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IMPLICATIONS OF THE EBR-II INHERENT SAFETY DEMONSTRATION TEST *

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Extensive thermal-hydraulics testing at EBR-II culminated in the Inherent Safety Demonstration Test on April 3, 1986. This work may well lead to fundamental changes in the approach to the design and licensing of liquid-metal-cooled reactor (LMR) power plants. The EBR-II test program has thus far demonstrated (1) passive removal of decay heat by natural circulation, (2) passive reactor shutdown for a loss of flow without scram, and (3) passive reactor shutdown for a loss of heat sink without scram. Supporting analyses indicate that these characteristics can be incorporated into larger commercial LMRs and be used as the basis for a totally new passive control strategy. Analyses and tests are now in progress to show that LMRs with these characteristics and the passive control strategy are also inherently safe for unprotected overpower accidents.

1. Introduction

The traditional approach to reactor safety is based on defense in depth and includes three levels of safety design. The first level emphasizes prevention of accidents with reliable equipment and intrinsically stable design features. An example is General Nuclear Plant Design Criterion 11, Section 10, Code of Federal Regulations Part 50, which requires the reactor to have a net negative prompt power coefficient of reactivity and assures stable operation in the power range. The second level of design focuses on protection against anticipated operational occurrences which may arise as result of equipment failures or adverse environmental conditions. An example is Criterion 20 of 10CFR50 which requires an automatic protection system to protect against consequences of operational occurrences. The third level of design is focused on protection and design margins for extremely unlikely and hypothetical events. LMR designs have included redundant diverse and independent reactor shutdown functions to protect against extremely unlikely equipment failures. Additional thermal and structural margins to accomodate anticipated transients without scram and/or hypothetical core disruptive accidents (HCDAs) in the plant and containment have been included.

A fundamentally different approach to reactor safety is evolving with the development of "inherently safe" LMR designs [1,2]. The initial approach to inherent safety has been directed toward the third level of safety design. The shutdown and subsequent heat removal have been accomplished by natural processes and in-core direct acting devices assuming safety equipment in the first two levels of safety design had failed. An integral part of the strategy in eliminating an HCDA from design consideration is showing that the reactor is protected even for anticipated transients without scram.

The results of the EBR-II tests validate the initial inherent safety design approach but suggest an even larger role for inherent safety in design and operation of LMRs – that is, inherently safe, passive shutdown can be used for the first and second level of safety design, and a much simplified automatic protection system can be used as a backup.

Two of the three worst-case events, loss-of-flow without scram (LOFWS) and loss-of-heat-sink without scram (LOHSWS), have been tested and found to be benign in EBR-II. The peak temperatures were within limits for normal operation and operational transients. A few years ago these two types of events were thought to result in total core disruption. The third event, transient-overpower without scram (TOPWS) can also be made to be benign and is the subject of follow-on testing at EBR-II. These results allow for a fundamentally different approach to LMR design and licensing.

The traditional second level of safety design, which emphasizes reliable engineered safety systems, has its advantages from the point of view of engineering design. Engineered-safety-systems can be made to be te-

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stable, and can be analyzed quantitatively for reliability. They are also flexible in that the electronic systems can be upgraded to meet changing criteria without major modification to the plant mechanical systems. Unfortunately, the high levels of reliability required bring significant penalties in increased complexity, cost, and difficulty in maintenance and operation. This is especially true when, to preserve independence of diverse and redundant systems, special care must be taken to ensure physical separation and protection against such environmental factors as fire and earthquakes.

A fundamentally different approach to prevention and accommodation of accidents is to rely on inherent characteristics of the system, thus eliminating or greatly reducing the number of active engineered-safety-systems. Such characteristics must be verified by test and, because they involve system interactions, overall plant testing is required. EBR-II is embarked on just such a program, defining and conducting plant tests to demonstrate the existence of inherent characteristics of the plant to ensure safety. These tests are seen to be useful not only for defining the design features required, but also in establishing a testing approach for future plants so that their response to upsets may be confirmed.

In broad terms, the essential function of control and protection of a nuclear plant is maintaining a proper balance between heat generation in the reactor and heat removal from it. If this balance is maintained globally and locally, temperatures of reactor structures, notably the fuel cladding and primary reactor boundary, will be maintained within design limits. As discussed previously this function is traditionally accomplished at the first, second, and third levels of design with active control and protection systems.

