**SURVEILLANCE AND EVALUATION OF THE
EBR-II FLOW-MONITORING SYSTEM**

by

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TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT.	9
I. INTRODUCTION.	11
II. SCRAM HISTORY OF THE FLOW-MONITORING SYSTEM	13
III. ANALYSIS OF FAILURE OF POWER TO PRIMARY PUMPS	15
A. Uncertainty Factors.	24
B. Comparison with Other Independent Studies.	24
IV. FLOW MONITORING DURING LOW-FLOW EXPERIMENTS	25
V. FLOW PERTURBATIONS.	26
VI. CONCLUSIONS	33
APPENDIX: Physical Properties of Some Reactor Materials.	44
REFERENCES.	46

LIST OF FIGURES

<u>No.</u>	<u>Title</u>	<u>Page</u>
1.	EBR-II Primary Cooling System.	12
2.	Flow Decay in EBR-II	17
3.	Power Decay in EBR-II.	18
4.	Fuel-centerline Temperature Transient (62.5-MWt Operation) . .	19
5.	Cladding Temperature Transient (62.5-MWt Operation).	20
6.	Temperature Transient of Channel Sodium (62.5-MWt Operation) .	21
7.	Initial Temperature Transients (62.5-MWt Operation).	22
8.	Pump No. 2: High-pressure Plenum Flow, 0516 hr, March 8, 1968	27
9.	Pump No. 1: High-pressure Plenum Flow, 0516 hr, March 8, 1968	28
10.	Pump No. 1: rpm, 0516 hr, March 8, 1968 :	29
11.	Pump No. 2: rpm, 0516 hr, March 8, 1968	30
12.	Total Reactor Flow, 0516 hr, March 8, 1968	31
13.	Reactor-blanket Flow, %, 0516 hr, March 8, 1968.	32
14.	Pump No. 1: High-pressure Plenum Flow, 1425 hr, Nov. 4, 1968.	34
15.	Pump No. 2: High-pressure Plenum Flow, 1425 hr, Nov. 4, 1968.	35
16.	Total Reactor Flow, 1425 hr, Nov. 4, 1968.	36
17.	Reactor-blanket Flow, %, 1425 hr, Nov. 4, 1968	37
18.	Pump No. 1: High-pressure Plenum Flow, 1510 hr, Nov. 4, 1968.	38
19.	Pump No. 2: High-pressure Plenum Flow, 1510 hr, Nov. 4, 1968.	39
20.	Total Reactor Flow, 1510 hr, Nov. 4, 1968.	40
21.	Reactor-blanket Flow, %, 1510 hr, Nov. 4, 1968	41

LIST OF TABLES

<u>No.</u>	<u>Title</u>	<u>Page</u>
I.	Statistics of Flow-monitor Scrams.	14
II.	Effect of Uncertainty Factors on Calculations of Temperature.	24

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ABSTRACT

Because the many flow monitors active in the EBR-II shutdown circuit have been a continuing source of spurious trips, a study was begun to determine whether the flow-monitoring system could be simplified without compromising reactor safety. A simultaneous failure of power to both primary pumps was hypothesized, and the resulting temperature transient in the reactor was simulated by using the THTB code in a computation by IBM-360 computer. Certain penalties were imposed on the computation to ensure conservative results.

The results of the THTB calculation indicate a temperature margin of 50°F (below the 1300°F level at which the eutectic between the fuel and cladding begins to form) for a case in which the reactor scram was delayed 4 sec after power failure to the pumps. More realistic assumptions about reactivity feedback effects and initial conditions would result in a wider temperature margin. Two conclusions that follow are: (1) the reactor can be protected adequately from loss of coolant flow by redundant fixed-setpoint flow monitors; and (2) the monitors that measure the rate of change of flow and the monitors on the pump auxiliary functions can be deleted safely from the reactor shutdown circuit.

I. INTRODUCTION

The shutdown circuit of the EBR-II contains monitors for coolant flow. These monitors, to assure reactor safety, cause reactor scrams if coolant flow fails. However, they have continuously caused spurious trips in the circuit. Therefore, a study was made to determine whether the flow-monitoring system could be simplified without compromising reactor safety. The entire primary cooling system of EBR-II (Fig. 1) is described in Volume II, Chapter 3, of the EBR-II System Design Description.

The shutdown circuit of EBR-II has 61 scram circuits active during power operation. Of these 61 circuits, 17 (listed below) are directed toward ensuring proper cooling of the reactor.

