Inherently Safe Shutdown of a Liquid Metal Reactor
Upon a Loss of Intermediate Cooling*

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Abstract

Two unprotected (i.e., no scram or plant protection system action) loss-of-heat sink transients are scheduled to be performed on the Experimental Breeder Reactor-II in the Spring of 1986. One is to be initiated from full power (about 60 MW) and the other from half power. The loss-of-heat sink in each test is to be accomplished by essentially stopping the secondary-loop sodium coolant flow in about 20 s. Pretest predictions for the two tests, provided herein, show the reactor to passively shut down with average reactor outlet coolant temperatures rising no more than 5°C before dropping and asymptotically approaching a quenching (or "smothering") temperature. This mild behavior is seen to be a direct consequence of the substantial negative reactivity feedback of the reactor coupled with the heat capacitance of the massive sodium pool in the primary tank.

Introduction

Within the last several years much attention has been given to nuclear reactor plant accidents which originate beyond the primary coolant system and cause a reduction or loss of the normal balance of plant (BOP) heat sink for the primary system. Limiting cases of this type of accident in a liquid metal reactor

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(LMR) plant are a complete loss of feedwater flow leading to a steam generator dry-out condition or a stoppage of secondary sodium flow. Both of these accidents result in the BOP being unable to remove heat from the primary coolant.

Typically, a loss of heat sink (LOHS) will cause an over-temperature condition in the primary system which will set off an alarm and may cause automatic plant shutdown via the plant protection system (PPS). Should this first line of defense fail, there could be a potential for reactor damage in some reactor plant designs. The Experimental Breeder Reactor-II (EBR-II), however, is predicted to undergo a benign response to this incident. The EBR-II is a pool-type LMR which has a large primary tank containing some 340 m³ of liquid sodium. The heat capacitance of this large volume and associated structure coupled with the reactor's negative reactivity feedback characteristics provide the reactor inherent protection against damage during a stoppage of secondary sodium flow. This transient can be safely initiated at normal operating conditions without a reactor scram or any other plant protective action.

Tests are planned to demonstrate this predicted capability in order to exploit the mechanism fully in future LMR designs. Another equally important goal is to gather data to be used in the validation of analytical models that are being used to study LMR behavior. Demonstrated inherent safety mechanisms coupled with data from validated simulation models should be of substantial value in the design and licensing of future LMR's. Simpler, more reliable, and more cost-effective designs should result if

inherent safety mechanisms allow multiple, redundant, and independent safety systems to be eliminated. Moreover, if the heat removal capability of the BOP is not needed to safely shut down the unscrammed reactor, then probably the amount of structural hardening (e.g., seismic resistance) in the BOP could be reduced to further mitigate plant costs.

Reactor test data for LOHS conditions is extremely limited, although in 1983 the French successfully performed an LOHS experiment on their Rapsodie reactor plant which was a test similar to those being discussed here (Essig et al., 1985). The purpose of this paper is to provide pretest predictions for two LOHS tests to be performed on the EBR-II plant in the Spring of 1986. These predictions are based on simulations performed with the aid of the NATDEMO computer code (Mohr and Feldman, 1981).

The EBR-II Reactor Plant

The EBR-II plant is located in Idaho and was designed by Argonne National Laboratory who operates it for the U. S. Department of Energy. EBR-II has been in operation since 1964 and has served primarily as a fast-flux irradiation facility since 1967. The plant produces about 20 MW of electricity when operated at its full power of 60 MW. The corresponding reactor, secondary, and steam system flow rates are approximately 465, 300, and 31 kg/s, respectively.

The EBR-II reactor, as shown in Fig. 1, is submerged in a large pool of liquid sodium. Two main primary pumps draw sodium from this pool, which is normally at 371°C, into the two reactor

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Fig. 1. EBR-II Primary System

inlet plena. About 86%* of the flow passes through the high pressure plenum and cools the core subassemblies while the remainder passes through the low pressure plenum and cools the radial reflector and blanket subassemblies. The flow through all subassemblies merges in the upper plenum before passing through the outlet pipe to the intermediate heat exchanger (IHX). After passing through the IHX, the flow exits into the primary tank before entering the main pumps. The typical driver subassembly in the reactor contains 91 pins of sodium-bonded metallic fuel.

