant boiling may occur during the design basis unlikely loss-of-flow transient for two of the proposed designs. The limiting pin designs are the He-bonded carbides and nitrides operating at the higher power-to-flow ratios. In the case of the He-bonded carbide fuel pin operating at 19 kW/ft and a maximum clad temperature of 1240°F, this initial analysis shows coolant temperatures of 1480°F at the core top during the unlikely loss-of-flow transient, without uncertainty factors. Application of calculational uncertainties could result in temperature predictions very close to the point of coolant boiling.

The most likely initiator of clad failure is coolant boiling, and the consequences of clad failure are unknown at present. Thus, in the event that temperature predictions, with uncertainties included, approach coolant boiling temperatures, design and operating conditions of the He-bonded pins with the larger fuel-clad gap should be critically reexamined and readjusted, within the constraints of the programmatic requirements, to minimize the potential for coolant boiling.

Preliminary investigation of other safety aspects of advanced fuels indicate that irradiation of significant numbers of experimental subassemblies in EBR-II should pose no safety problems. Requirements of the Guide for Irradiation Experiments in EBR-II are explicit and sufficient to

assure safety with respect to protection of the reactor from pin overpressure and subassembly malfunction. Other safety questions such as MFCI, fuel melting and/or slumping, loss of Na bond and PPS response to mitigate adverse pin and coolant conditions for reactivity insertion events do not appear to pose any concern for safe operation when considered for advanced fuels in general. However, specific advanced fuel designs (including operating conditions) may generate safety-related questions yet to be fully resolved. For these, an assessment of risk will be conducted and a determination made as to whether the risk is acceptable in view of the programmatic benefits to be derived from the specific irradiations. The interim criteria, which are conservative relative to safety, may be relaxed as such information is generated.

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APPENDIX A

PRELIMINARY DESIGN DESCRIPTION
NEW BYPASS FLOW SUBASSEMBLY MARK U-19

For
ADVANCED FUELS PROGRAM

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June 6, 1975

Introduction

At the request of the Manager of the Advanced Fuels Program, a preliminary subassembly design has been prepared having an uninsulated internal bypass flow, accepting 0.370 diameter fuel elements of 40 inch nominal length, an 0.067 inch spacing wire, a resultant pitch-to-diameter ratio of 1.181, an 1165°F mixed-mean fuel-pin-bundle coolant exit temperature, and a mixed-mean subassembly coolant outlet temperature (1000°F for row 4) within the $\pm 80^\circ\text{F}$ criterion of the adjacent driver fuel subassemblies. This writing addresses preliminary design details based on a 32 kW/ft heat generation and provides calculated flow requirements and temperature distributions considering horizontal heat transfer in this and its adjacent subassemblies.

The vertical elevations in this proposed subassembly are the same as the existing Mark A-19B subassembly previously applied to this program, and about as used in the existing Mark J-19B bypass flow subassembly depicted in Fig. 1. This new subassembly omits the thermal insulation between the fuel-bundle flow and the bypass because of a lack of diametral space dictated by the size of the outer hexagonal tube of all the EBR-II subassemblies.

Subassembly Description

The proposed subassembly is similar to present designs of bypass flow subassemblies, except that it does not have an insulating gas space between the fuel-bundle flow and the bypass flow. The insulating gas space is omitted to provide space for the installation of 19 test elements of the desired 0.370 diameter.

Figure 1 is an example of an existing design (a Mark J-19B) having bypass flow similarities. Coolant sodium flow enters the lower adapter through the standard orifice holes, passes up through the radiation shield and into the upper chamber of the shield. At this point the flow is divided by separate orifices, into bypass flow and capsule flow. The bypass flow passes through six radial orifice holes in the upper chamber of the radiation shield. The orifice holes are sized to give a bypass flow which when mixed with the hotter capsule flow will provide an exit temperature about equal to that of the adjacent subassemblies. The bypass flow, after passing through the bypass orifices, flows through the annular space between the outer hex liner tube and the outer hex tube to the top of the preassembly and into the mixer section. The flow that does not enter the bypass orifices is directed through an orifice plate at the radiation shield into the capsule region. The seven holes in the orifice plate are sized to give correct capsule flow to maintain desired fuel capsule clad temperature. After passing through the test capsule section, the capsule flow enters the mixing section and is mixed with the cooler bypass flow. There is an insulating gas space between the inner hex liner tube and the outer hex liner tube to prevent heat transfer from the fuel-bundle flow to the bypass flow.

The cross-section of the proposed U-19 bypass flow irradiation subassembly is shown in Fig. 2. There is no insulating gas space between the fuel-bundle flow and the bypass flow. This was achieved by eliminating the inner hex liner tube. The element diameter is 0.370 in. dia, with an 0.067 in. spacer wire, resulting in a pitch-to-diameter ratio (P/D) of 1.181. There are 12 filler strips on the inside wall of the hexagonal liner tube. The bypass flow flows in the 0.059 in. annulus between the hexagonal liner tube and the outer hex tube.

Thermal Analysis

The required thermal characteristics are summarized in Table I. The subassembly is to have an outlet temperature within 80°F of the adjacent subassemblies. For a row 4 position in EBR-II, this means the max. subassembly outlet temperature is 1000°F , as the adjacent subassemblies have an average outlet temperature of 920°F .