- (1) Reactor inlet coolant (h-p) No. 2--flow low
- (2) Reactor inlet coolant (h-p) No. 2--rate of change high
- (3) Reactor inlet coolant (l-p) No. 1--flow low
- (4) Reactor inlet coolant (l-p) No. 2--flow low
- (5) Reactor outlet coolant--flow low
- (6) Primary-coolant pump No. 1--motor-winding temperature high
- (7) Primary-coolant pump No. 2--motor-winding temperature high
- (8) Primary-coolant pump No. 1--motor breaker open
- (9) Primary-coolant pump No. 2--motor breaker open
- (10) Primary-coolant pump No. 1--M-G set clutch voltage low
- (11) Primary-coolant pump No. 2--M-G set clutch voltage low
- (12) Primary-coolant pump No. 1--M-G set cooling-water pressure low
- (13) Primary-coolant pump No. 2--M-G set cooling-water pressure low
- (14) Primary-coolant pump No. 1--motor current low
- (15) Primary-coolant pump No. 2--motor current low
- (16) Primary-coolant pump No. 1--motor-generator breaker open
- (17) Primary-coolant pump No. 2--motor-generator breaker open

Note: A flow monitor on the high-pressure inlet of pump No. 1 failed in April, 1970 and cannot be replaced.

Fig. 1. EBR-II Primary Cooling System

In addition to these 17 flow-oriented trip circuits, three pre-startup trip circuits (bypassed during operation) pertain to the primary coolant auxiliary pump:

- (1) Rectifier output voltage, low;
- (2) Current to pump, low; and
- (3) Battery charge or discharge current, high.

The existing design of the flow-monitor circuit was based on the assumption that the reactor always would be operated with the primary flow matched to the reactor power. This method of operation would have required the reactor operator to adjust the primary reactor flow continuously in step with the reactor power. Had the variable-flow mode of operation been adopted, the fixed-setpoint flow monitors could not have been preset to provide protection during periods of power change; hence monitors to control the rate of change of flow were incorporated into the design. Because the variable-flow mode of operation was never adopted, the problem of monitoring flow is greatly simplified.

Thirteen of the 17 flow-oriented trip circuits in the existing circuit operate on parameters other than actual flow. At present these scram circuits are attempts to anticipate failure of the flow system. Their incorporation into the circuit was based on the assumption that the speed of shutdown could be crucial after a failure of the pump flow. After six years of operating experience in EBR-II this aspect can now be considered in light of actual operating experience and with a better knowledge of the system dynamics.

II. SCRAM HISTORY OF THE FLOW-MONITORING SYSTEM

Table I lists some statistics on scrams, caused by the flow-monitoring system, for power levels over 1 MWt.

TABLE I. Statistics of Flow-monitor Scrams

Year	Total No. of Power Scrams	Scrams Originating from Flow-monitoring System	Scrams Originating from Rate-of-change Flow Monitors
1965	54	18	11
1966	42	14	3
1967	20	4	2
1968	37	17	10
1969	40	16	13

All flow-monitor scrams from 1965 through 1967 were spurious. However, the pattern of gradual improvement in the scram history was interrupted in 1968. During that year four scrams were clearly spurious. All others occurred from a series of elusive, intermittent malfunctions in the speed-control circuit of the pump.

Control of the pump speed with sufficient accuracy to satisfy the restraints of the flow-monitoring systems continues to be a problem. In the fall of 1969 after run 38, the thyatron speed-control circuits serving the primary pumps were replaced by solid-state controllers. Although it is too early to identify the extent of improvement, some favorable indication is inferred from the scram statistics in the second half of 1969. From June 21, 1969 to the end of the year only four scrams out of 17 originated from the flow-monitoring system; of these, two were rate-of-change trips.

During 1968 eight significant losses of operating time were caused by problems originating in the EBR-II flow system. The total time lost in these eight incidents was 14-1/2 days. Most of this time was used in troubleshooting the circuits that control the pump.