The EBR-II shutdown heat removal testing program [3] has experimentally examined the inherent capability of LMR's to maintain the balance between heat generation and heat removal. Thus far the program has focused on upsets in the heat removal. The two large categories of reduced heat removal events, loss of flow and loss of heat sink, have been addressed, and it has been found that the reactor will passively reduce power for events in either category and inherently balance heat generation to heat removal.

Specifically the data from the tests have demonstrated:

(1) Passive decay-heat removal by natural circulation following reactor shutdown, Planchon et al. [4].
(2) Passive reactor shutdown following a loss of forced circulation (loss of flow without scram), Mohr et al. [5].
(3) Passive reactor shutdown following a loss of balance-of-plant-heat-sink (loss of heat sink without scram), Feldman et al. [6].

In this paper we focus on key measurements of the loss of flow and loss of heat sink experiments, point out the more important design features of EBR-II which provide passive shutdown, and show the results are generally prototypic of a larger metal fuel LMR.

To be inherently safe for all operational accidents, an LMR must also passively limit the consequences of accidental increases in heat generation. A passive power control scheme for inherently safe reactors is being developed and tested. Preliminary measurements and analysis indicate that the passive control scheme could help realize a reactor system truly inherently safe for accidental losses of heat removal or for accidental increases of heat generation.

2. Review of key test results

2.1. Natural circulation tests

The purposes of the natural circulation tests was to demonstrate and provide measured data for the transition from forced to natural convective heat removal regimes. Eighteen tests were run from various shutdown or at-power conditions. All the tests involved a loss of pumping power and a normal reactor scram. Representative results of the most severe test, a loss of all pumps and a simultaneous scram from 100% power, are reproduced in fig. 1 [4]. The temperatures were measured with a representative thermocouple in the incore instrumented subassembly XX09 (Messick et al. [7])

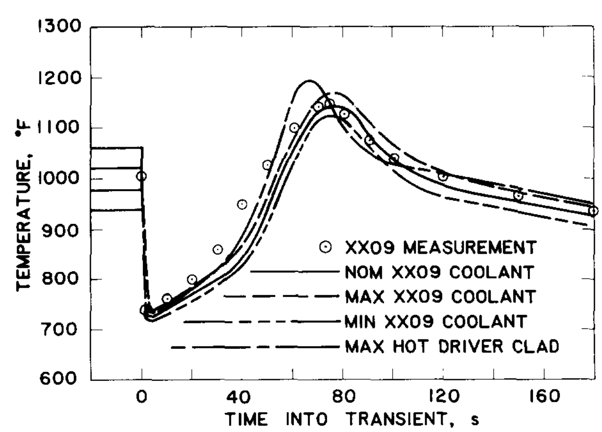

Fig. 1. Natural circulation from 100% power test 17. Pretest predictions and measured in-core temperatures.

near the top of the of the active core. The pretest predictions were made with the NATDEMO (Mohr and Feldman [8]) and HOTCHAN computer codes. The key observations are that

(1) Peak temperatures are low and do not challenge the core (the limiting fuel-clad temperature for normal operation and for anticipated transients is 1319°F) *. The safety limit is sodium boiling at about 1650°F.

(2) The temperatures are predictable. There is good agreement between the pretest predictions and the measured temperatures.

The results strongly suggested that natural circulation should be the principle safety-related means of decay heat removal in future plants, i.e., the use of safety-grade auxiliary pumps or pony motors on backup power supplies is unnecessary.

2.2. Loss of flow without scram (LOFWS)

The loss-of-flow-without-scram tests involve bypassing the normal loss of flow scram function, deenergizing the control rod drive motors and tripping the main coolant pumps. Some 19 LOFWS tests have been run from various initial powers and flows and with various pump rundown times. All the tests have demonstrated passive power reduction caused by reactor feedback mechanisms. Figs. 2 and 3 are pretest predictions and temperatures measured with a representative thermocouple in XX09 near the top of the core for two loss-of-flow-without-scram tests from 100% power [5]. Test 39 utilized a relatively long pump rundown (300 s to pump stop). After the pump stopped, all flow was provided by natural convection. Test 45 (simulating a loss of bulk AC power in EBR-II) had a short pump coastdown (100 s to pump stop). After the pump stopped, flow was provided by natural circulation and supplemented by a battery-backed, electromagnetic, auxiliary pump. The key observation from these test results are:

1. The peak temperatures are relatively mild. With the 300 s coastdown the peak temperatures were within limits for an anticipated event. As shown by Chang [9] there was no reduction in fuel lifetime even for the most severe test, 45.
2. There is good agreement between measured and predicted results. This suggests that the important phenomena governing the passive power reduction and coupled natural convective heat removal have been identified and are adequately modeled.
3. The long-term sodium temperatures at the core exit return to near their initial full-power value. This suggests that the reactor feedbacks tend to match power to the natural circulation flow rate and maintain a constant core ΔT.
4. The transient peak temperatures are significantly reduced by longer pump coastdown times. Comparing

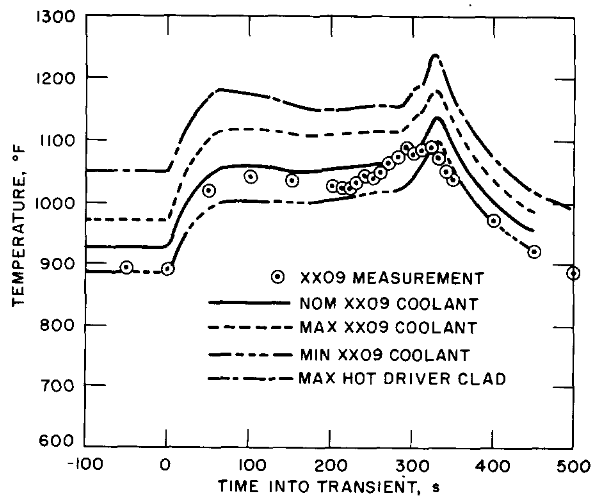

Fig. 2. Loss of flow without scram from 100% power with 300 s pump coastdown. Test 39. Pretest predictions and measurements of in-core temperatures.

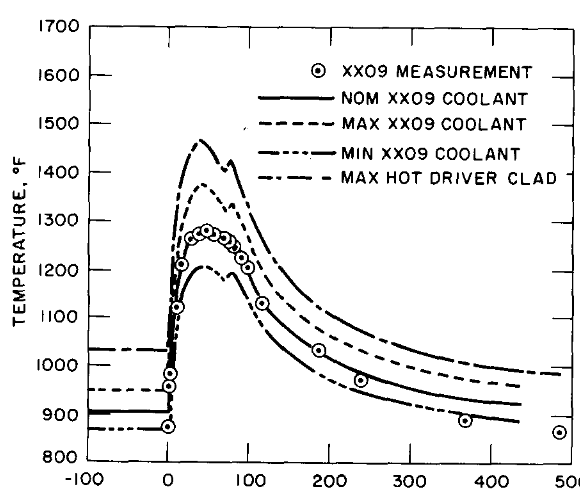

Fig. 3. Loss of flow without scram from 100% power with 100 s pump coastdown time. Test 45. Pretest predictions and measurements of in-core temperatures.

* Use the following factors for converting customary units to SI: °C = (°F − 32)/1.8; 1 gpm = 6.309×10^{-3} m^3/s.

figs. 2 and 3 shows the peak temperature measured in XX09 as 1080°F for a 300 s pump coastdown vs. 1280°F for a 100 s pump coastdown. This shows that peak temperatures can be kept within an acceptable band in a design by selection of pump coastdown rate.

2.3. *Loss of heat sink without scram (LOHSWS)*

Loss of heat sink involves a loss of normal means of transferring heat from the sodium pool to the balance of plant where electricity is generated. Limiting LOHSWS tests were conducted in EBR-II by stopping flow in the secondary sodium loop and thereby essentially stopping the normal heat rejection from the primary pool. There was no automatic or manual control. The reactor was passively shut down by inherent reactivity feedbacks. EBR-II does not have reactor inlet temperature or other "cold end" reactor scram functions to protect against loss of heat sink. The plant does, however, have reactor outlet temperature scrams, but because the power and reactor outlet temperature decrease so fast in a LOHSWS, it was not necessary to bypass these scrams to conduct the experiment. The measured results of the most severe LOHSWS test from 100% power are shown in fig. 4. These include pretest predictions and raw measured data plotted along with the predictions [6].

The key observations from these tests are:

(1) peak reactor outlet sodium temperatures are quickly reduced to levels lower than their normal full power temperatures.

(2) Transient overtemperatures are negligible.

(3) There is good agreement between the measured and predicted temperatures.

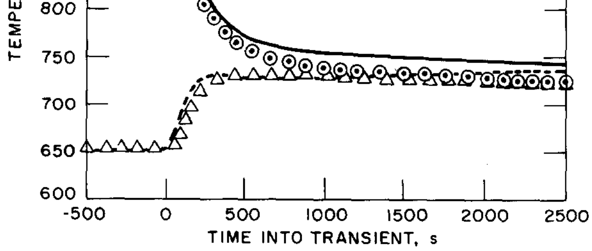

Fig. 4. Loss of heat sink without scram from 100% power. Test B302. Pretest predictions and measurements of reactor temperatures.