FIG. 1. EBR-II MARK J-19B IRRADIATION SUBASSEMBLY

FIG. 2
U-19 IRRADIATION SUBASSEMBLY
CONCEPTUAL DESIGN

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TABLE I. Thermal Characteristics of Proposed Subassembly U-19

Subassembly Inlet Temperature (T_1)	700°F
Average Fuel Bundle Exit Temperature (T_2 - At Top of Core)	1165°F
Subassembly Exit Temperature (T_4)	1000°F
Fuel Pin Heat Rate	32 kW/ft
Total Heat Generated in Core Section	2.336×10^6 BTU/hr
Adjacent S/A Inlet Temperature (T_1)	700°F
Adjacent S/A Outlet Temperature (T_5)	920°F

A computer program was written to determine the required fuel bundle and bypass flowrates. This program took into account the heat transfer from the fuel-bundle flow to the bypass flow, and the heat transfer from the adjacent subassemblies to the bypass flow. The equations developed in Reference 1 were used to determine the temperatures within the subassembly. Figure 3 shows a schematic of the subassembly flow path and the internal temperatures.

Fig. 3. Schematic Diagram of Proposed Subassembly

In the heat transfer analysis, the subassembly was divided into two regions:

- Section I: Heat Generating (Core) Section
- Section II: Equalizing Section

The equalizing section extends from the top of the core to the mixer.

The thermal characteristics calculated by the computer program are summarized in Table II.

TABLE II. Calculated Thermal Characteristics
of Proposed Subassembly

Total S/A Flowrate (W)	63.1 gpm
Fuel Bundle Flowrate (W_1)	33.9 gpm
Bypass Flowrate (W_2)	29.2 gpm
T_3 (Bypass flow temp. at top of core)	801°F
T_7 (Bypass flow temp. at top of equalizing section)	946°F
T_6 (Fuel bundle flow temp. at top of equalizing section)	1048°F
Heat Transferred from Fuel Bundle to Bypass Flow in Core Height	0.2971×10^6 BTU/hr
Heat Transferred from Adjacent Subassemblies to Bypass Flow (over core heights I and II)	0.1036×10^6 BTU/hr

The sodium heat transfer coefficients were calculated by the Monson equation:

$$Nu = 2.3 + 0.23 Pe^{.5},$$

$$\text{where } Nu = \frac{h D_e}{k} ; \quad Pe = \frac{\rho C_p v D_e}{k}$$

The overall heat transfer coefficient between the fuel-bundle flow and the bypass flow was calculated to be 2502 BTU/hr ft² °F. The overall heat transfer coefficient from the flow in the adjacent subassembly to the bypass flow was calculated to be 1098 BTU/hr ft² °F. These values were used as input to the computer program.

Estimate of Max. Temperature Drop Across Hex Liner Tube

There is a temperature drop across the hex liner tube because of the temperature difference between the fuel-bundle flow and the bypass flow. The largest temperature drop across the hex liner tube will occur at the top of the core section. The temperature difference between the average fuel bundle temperature ($T_2 = 1165^\circ\text{F}$) and the bypass flow temperature at this point ($T_3 = 801^\circ\text{F}$) is 364°F . However, the bundle flow temperature adjacent to the hex liner tube is much lower than 1165°F , due to the heat transfer from the outer channels of the fuel-bundle flow to the bypass flow.

The fuel-bundle flow temperature in the outer channels was estimated by making the following assumptions:

1. The flowrate in the outer channels to the total fuel bundle flowrate is in the same ratio as the outer channel flow area to the total bundle flow area. The outer channel flow area with filler strips was calculated to be 31% of the total bundle flow area, so the flowrate in the outer channels was estimated to be 10.5 gpm (1.24 lb/sec).
2. No mixing occurs between the outer flow channels and the inner flow channels.
3. All of the heat transferred from the fuel-bundle flow to the bypass flow over the core section results in lowering of the outer channel flow temperature. The heat transferred is 0.297×10^6 BTU/hr, or 82.53 BTU/sec.

The sodium temperature in the outer channels is then:

$$\begin{aligned} T &= 1165^\circ\text{F} - \frac{(82.53 \text{ BTU/sec})}{(1.24 \text{ lb/sec}) (.3041 \text{ BTU/lb}^\circ\text{F})} \\ &= 1165^\circ\text{F} - 218^\circ\text{F} \\ &= 947^\circ\text{F}. \end{aligned}$$

Hence, the maximum temperature drop across the hex liner tube is estimated to be 146°F (947°F - 801°F).

The thermal stress resulting from this temperature drop is:

$$\begin{aligned}\sigma_t &= \frac{E\alpha\Delta T}{2(1-\mu)} \\ &= \frac{(23.5 \times 10^6) (9.8 \times 10^{-6}) (146)}{(2) (1-.3)} \\ &= 24,017 \text{ psi.}\end{aligned}$$

This is lower than the yield strength of cold-worked 316 SS at the subassembly operating temperature.

Reference

1. EBR-II Technical Memorandum No. 71, Design and Pressure Drop Characteristics of the EBR-II Irradiation Subassembly Mark C, May 29, 1967.