III. ANALYSIS OF FAILURE OF POWER TO PRIMARY PUMPS

The standard startup procedure at the EBR-II facility requires that 100% of full primary-coolant flow be established before reactor startup. This condition is maintained throughout startup, steady-state operation, and shutdown operations. Only during special tests, such as reduced-flow experiments, is the flow less than 100%. Adequate cooling is ensured by activating the flow monitors before the reactor is made critical. The scram setpoints for the high-pressure inlet and reactor outlet flow monitors are set manually at a value $6 \pm 2\%$ below full flow. For standard, full-flow operating conditions, the fixed-setpoint monitors provide the main defense against loss of flow. The monitors for rate of change of flow may offer increased protection during special reduced-flow experiments when the flow and trip points are not maintained fixed. Unfortunately, the differentiating circuit required in the rate-of-change monitors makes them inherently more sensitive to noise. For that reason, this circuit has precipitated many spurious power scrams throughout the history of the reactor.

A computer study has verified that the fixed-setpoint flow monitors can themselves protect the reactor from all postulated flow malfunctions. The conditions assumed for this study were as follows:

- (1) Power to both primary pumps failed totally and simultaneously, and the flowrate decreased as coastdown progressed.
- (2) The reactor power was 62.5 MWt.
- (3) The auxiliary pump was inoperative.
- (4) Only the fixed-setpoint flow monitors were available to scram the reactor.
- (5) The reactor was scrammed by the flow monitors at 92% of normal flow.
- (6) The only cooling to the reactor was from coastdown of flow.
- (7) Flow convection after coastdown was 3%.
- (8) There was no reactivity feedback from temperature effects.
- (9) Subassembly 5A2 was the hottest and therefore the controlling subassembly.
- (10) Maximum heat production was 9.09 kW/ft.
- (11) Inlet temperature to the reactor at the time of the pump trip was 700°F.

The characteristics of flow coastdown used in the calculation were obtained from flow tests made in June, 1964. Some of the results of those tests have been reported.¹ Figure 2 shows the flow-decay curve as measured and its linearized version used for this study. Results of subsequent flow tests agree with those reported in Ref. 1.

The THTB computer code was used in this study. In this computer program, input data for power and flow decay are taken as discrete points along the time curve. Values for conductivity and specific heat for fuel, cladding, and coolant (Appendix) were similarly included in the input. In the actual computation the computer interpolates between points. Figure 3 shows the power-decay curve used by the computer and the curve from which it was approximated.² This figure, reproduced from the curve for 4% rod worth taken from Fig. A-25 of Ref. 1, represents the response of the reactor from ten active control rods, 60% inserted, with a total worth of 2.8% Δk following scram.

After a pump failure, about 2.5 sec elapse before the flow value decays to the trip-level setting of the flow-trip monitors. The flow monitors require about 700 msec to act. The actual measured values for each flowmeter are: No. 1 $\approx$ 700; No. 2 $\approx$ 840; and outlet $\approx$ 620 msec.

To these values should be added the 16 msec required for the shutdown circuitry to respond; the maximum travel time for the control rods is about 300 msec. Total time for the instant of pump failure until full insertion of the control rods may be as much as 4 sec. These response times, obtained by measurements made specifically for this study, agree with earlier independent measurements (Ref. 3, p. 46).

To characterize the effect of flow-monitor delay times, a computer study was conducted to determine the transient temperature response of the reactor for a series of different delay times ranging from 2.5 to 7 sec. Figures 4 through 7 show the calculated results for temperatures of the fuel centerline, of the cladding at the fuel interface, and of the sodium coolant. All temperatures correspond to locations at the upper end of the active portion of the subassembly, where the maximum temperatures occur. The inside cladding temperature is most significant because maximum temperature limits are set by the formation of the eutectic between the uranium - 5 wt % fissium fuel alloy and the Type 304L stainless-steel cladding. The Materials Science Division of ANL has investigated the

Fig. 2. Flow Decay in EBR-II

Fig. 3. Power Decay in EBR-II

Fig. 4. Fuel-centerline Temperature Transient (62.5-MWt Operation)

Fig. 5. Cladding Temperature Transient (62.5-MWt Operation)

Fig. 6. Temperature Transient of Channel Sodium (62.5-MWt Operation)

Fig. 7. Initial Temperature Transients (62.5-MWt Operation)

eutectic formation between uranium - 5 wt % fissium alloy and Type 304L stainless steel.⁴ Liquid-phase alloy begins to form slowly at $705 \pm 5^{\circ}\text{C}$ ($\approx 1300^{\circ}\text{F}$). Experimental samples maintained at 710°C for one week showed evidence of eutectic formation, limited to the area of the grain boundaries.