Figure 2 provides a schematic of the entire EBR-II plant-- note the connection of the BOP to the primary system via the IHX. Some 60 m of pipe carries sodium from the IHX to the steam generator which is located outside the reactor building in the sodium boiler building. A similar length of pipe takes return flow to the IHX inlet. An electromagnetic (EM) pump in the return leg drives the secondary sodium flow. The steam generator has two parallel superheaters and seven parallel evaporators.

Selection of Transients for LOHS Tests

Several potential test transients involving loss of heat sink were simulated with the NATDEMO code. A total loss of feedwater was compared with a complete stoppage of secondary flow in terms of meeting test objectives and operational constraints. Both were initiated near rated conditions and assumed no PPS action. In both cases the primary system responded essentially the same except that the loss-of-feedwater case effectively

*This value was recently changed to about 84%.

Fig. 2. Schematic of EBR-II Plant for NATDEMO Code

delayed the loss of heat sink (of the reactor) by 2 to 3 minutes. This time was required first to dry out the steam generator and then for the secondary sodium to traverse the distance from the steam generator to the IHX. While both cases produced the desired condition in the primary system, a dryout could be detrimental to the steam generator integrity and would require a considerable amount of time for the plant to be returned to its normal operating condition. Therefore, we focused our attention on the stoppage of secondary flow case.

The idealized situation of no primary system heat losses is unattainable in EBR-II. The shutdown coolers, Fig. 1, remove about 350 kW when the louvers on the air-side heat exchangers are open, but still remove about 120 kW when they are closed. Other sources of parasitic heat losses, such as the instrument thimble and shield cooling systems, the primary sodium purification system, and the heat transfer through the walls of the primary tank, remove at least an additional 250 kW. Another means of heat loss is via the IHX, whose secondary-side flow is very difficult to stop completely. There are no valves for stopping the secondary flow and, if total power to the pump is turned off, the natural convective flow remains almost indefinitely at 5-8% of its rated value. If power to the pump is available, the pump head can be reversed to oppose the natural buoyancy in the secondary loop. Reactor plant operating procedures, however, prohibit reverse flow and potential errors in flow readings therefore limit the minimum indicated flow to no less than 0.5%

of full flow. The assumed minimum value of 0.5% secondary flow removes about 200 kW via the IHX.

The only significant sources of heat gain (except for the reactor) are the primary pumps, which produce about 200 kW, and the tank heaters. The heaters are capable of adding about 600 kW to the primary tank sodium but will be turned off during the planned tests.

In order to assess the influence of unavoidable primary tank heat losses on an LOHS test, two transients were simulated with the NATDEMO code (Feldman and Mohr, 1984). Both were initiated from the same assumed conditions which closely approximate the rated conditions of the plant. In Case 1 the secondary sodium flow rate was linearly reduced to 0.5% of its initial value in about the first 20 seconds of the transient and then held constant. In Case 2 the transient was initiated by simply tripping the pump in the secondary sodium loop. Since this is an EM pump with no moving parts, the flow decay characteristic is determined by the inertance of the moving sodium. After the coastdown, the flow rate is determined by the net loop buoyancy. Case 1 represents a situation where the net power losses from the primary tank are minimal. Case 2 represents a contrast where the net heat loss is more than ten times greater than for Case 1. There was no scram or PPS action in either case.

For the two cases, temperature histories of the reactor inlet and of the outlets of the driver, reflector, and blanket regions were compared. Differences tended to be only of the order of 5°C. After about the first 1800 seconds, changes in

both cases were very gradual and important residual differences between the two cases could be discerned. At 2600 seconds, for example, the very low secondary sodium flow of Case 1 permitted a total of only 0.3 MW to be transferred from the primary system versus 3.8 MW in Case 2. This 3.5 MW difference in heat removal rate was almost entirely offset by the difference in fission power which was 1.9 MW for Case 1 versus 5.1 MW for Case 2. As a result, the net power to the primary reactor tank at 2600 s was essentially the same for both cases, being 2.5 MW for Case 1 and 2.4 MW for Case 2. Thus, we conclude that even if substantial, unavoidable heat losses are present, a very meaningful unprotected LOHS test can be conducted with the EBR-II.

The more severe of the two planned tests is very similar to analytical Case 1 as described above except that the initial reactor inlet temperature has been reduced by 19°C from its normal value of 371°C to 352°C. This reduction is to help minimize the thermally-induced stresses in some of the primary tank structures. The milder test will be performed first and is the same as the more severe one except it is to be initiated from 30 MW instead of 60 MW. This 30 MW test provides an opportunity to check out the testing methods and further validate our analytical model before performing the 60 MW test.