3. Key elements of EBR-II inherently safe response and prototypicality

Observing the passive power reduction for the LOFWS tests and the LOHSWS tests and observing the capability to passively remove shutdown heat by natural circulation, there are two overriding questions:

1. What principal features in the EBR-II design promote this inherently safe response?
2. Are the overall test results prototypic, i.e., could a larger LMR of commercial size passively shut down and remove heat as EBR-II did in these tests?

The analysis leading to the tests and preliminary examination of the test measurements identified a few reactor and plant features which dominate the EBR-II response to a LOFWS or a LOHSWS. These features also largely determine the prototypicality of the results.

3.1. *Natural circulation*

The most important parameters governing the peak transient temperatures during the transition to natural circulation were discussed in [4]. They were (1) heat generation (decay power), (2) flow coastdown rate, (3) heat transfer and flow redistribution within the reactor, and (4) component elevations and pressure drops which fix the steady state temperature and flow at which thermal driving heads balance the flow pressure drops.

The peak sodium temperature in the reactor in the natural circulation tests reported in [4] are typical of results in other liquid metal reactors. Although many details are different in reactor designs, U.S. designers generally have elevated the IHX with respect to the reactor and allocated pressure drops within the primary flow circuit to attain 3 to 4% steady state flow at rated core ΔT_0 (EBR-II, a pool type reactor, develops 3.3% flow at rated ΔT. FFTF, a loop type reactor, was predicted to develop about 3% flow at rated ΔT). Pump coastdown is also fixed, considering decay power levels and core heat transfer and flow redistribution, to limit the transient peak temperatures during the transition to natural circulation. The measured peak sodium temperatures in the EBR-II transition to natural circulation from 100% power was about 1150°F for a pump total coastdown time of 40 to 50 s. The measured peak sodium temperatures in the FFTF transition to natural circulation from 100% power was about 1000°F for a total pump coastdown time of about 100s, Cheung et al. [10]. In both cases there is a large margin to the sodium boiling temperature limit (about 1650°F), and in both types of plants the margin could be increased by increasing the pump coastdown time.

3.2. *Loss of flow without scram*

Our analyses and test results have shown that the key plant characteristics which govern the response to a loss of flow without scram are (1) the natural circulation flow, (2) the pump coastdown time and (3) the reactivity feedbacks.

The natural circulation flows have been found to be adequate to remove the residual fission and decay heat after a passive shutdown. The transition to natural circulation is much smoother for a LOFWS than for a loss of flow with a reactor scram. This is because the scram tends to overcool the sodium in the reactor and in the reactor outlet (see fig. 2 [4]). The overcooling temporarily reduces the natural driving heads. In contrast for a LOFWS, the thermal heads tend to build continuously from the steady state to the peak temperature condition. The reactivity feedbacks, as discussed in section 3.2.2, reduce power to match flowrate. Thus, as will be indicated by eq. (4), long term steady state temperatures are not sensitive to natural circulation flow rate.

3.2.1. *Pump coastdown*

The pump coastdown time is a major parameter determining the peak transient temperature in a LOFWS, as already noted in 2.2. The peak transient temperature is principally determined by the imbalance between power and flow during the transient. To keep this peak temperature low, the flow time-constant must be long compared to (1) the thermal response of the core and control rod structures which expand and provide negative feedback, and (2) the nuclear response time which is fixed by the delayed neutron fractions, delayed neutron lifetimes, and reactivity.

The measurements shown in figs. 5, 6, and 7 are for test 30 (individual pump stop times were 70 s and 80 s), test 45 (pump stop time was 100 s), and test 39 (pump stop times were 300 s and 310 s). Test 30, Chang, Mohr, and Planchon [11], was run from 16.7% power and 19% flow; tests 39 and 45 were run from 100% power and flow. The data in the three figures illustrate the effect of pump coastdown time, thermal delays, and nuclear kinetics delays on transient temperatures.