Other tests⁵ indicate that the penetration rate of fuel-cladding material with molten fuel alloy (mp 1002°C) remains low (less than 10^{-4} in./sec) below 1070°C . The rate increases abruptly when the temperature increases from 1070 to 1085°C but drops again at higher temperatures. The maximum penetration rate at 1070 to 1085°C is of the order of 0.04 in./sec.

The effect of the time delay can be seen primarily in the first few seconds of the transient temperature response (see Fig. 7). With successively longer delays, a higher peak temperature is observed near the instant the reactor is scrammed. A lesser effect can be seen when a later peak temperature occurs at about 90 sec. Except for fuel centerline temperature, the second peak exceeds the magnitude of the initial peak. However, the second peak has been accentuated in this calculation by assuming that the auxiliary EM pump is inoperative. Minimum flow with the pump operating is never less than 5.5%. With the pump operative the magnitude of the second peak would be reduced and would occur several seconds earlier. The first peak has also been accentuated in this calculation by assuming that reactivity feedback from temperature effects is zero. This coefficient would have considerable effect in reducing the transient temperature values leading to the initial peak.

The margin between the peak temperature and eutectic-formation temperature for the THTB calculation (assuming a time delay of 4 sec) is about 50°F . For short-lived transients the temperature of eutectic formation could be exceeded without damage to the reactor core. Also significant is the indication that the response time of the flow monitors for power failure of the pump is not as important as had been anticipated, primarily because the coolant flow is relatively well sustained by inertial effects (see Fig. 2).

A. Uncertainty Factors

The temperature calculations reported above have not included any uncertainty factors. To demonstrate their effect, these factors have been applied to the maximum fuel temperature and to the maximum inside cladding temperature for the steady-state case and for scram delays of 2.5 and 3.75 sec. Table II shows the results of applying the uncertainty factors taken from Ref. 6 with one change: the factor for a 5% transient overpower was applied as a multiplier, rather than in the square root of the sum-of-the-squares technique. Applying this change has increased the effective uncertainty factor from 1.114 to 1.157.

TABLE II. Effect of Uncertainty Factors on Calculations of Temperature

Maximum Fuel Temperature, °F					
Steady State		2.5 sec Scram		3.75 sec Scram	
1208	1288*	1233	1317*	1265	1354*
Maximum Cladding Temperature at Fuel Interface, °F					
1087	1148*	1115	1180*	1150	1221*

* with uncertainty

B. Comparison with Other Independent Studies

The possibility of pump failure was considered initially in ANL-5719 (Addendum), in Section 2 of Appendix F,⁶ which includes a transient-temperature curve for that study. Although the computed flow-decay curve differs somewhat from the flow curves obtained by observation (see Fig. 2) and used for this study, these curves are in essential agreement. The transient-temperature curves also are comparable, except that no time delay was assumed for the earlier work; for this reason the peak temperatures are lower.

The results of an earlier study⁷ of the characteristics of transient temperatures in a special subassembly operated at reduced coolant flow (to simulate operation at 62.5 MWt) agree with those obtained in this study. Because no scram delay time was included, the corresponding peak temperatures were lower. The curves obtained in the earlier study also agree with those from this study.

Another independent study, by A. V. Campise of the EBR-II Test and Analysis Group has been directed toward establishing safe operational limits for several reactor parameters. Part of the work dealing with response times of the monitors of reactor temperature has produced a preliminary set of transient-temperature curves that can be compared directly with the THTB computer study reported here. The approach, however, is different in that the AIROS code was used. Input to the AIROS program is in the form of algebraic expressions for the flow-decay curve, the rod-scram curve, the rod reactivity worth, and the reactivity feedback. Preliminary results of this work support the general characteristics described by the THTB calculation and include delay in the reactor shutdown time. As was expected, the THTB calculation gave higher temperature values than did the AIROS calculation. The major contributing factor to this difference is that the AIROS calculation included reactivity feedback; whereas the THTB calculation assumed no reactivity feedback.