NATDEMO Simulation Model

Figure 2 provides a schematic representation of the NATDEMO model. Important features of the primary system portion include detailed descriptions of power generation, parallel-channel thermal-hydraulics with buoyancy, reactivity feedback, reactor

inlet and outlet piping, the IHX, the main primary pumps, the auxiliary pump, and the primary sodium pool. The two superheaters, the seven evaporators, the EM pump, and the hundreds of meters of secondary sodium piping are included in the secondary system model. Variations in pipe cross section and elevation, and pipe wall and coolant heat capacitance are represented and are needed to accurately simulate natural convection within the secondary sodium loop.

The steam system portion of NATDEMO includes a thermal equilibrium drum model, the hydraulics for the natural convective recirculation flow through the seven evaporators, the steam pressure control system, the steamline and turbine, and the turbine bypass line. In the model the turbine generator, the condenser, and the feedwater train are not explicitly represented. There is an optional steam drum level control model, but the feedwater temperature and the extractions from the main steamline (which are used to heat the feedwater) must be specified by inputs.

Since a more detailed description of the NATDEMO model is provided by Mohr and Feldman (1981), we will elaborate further only on those aspects of the model which are particularly pertinent to the transients under consideration. These include power generation, reactivity feedback, the reactor tank, and the reactor inlet piping.

The 16 concentric rows of reactor subassemblies are divided into three basic regions: driver fuel (rows 1-6), radial stainless steel reflector (rows 7-10), and radial depleted ^{238}U blanket (rows 11-16). Each of these regions has separate power

generation functions containing both prompt and delayed components. The Shure correlation (Shure, 1961 and Shure, 1972) for decay heat is utilized, but adapted so that any desired reactor operating history prior to final shutdown can be input as the prerequisite to the decay heat evaluation. In addition, the effect of current power level (with a critical core) upon the level of decay power present is included in the simulation. This latter effect was found to be important in the present study.

Table I provides the values of the nine components of reactivity feedback used in the NATDEMO code along with their estimated uncertainties. These characteristics are essentially the same as those of Mohr and Chang (1985). All of the nine reactivity feedback coefficients are negative, with the frequent exception of the nonlinear (bowing) effect. Most of the reactivity effects used in the model have been verified by rod drop, reactor oscillator, or power reactivity decrement (PRD) test data. A brief description of the nine components follows and parallels the order given in the table.

The driver fuel expansion effect is assumed to be directly related to the expansion of the ^{235}U fuel and therefore is confined to the active fuel height of the core and also to the first six rows of subassemblies. The driver sodium and steel expansion effect is directly related to the driver sodium coolant temperature and again is confined to the fueled driver region in the first six rows of subassemblies. The axial upper reflector sodium expansion effect is related to the sodium coolant tempera-

TABLE I. Calculated Reactivity-feedback Components and Uncertainties

Components	Contributing Region, Rows	Nominal Reactivity-feedback Coefficient (values used in the analysis, 10^{-4} $\$/^{\circ}\text{C}$)	Power Reactivity Decrement (PRD), $\$$	Estimated Uncertainty, %
Reactor fuel expansion ^a	1-6	-4.91	-0.0687	11
Reactor sodium and steel expansion ^a				
Sodium	1-6	-11.9	-0.0812	7
Steel	1-6	-2.2	-0.0155	10
Total	1-6	-14.1	-0.0967	7
Reactor upper reflector sodium expansion ^a	1-6	-6.32	-0.0667	7
Reactor reflector sodium expansion ^b	7-10	-2.20	-0.0104	--
Optical effect ^a	1-6	-0.671 ^c	-0.0109	10
Control-rod bank extension ^a	5	-7.16 ^d	-0.0672	5
Reactor grid plate expansion ^b	16	-14.5 ^e	0	20
Thermal bowing ^{b,f}	2-10	+9.8	+0.025 ^g	25
Reactor lower reflector sodium expansion ^a	1-6	-6.32	0	7

^a term.

^b term.

^c value of coefficient assuming $\partial k/\partial T = -0.0546/(T_f + 273)$.

^d corresponds to the extension-rod portion within the reactor and a rod-bank of 0.154 $\$/\text{cm}$.

^e corresponds to a uniform change throughout the entire grid.

^f related to effective total coolant ΔT .