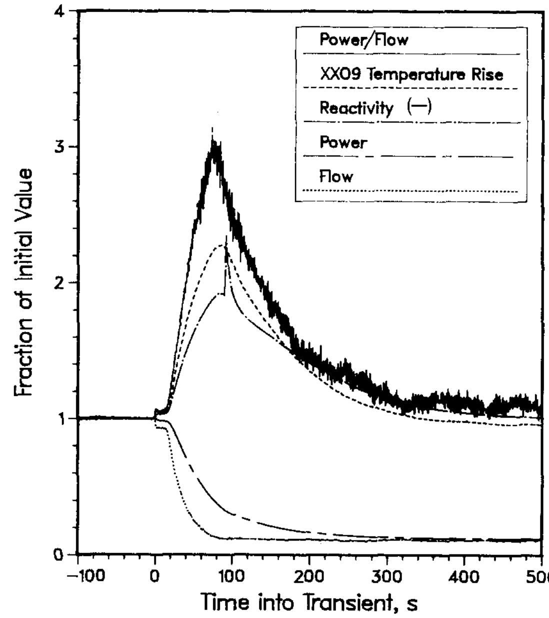

Fig. 5. Comparison of power to flow ratio, core ΔT, and reactivity ratio for loss of flow without scram. Test 30. Pump stop times 70 and 80 s. Initial power 17%. Initial flow 20%. Auxiliary pump off.

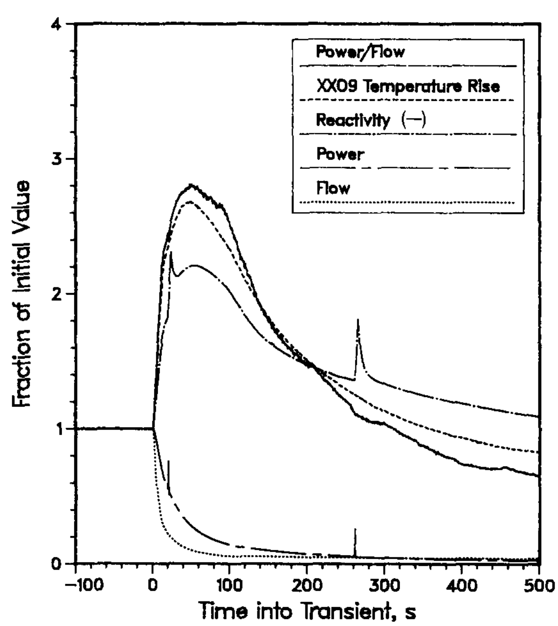

Fig. 6. Comparison of power to flow ratio, core ΔT, and reactivity ratio for loss of flow without scram. Test 45. Pump stop time 100 s. Initial power 100%. Auxiliary pump on battery power.

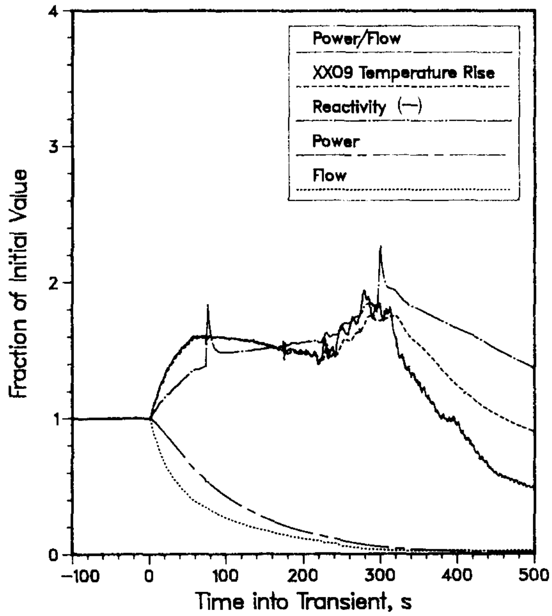

Fig. 7. Comparison of power to flow ratio, core ΔT and reactivity ratio for loss of flow without scram. Test 39. Pump stop times 300 and 310 s. Initial power 100%. Auxiliary pump off.

The power was measured with an ion chamber; flow was measured with the instrumented assembly XX09. The power-to-flow ratio is the simple quotient of the measured power and flow and does not include decay power. The ΔT is the difference between measured temperatures at the inlet and core top position in XX09. The ΔT is normalized by the initial temperature difference. The differences between this curve and the P/F curve are principally due to thermal capacitance and flow and temperature redistribution. The reactivity curve is 1.0 plus excess reactivity (calculated from measured power with inverse kinetics) divided by the power reactivity decrement from zero power to the initial conditions for the test. If reactivity feedbacks were linearly proportional to ΔT then the ΔT curve and reactivity curve would coincide. The inlet temperature changes more in test 39 (long coastdown time) than in the other two tests. Therefore, the reactivity curve for test 39 has an additional large component not proportional to the reactor ΔT.