IV. FLOW MONITORING DURING LOW-FLOW EXPERIMENTS

Occasionally, special experiments are conducted at EBR-II that require conditions of reduced flow. Such experiments require that the flow and flow setpoint levels be lowered. Administrative procedure requires that the flow setpoints on the high-pressure inlet and reactor outlet flow monitors be set at $6 \pm 2\%$ below the established flowrate for all flow conditions. This procedure sets minimum value restraints on flow for all operating conditions. Although the fixed-setpoint flow monitors do not establish a maximum flow value, the pressure monitor in the reactor outlet plenum does establish one. The normal operating pressure in the outlet plenum is 6 psig, and the scram setpoint of the pressure monitor is set at 7 ± 0.5 psig. During the reduced-flow experiments the band between minimum and maximum flow is wider than normal. Assuming that monitors

for rate of change of flow are deleted from the shutdown circuit, malfunction of the pump controllers may permit the flow to assume any value not exceeding the minimum and maximum values without initiating a scram signal. Such a situation is acceptable if the reactivity effects induced by such flow perturbations are small enough. Power-reactivity decrement has been measured to evaluate the magnitude of the reactivity change induced by flow changes. These measurements have been made for flow values as low as 50% for several typical reactor loadings with depleted-uranium blankets. The maximum reactivity effect induced by a flow change of from 50 to 100% is approximately 26 Δk , which corresponds to a period of 108 sec. A positive period would exist only as long as the reactor remained overcooled. From this information, it is concluded that low-flow experiments do not present any special need for monitors for rate of change of flow.

V. FLOW PERTURBATIONS

Flow in the EBR-II is established by manual adjustment of controls at the control console. When the flow has been established it is maintained by a servo-feedback system that operates on pump speed. The flowmeters are not used in this servo system. Because the pumps are speed-controlled, the flow is dependent on the pressure drop through the flow system. The flow is somewhat dependent on temperature and is also slightly affected by the position of the control rods. Neither effect is significant in magnitude, and both are obscured by their slow rates of change. A more important effect relates to the two primary pumps, which are hydraulically coupled through the high-pressure plenum. However, because of the throttling of the flow in the supply lines, almost no coupling of the pumps through the low-pressure plenum exists. Hydraulic coupling causes the individual pump flows to be interdependent. A perturbation in the flow of either pump causes an opposite perturbation by the other pump. Because of this interaction, a small perturbation in the behavior of either pump produces an almost insignificant effect in the net total flow.

Three different incidents of flow perturbation in the history of the EBR-II have been documented to illustrate this effect. Figures 8 through 13 show the time trace of flow and rpm for the two pumps and the

Fig. 8. Pump No. 2: High-pressure Plenum Flow, 0516 hr, March 8, 1968

Fig. 9. Pump No. 1: High-pressure Plenum Flow, 0516 hr, March 8, 1968

Fig. 10. Pump No. 1: rpm, 0516 hr, March 8, 1968

Fig. 11. Pump No. 2: rpm, 0516 hr, March 8, 1968

Fig. 12. Total Reactor Flow, 0516 hr, March 8, 1968

Fig. 13. Reactor-blanket Flow, %, 0516 hr, March 8, 1968

net flow for a flow variation that occurred March 8, 1968, at 0516 hr. This perturbation was apparently precipitated by a fluctuation in line voltage associated with closing the turbo-generator breaker. Pump No. 2 responded to this power fluctuation with a reduced-flow perturbation, (Fig. 8) which caused an increase in flow for pump No. 1 (Fig. 9). A review of the curves for pump rpm (Figs. 10 and 11) identifies pump No. 2 as the offending pump that caused the perturbation. The rpm curve of pump No. 1 (Fig. 10) shows a speed disturbance caused by the drop in back pressure from pump No. 2, but the correction was very rapid. The output of pump No. 1 continued at an increased level until pump No. 2 recovered full flow a few minutes later. The net change in total flow from this perturbation is very slight (Fig. 12). The perturbation can be seen in the reduced flow of both traces of the blanket flow (Fig. 13). Minor flow perturbations of the pumps are common. They are caused by imperfections in the speed-control of the servo system rather than by system failure.

Two similar occurrences are described in Figs. 14 through 21. The first occurred at 1425 hr on Nov. 4, 1968. The event began with an upward perturbation originating with pump No. 1 (Fig. 14). The leveling effect caused by the pump interaction can be seen by the negligible, but perceptible, net increase in total flow (Fig. 16). The second event occurred at 1510 hr on Nov. 4, 1968. This was a much more violent perturbation initiated from pump No. 1 with flow changes in both directions (Fig. 18). It ultimately caused the reactor to scram. However, the net total reactor flow was only slightly affected (Fig. 20). All three events described evidently were precipitated by disturbances on the power bus and predate the extensive repair work on the speed-control servo of the pump.