^g component has a threshold ΔT at 0.75 of nominal.

tures above the core including the upper stainless steel shield region in each driver subassembly within rows 1-6. The radial reflector sodium expansion effect is evaluated for the sodium region within the radial stainless steel reflector subassemblies of rows 7-10 for the total axial height of each subassembly. The Doppler effect in EBR-II is very small and varies approximately as the reciprocal of the absolute fuel temperature.

The thermal response of the control-rod bank extension effect is determined by dividing the subassembly and extension rod into sections, each of which responds to changes in the surrounding sodium temperature during a reactor transient. The control rods (movable fueled subassemblies) in EBR-II are suspended from the control-rod drive supports mounted above the vessel on the reactor tank cover. Thus, the actual control portion (fueled zone) is suspended by long shafts extending from the core upward through the outlet plenum, reactor vessel cover, and tank sodium. The total rod bank effect is obtained by summing the individual contributions of axial displacement, weighted by node length, and multiplying by the rod bank differential worth appropriate for a specific bank insertion and core loading.

The upper grid plate expansion effect takes into account the time response of the massive perforated grid plate, which supports the subassemblies at their lower end; the incoming temperature wave is applied uniformly to the grid plate in all directions. Once the row-wise radial displacements are known, the reactivity contribution is determined by multiplying the individual movements by the reactivity "worth" of the row displacements.

The greatest contribution to nonlinear effects is assumed to be subassembly thermal bowing, which can be a positive or negative effect, depending on the specific core loading. It is related to the radial displacement of the subassemblies and driven by the temperature differences across hex cans located principally in the radial reflector region. In the model described here, the effect is treated on a quasi-steady basis and is assumed to be related directly to coolant temperatures in the radial reflector. In contrast to almost all of the above reactivity effects, bowing effects are not necessarily derived by applying a local material temperature change to a reactivity worth of that material. Thus, for example, the radial displacement of driver fuel may not be caused by local temperature changes, but instead by temperature changes in the outer reflector region. For some core loadings, these bowing forces tend to compact the driver fuel region increasingly as the reflector power-to-flow ratio increases--thus causing a positive reactivity effect. For many core loadings, however, the bowing effect is small (or negative) because no significant bowing forces are generated in the outer regions. The time response of the bowing feedback tends to be slow because the thermal dynamics of the radial reflector are sluggish due to high heat capacity and low coolant velocities.

In June 1984 a series of eight tests was performed on the EBR-II to investigate the inherent negative reactivity feedback characteristics of the reactor (Mohr and Chang, 1985). In four of the tests the reactor inlet temperature was perturbed and the change in power was noted, and in the other four the primary flow

rate was perturbed while the reactor inlet temperature was held constant. Prior to these tests the NATDEMO model contained all of the reactivity components given in Table I except the last one. The need to include the axial lower reflector sodium expansion effect in the model was clearly demonstrated by the four inlet temperature perturbation tests. The effect of sodium expansion in the lower reflector is analogous to that in the upper reflector, except that it is related only to reactor inlet temperature. Like the grid-plate effect, its contribution to the PRD at power is negligible.

The total reactivity feedback thus comprises nine terms and all are evaluated relative to a base temperature of 371°C, i.e., the nominal sodium pool temperature. The first three, the fifth, sixth and ninth terms are regarded as "prompt" and have time responses in the range of 0.1 to ~ 1 second at rated conditions. The remaining feedbacks are "delayed" and have time responses in the range of ~ 1 to 50 seconds.

The effects of uncertainties in reactivity feedback (Table I) are expected to have only a minor impact on the results. Although individual uncertainties may be as high as $\pm 25\%$, the overall uncertainty is about $\pm 4\%$. Moreover, the model was well confirmed by the June 1984 test data. In addition, the significant feedback terms are all negative and thus all contribute to a decreasing reactor power response to a rising core inlet temperature. Since the time scale of the transient in each case is dictated by the slow reactor tank dynamics, the reactor power uncertainty is expected to be similar to the above

overall feedback uncertainty. Thus, the impact of these uncertainties should be small with regard to both the peak core temperatures and the tank temperature behavior.

The arrangement of the primary system components within the primary tank is as shown in Fig. 1. A two region primary pool model has been derived which includes an inactive (essentially non-flowing) region and an active region comprising the primary coolant streams as they flow from the IHX to the two primary pump inlets. The thermal behavior of these streams is described by a "leaking pipe" mixing model wherein each coolant stream is imagined to flow within a pipe with a porous wall and thus mix with the bulk sodium. The total volume taken up by the two "porous pipe" regions is approximately 22%* of the reactor tank sodium. The flow rate associated with the streams "leaking" out of (and into) each porous pipe is adjusted as a fixed fraction of the primary flow rate. A heat balance on the primary tank structure and sodium pool provides a dynamic model of the bulk sodium in the primary tank which interacts with the above mixing model.