In selecting a pump coastdown characteristic, one must consider the nuclear time constants of the fuel types to be used. With all else constant, delayed neutron fraction for the uranium fuel in EBR-II ($\beta_c = 0.0067$) provides a time constant about twice as long as in a plutonium fueled reactor ($\beta_c = 0.0034$). Linearization of the point kinetics equations can be used to show that the peak temperature overshoot in a LOFWS is determined by

$$\phi_0 = \lambda\tau\ (\text{P.C.})^2/\text{F.C.}, \tag{1}$$

where λ = one-group delayed neutron precursor decay constant, τ = primary pump characteristic time, P.C. = power coefficient, F.C. = flow coefficient.
If ϕ_0 is large compared to one, the overshoot is small. If ϕ_0 is small, the overshoot is large. For test 45, $\phi_0 = 0.20$.

The fuel-clad temperatures calculated with NATDEMO, shown in fig. 8, further illustrate the trade-off between nuclear time constant and pump coastdown time. The base case curve was calculated for a loss of flow without scram from 100% power in EBR-II. The pump was assumed to stop in 110 s. The case is nearly like test 45. The curve labeled "longer pump coastdown" was calculated like the base case except the time base for the pump coastdown was doubled. The curve $\beta_c = 0.0034$ was calculated like the base case except the delayed neutron fraction was halved, and as a result, the power and flow coefficients, as measured in \$, were roughly doubled. The halving of the rate of flow decrease (governed by pump coastdown rate) has about the same effect on the imbalance in power and flow as does doubling the rate of power decrease (principally governed by the delayed neutron fraction). Therefore, as shown, and as predicted by eq (1) the peak transient temperatures which are governed by the imbalance between power and flow are about the same for the two cases. An important consequence is that the temperature overshoot for a LOFWS in a larger plutonium-metal-fueled plant with comparable feedback and coastdown can be comparable or more favorable than the temperature overshoot measured in EBR-II.

3.2.2. Reactivity feedback

As was pointed out in section 2.2, the long-term sodium temperatures at the reactor exit tend to return to their initial full power value for a loss of flow without scram. This is an important response characteristic that assures the temperatures are not a long-term challenge to reactor structures. Analysis shows that this type of response is caused by the reactivity feedbacks associated with the metal fuel in EBR-II. The analysis also suggests that this response is typical of larger LMR's that have metal fuel. Detailed analysis is used to predict the behavior, but it can be explained by considering a change in reactivity ($\delta\rho$) due to a change in power