VI. CONCLUSIONS

Some specific conclusions obtained from this study are:

- (1) the fixed-setpoint flow monitors adequately protect the reactor from total failure of power to the primary pumps;
- (2) the fixed-setpoint flow monitors also provide adequate protection against flow perturbation induced by malfunction of individual pump-speed controllers; and

Fig. 14. Pump No. 1: High-pressure Plenum Flow, 1425 hr, Nov. 4, 1968

Fig. 15. Pump No. 2: High-pressure Plenum Flow, 1425 hr, Nov. 4, 1968

Fig. 16. Total Reactor Flow, 1425 hr, Nov. 4, 1968

Fig. 17. Reactor-blanket Flow, %, 1425 hr, Nov. 4, 1968

Fig. 18. Pump No. 1: High-pressure Plenum Flow, 1510 hr, Nov. 4, 1968

Fig. 19. Pump No. 2: High-pressure Plenum Flow, 1510 hr, Nov. 4, 1968

Fig. 20. Total Reactor Flow, 1510 hr, Nov. 4, 1968

Fig. 21. Reactor-blanket Flow, %, 1510 hr, Nov. 4, 1968

- (3) reduced-flow experiments do not present any special need for flow protection that is not met by the fixed-setpoint flow monitors.

These three cases represent all forms of pump malfunction. Therefore, it is concluded that monitors measuring the rate or change of flow and the many anticipating monitors on pump functions (see p. 15) can be deleted from the reactor shutdown without compromising reactor safety. The necessary redundancy in flow monitoring can be obtained by operating the existing fixed-setpoint flow monitors in series-trip logic. Arrangements other than series-trip logic also may meet the necessary requirements. These conclusions appear to be compatible with the new RDT standard C-16-1.

APPENDIX

Physical Properties of Some Reactor Materials

Stainless Steel Cladding--Density = 488 lb/ft³; Phase Change at 2610°F

Temperature, °F	Conductivity, $\frac{\text{Btu}}{\text{hr-ft-°F}}$	Specific Heat, $\frac{\text{Btu}}{\text{lb-°F}}$
390	10.1	0.11
572	10.7	0.12
932	12.4	0.16
1200	13.4	0.18
1400	14.2	0.17
1700	15.2	0.17
2600	15.2	0.16

Sodium--Density = 51.8 lb/ft³; Phase Change at 1620°F

Temperature, °F	Conductivity, $\frac{\text{Btu}}{\text{hr-ft-°F}}$	Specific Heat, $\frac{\text{Btu}}{\text{lb-°F}}$
390	47	0.32
500	45	0.315
650	43	0.31
850	40	0.305
1050	37	0.30
1250	34	0.30
1550	30	0.305
1620	29	0.307
1620	45	0.307
2300	45	0.345
4000	45	0.345

Fuel Alloy--Uranium-5% Fissium (86% density)

Density = 961 lb/ft³; Phase Change at 1850°F

Temperature, °F	Conductivity, $\frac{\text{Btu}}{\text{hr-ft-°F}}$	Specific Heat, $\frac{\text{Btu}}{\text{lb-°F}}$
700	11.1	3.1
932	12.9	3.1
1040	14.1	3.1
1184	15.8	3.1
1292	16.9	3.1
1382	17.8	3.1
1652	19.1	3.1
1850	20.1	3.1
2030	14.9	3.1
4000	14.9	3.1

REFERENCES

1. H. O. Monson et al., Primary Flow Decay Experiments (Zero Power, 600°F), EBR-II Technical Memorandum No. 20 (June 24, 1964).
2. L. J. Koch et al., Hazard Summary Report, Experimental Breeder Reactor II (EBR-II), ANL-5719 (May 1957).
3. G. S. Brunson, J. R. Karvinen, and R. A. Call, "Response-Time Measurements," in Reactor Development Program Progress Report for 1969, ANL-7618, pp. 44-47 (Oct. 27, 1969).
4. S. T. Zegler, H. V. Rhude, and J. A. Lahti, Compatibility of Uranium-5 w/o Fissium Alloy with Types 304L and 316 Stainless Steel, ANL-7596 (September 1969).
5. C. M. Walter and L. R. Kelman, The Interaction of Iron with Molten Uranium, J. Nucl. Mat. 20 (3) 314-322 (June 1966).
6. L. J. Koch et al., Addendum to the Hazard Summary Report, Experimental Breeder Reactor II (EBR-II), ANL-5719 (Addendum) (June 1962).
7. R. K. Winkleblack to R. F. Sweek, "EBR-II Controlled-Flow Experiment," private communication (August 1968).