The reactor inlet piping model (main pumps to inlet plena) describes the four piping paths shown in Fig. 1. Two piping runs are described for each inlet plenum, i.e., the high pressure plenum (HPP) and low pressure plenum (LPP). The piping for both the HPP and LPP in each run are described by three axial and three radial nodes. The flowing sodium can exchange heat with

*In Mohr and Feldman (1981) this value is estimated to be 10%. The current value is based on subsequent test data.

the pipe walls and ultimately with the sodium pool (sink) between the pumps and the reactor.

Numerous transients have been analyzed with the NATDEMO code including loss-of-flow (LOF) events, reactor scram events, reactivity disturbances, secondary flow disturbances, and combinations of the above. In many cases, direct comparisons were made between predicted and measured results, particularly for scram and LOF/scram incidents (Mohr and Feldman, 1981, Lambert et al., 1981, Singer et al., 1976, Singer et al., 1981, Singer et al., 1982, and Singer et al., 1979). Because of the generally good comparison between predicted and measured results, substantial confidence is placed in the NATDEMO simulation.

Results of Pretest Predictions

Both of the planned tests are to be initiated by linearly reducing the secondary-loop sodium flow rates from its initial value to 0.5% of that value in 20 s, Fig. 3. This is to be the only forcing function and there is to be no scram or other reactor protective action. The near stoppage of secondary flow virtually eliminates heat transfer from the primary system to the secondary via the IHX. Thus, in both cases the IHX primary outlet temperature rises rapidly toward the primary inlet value. At only 30 s into the transient the inlet and outlet temperatures differ by about 1°C and remain close together throughout the remainder of the transient, Fig. 4. Note that the primary flow rate is maintained at its full power value in both cases, and the IHX primary-side inlet temperature, Fig. 4, is essentially the same as the reactor outlet temperature, Fig. 5.

Fig. 3. Secondary-Loop Flow Rate

Fig. 4. IHX Primary-Side Inlet and Outlet Temperatures

Fig. 5. Reactor Inlet and Outlet Temperatures

The primary sodium flow exiting from the IHX mixes with the bulk sodium in the primary tank before entering the two main coolant pumps located at the reactor inlet. This mixing plus heat transfer transforms the IHX outlet temperatures of Fig. 4, to the reactor inlet temperatures of Fig. 5. Since the sodium only partially mixes with the bulk sodium on its journey from the IHX outlet to the reactor inlet, the reactor inlet temperature responds much more rapidly than does the reactor tank bulk mixed-mean temperature, Fig. 6.

Because the negative reactivity feedback effects in the EBR-II are substantial, the rising reactor inlet temperature causes a dramatic reduction in reactor power, Figs. 7 and 8. In Fig. 7, which provides the total reactor power, the driver power and its fission and decay components for the 60 MW case, all percentages correspond to the 60 MW initial total reactor power. Figure 8 provides the analogous set of results for the 30 MW case.

Figure 9 provides the average outlet coolant at the top of the subassemblies for the driver, blanket and reflector regions in the 60 MW case. The rising reactor inlet temperature and the decreasing reactor power have opposing effects on the exit coolant temperature. The drop in reactor power is so dramatic that the peak values in all three curves of Fig. 9 are within 0.2°C of their respective initial values. In the blanket region changes in the inlet coolant temperature are translated relatively slowly to the outlet because of the greater thermal inertia of the blanket elements. Thus, the dropping power effect dominates

Fig. 6. Reactor Tank Average Temperature

Fig. 7. Power, 60 MW Case

Fig. 8. Power, 30 MW Case

Fig. 9. Average Subassembly Outlet Coolant Temperature, 60 MW Case

initially and causes the slight initial dip in blanket temperature shown in the figure.

Figure 10, which is analogous to Fig. 9, provides the outlet coolant temperature histories for the 30 MW case. In Fig. 10 the peak values in the driver and reflector curves are only 2°C higher than their respective initial values and the peak in the blanket curve is only 5°C higher than its initial values.

A key observation from Fig. 5 is that for each test case the reactor inlet and outlet temperature curves tend to asymptotically approach constant values which differ only slightly from each other. The quenching temperature, which is the core temperature at which neutron fission goes to zero (via reactivity feedback with no change in rod position) lies between these two asymptotes. Thus, one can estimate the quenching temperature to be 396.5°C for the 60 MW case and 377.0°C for the 30 MW case.