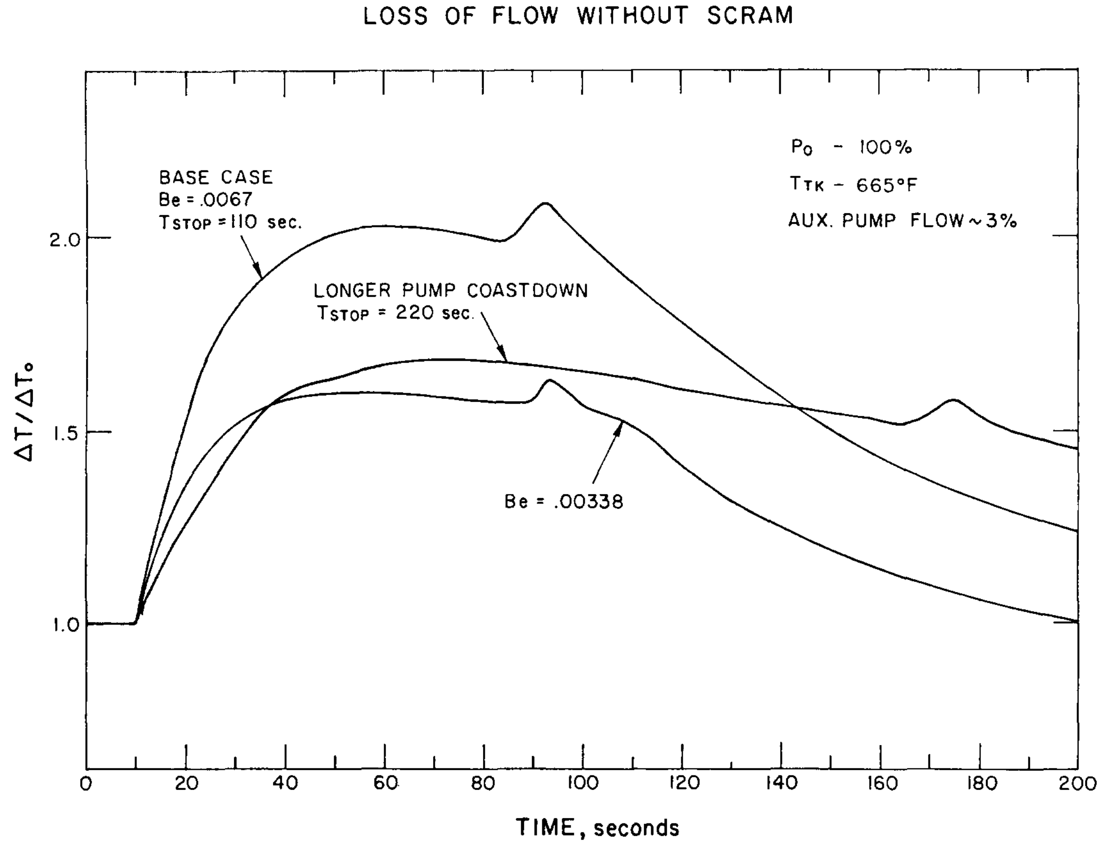

Fig. 8. Sensitivity of LOFWS transient temperatures to pump coastdown time and delayed neutron fraction.

(δP), a change in power to flow ratio ($\delta(P/F)$), and a change in inlet temperature (δT_i). A quasi-static approximation for the reactivity perturbation is:

$$\delta\rho = A\delta P + B\delta(P/F) + C\delta T_i. \tag{2}$$

The detailed components of reactivity feedback presented by Mohr and Chang [12] appear in the above lumped equation as follows: The coefficient A includes reactivity feedbacks proportional to power change, principally Doppler and fuel expansion feedbacks. A is essentially the integral product of fuel to sodium ΔT at rated conditions times the Doppler and fuel expansion reactivity feedback coefficients. The coefficient B includes feedbacks which are proportional to the power-to-flow ratio or the flowing sodium temperature rise across the core. These feedbacks include the Doppler and fuel expansion feedback and the more dominant feedbacks due to thermal expansion of the sodium and steel in the core, reflector, and control rod drivelines. The reactivity due to bowing of core assemblies is considered to be a nonlinear function of core sodium ΔT and is also included in B. The quantity $(A + B)$ is the power reactivity decrement (PRD) – the reactivity addition necessary to raise power from zero power hot critical to rated conditions ($\delta P = 1.0$) with constant inlet temperature and rated flow $F = 1.0$. In EBR-II the PRD is about 0.30$. The inlet temperature coefficient C includes feedback from Doppler, fuel and core expansion, and feedback from thermal expansion of the inlet reflector and core support grid.

For a LOFWS assume the inlet temperature is constant, i.e., $\delta T_i = 0$. Then if there is no control rod position change (except for differential thermal expansion between the control rods and the core), the net reactivity change in going from the initial steady state (o) to the final equilibrium state (f) is:

$$\delta\rho = A\delta P + B\delta(P/F) = 0. \tag{3}$$

Initially, for a loss of flow, F decreases and negative reactivity is inserted from the $B\ \delta(P/F)$ term. Power decreases and introduces positive reactivity from the $A\ \delta P$ term. In the end a steady state evolves in which the two terms balance. The amount the P/F has to increase depends on the A/B and the final flow rate F_f. Noting that the sodium temperature rise in the reactor, ΔT, is proportional to P/F and solving for $\Delta T_f/\Delta T_0$ gives:

$$\frac{\Delta T_f}{\Delta T_o} = \frac{1 + (A/B)F_o}{1 + (A/B)F_f} \tag{4}$$

Estimates of A/B for EBR-II (20 MWe) and larger metal and oxide fuel reactors have been made. Solutions for eq. (3), taking $F_o = 1.0$ and $F_f = 0.02$ (conservatively), are given in table 1.

Even though the A/B values in table 1 are preliminary, further refinement is unlikely to change them sufficiently to alter the important conclusions that follow from examination of the table:

1. The value of A/B is relatively small for a metal-fueled LMR; P/F feedback is dominant.
2. Therefore, the asymptotic LOFWS temperature from tests run in a small metal-fueled LMR (EBR-II) are prototypic of results to be expected from metal-fueled LMRs of all sizes of commercial interest.
3. The reactor ΔT at the end of a LOFWS in a metal-fueled LMR is 20% to 50% greater than the initial normal ΔT.
4. The value of A/B is relatively large for an oxide-fueled LMR due principally to low thermal conductivity of the fuel and the dominance of the Doppler feedback.
5. Therefore the LOFWS temperature from the EBR-II tests is lower and not directly comparable to the temperatures expected in a large oxide fueled LMR. (EBR-II test data support oxide core design indirectly through computer code validation).
6. Asymptotic temperatures are insensitive to variations in F_f. Power adjusts to the final flow rate so temperature is relatively insensitive to design uncertainties which affect natural circulation flow rate.