At substantially longer times than is practical to include in the transient analyses, a steady-state condition could be predicted in each case in which the reactor power is consistent with the specified 0.5% secondary flow and the reactivity feedback is unchanged from the initial condition. Additional steady-state analyses were performed with the NATDEMO code to determine this long term steady-state condition for the 60 MW case. This analysis indicates that the reactor inlet temperature goes to 394.4°C and reactor power goes to 283 kW. This power coupled with the total reactor flow produces a 0.5°C reactor temperature rise. Hence, the quenching temperature is about 394.6°C

Fig. 10. Average Subassembly Outlet Coolant Temperature, 30 MW Case

(i.e., $394.4 + 0.5/2$) or about 1.9°C less than estimated from Fig. 5.

The excess reactivity, Fig. 11, is determined by a weighted average of the global reactor temperature. The rising reactor inlet temperature initially causes a substantial dip in excess reactivity, but the subsequent drop in reactor power causes the excess reactivity to return to nearly zero as the average reactor temperature returns toward its initial value and the reactor outlet temperature drops. Although the final portion of the curve for the 60 MW case shows a downward slope, the slope must eventually turn upward and the curves of both cases should asymptotically approach zero. Thus, the reactor will in the long term return to a critical state when the sum of fission plus decay power equals the heat losses. The increase in reactivity needed for the curves to approach zero can only result from a decrease in reactor temperatures. This is consistent with the 60 MW case quenching temperature predicted by steady-state analysis being lower than that estimated from Fig. 5. Similarly, we can expect the quenching temperature for the 30 MW case to also be slightly less than the 377.0°C value estimated from Fig. 5.

Discussions and Conclusions

Although individual reactor subassemblies can have exit coolant temperatures which are somewhat larger than those presented in Figs. 9 and 10 for the average subassemblies, the unprotected LOHS transient in EBR-II is so mild that differences between the behavior of the hottest subassembly in each region and the average one are inconsequential. Moreover, the steep

Fig. 11. Excess Reactivity

monotonic drop in reactor power reduces the radial temperature rise from the coolant to the fuel element.

The analysis was performed on a nominal basis to provide the best estimate of the performance of the reactor without considering uncertainties. In addition to the normal modeling uncertainties, however, there are the uncertainties caused by changes in reactor core loading from run to run. A typical change in core loading, for example, can alter the bowing component of reactivity enough to cause up to a 5°C change in primary tank temperature rise in the 60 MW case.

Fortunately, the phenomena governing the behavior of the reactor are well understood and we have extensive measured data which has been used to validate the NATDEMO code. The beginning of SHRT test 26 (Mohr and Chang, 1985), which was run in June 1984 for example, has many similarities to the 60 MW case. Both were initiated from a relative power-to-flow ratio of about 1.0, although the SHRT test 26 initial power level is about 70% of that of the 60 MW case. Both were initiated by a rapid reduction in secondary flow (without scram) causing a rise in reactor inlet temperature and a reduction in reactor power. Although the maximum increase in reactor inlet temperature is limited to about 16°C in SHRT test 26, the data obtained were valuable in validating the reactivity feedback model and the primary tank mixing model. Perhaps even more important is that this and other prior experiences provide substantial confidence in the generally benign nature of the tests as demonstrated in the above analytical results.

In conclusion, the pretest predictions clearly show that the EBR-II reactor and primary system should respond rather benignly to an unprotected LOHS transient from full power in spite of the potential severity this type of transient may suggest. In addition, the tests planned for the Spring of 1986, should provide experimental confirmation of this conclusion and of the pretest predictions contained herein.

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Captions for Figures

- Fig. 1. EBR-II Primary System
- Fig. 2. Schematic of EBR-II Plant for NATDEMO Code
- Fig. 3. Secondary-Loop Flow Rate
- Fig. 4. IHX Primary-Side Inlet and Outlet Temperatures
- Fig. 5. Reactor Inlet and Outlet Temperatures
- Fig. 6. Reactor Tank Average Temperature
- Fig. 7. Power, 60 MW Case
- Fig. 8. Power, 30 MW Case
- Fig. 9. Average Subassembly Outlet Coolant Temperature,
60 MW Case
- Fig. 10. Average Subassembly Outlet Coolant Temperature,
30 MW Case
- Fig. 11. Excess Reactivity