3.3. Loss of heat sink without scram

The analysis and experimental results have shown that the single most important set of parameters that determine the temperature response to the loss of heat sink without scram is the reactivity feedbacks. The reactor system heat capacities that together with reactivity feedbacks govern the transient peak temperature are also important.

3.3.1. Pool heat capacity

The peak transient temperature is limited in a loss of heat sink without scram if the rate of change of driving temperature at the reactor inlet is slow compared with the response time of the reactor. Our analysis, partially discussed in [6], indicated that the temperature overshoot was not significant in EBR-II for a wide range of loss of heat sink initiators (loss of feedwater or loss of secondary flow) or a wide range of tank mixing. Further detailed analysis with geometries of other plants is desirable but in the limit, a serious transient temperature overshoot can be avoided if the sodium volume and mixing in a cold pool (or inlet plenum in the case of a loop plant) results in a time response of sodium temperature at the reactor inlet which is slower than: (1) the thermal time response of the reactor support structure, the lower reflector or blanket, and other reactor structures which heat up and expand to provide shutdown reactivity; and (2) the nuclear time response to negative reactivity.

3.3.2. Reactivity feedback

Integral reactivity feedbacks determine the long term steady state temperature that will reduce the fission power to zero. This is explained by referring back to eq. (2). Applying the equation to a loss of heat sink without scram, the inlet temperature δT_i increases due to the

Table 1
Comparisons for LOFWS

	Metal fuel				Oxide fuel		
	20 MWe (EBR-II)	100 MWe	330 MWe	1000 MWe	100 MWe	330 MWe	1000 MWe
Reference		[13]	[13]	[14]	[13]	[13]	[14]
A/B	0.1	0.38	0.44	0.69	1.86	2.38	3.38
$\Delta T/\Delta T_0$	1.2	1.37	1.43	1.66	2.76	3.23	4.10
final flow F_f equal 2%							

Table 2
Comparison for LOHSWS

	Metal fuel				Oxide fuel		
	20 MWe (EBR-II)	100 MWe	330 MWe	1000 MWe	100 MWe	330 MWe	1000 MWe
$(T_{if} - T_{io})$, °F	78	184	214	270	628	788	1012

Ref. [13] for 100 and 330 MWe cases; ref. [14] for 1000 MWe cases.

loss of heat sink, and flow F stays constant at F_o. Defining the initial conditions as $P_o = 1$ and $F_o = 1$, then in the long term limit

$$\delta\rho = A\delta P + B\delta(P/F_o) + C\delta T_i = 0$$

or

$$(T_{if} - T_{io}) = (A + B)/C. \tag{5}$$

Physically interpreted, the inlet temperature rises so that the total negative reactivity from the inlet temperature change is equal to and balances the power reactivity decrement – the positive reactivity from reducing power from 100% conditions to zero power. Table 2 above compares the primary tank temperature rise, $T_{if} - T_{io}$, for LOHSWS accidents in the same LMRs as cited in table 1; A, B, and C values are results of preliminary analyses done on a consistent basis.

In EBR-II the power reactivity decrement was measured to be about 0.30 \$ and the inlet temperature coefficient was about 0.004 \$/°F. Thus a "back of the envelope" asymptotic temperature rise of 75°F is predicted. The temperature rise measured at 2500 s as shown in fig. 4 was 70°F.

4. Inherent safety for transient overpower – inherent control

Thus far, test results have shown that EBR-II inherently preserves the global balance between heat generation and heat removal by passively reducing power for a loss of flow or a loss of heat sink. It remains to be shown that the balance can be inherently preserved for failures in power control equipment, i.e., failures causing the traditional rod runout accident. Therefore, to complete the inherent safety investigation one must also show (1) that a maloperation or failure of an active control system cannot override a passive shutdown for LOFWS or LOHSWS, and (2) that a transient overpower accident without scram (TOPWS), such as a rod runout, can be passively accommodated. A novel approach to control therefore appears to be necessary if truly inherent safety is to be realized for all operational accidents.

A passive power control scheme that is inherently safe is suggested by measurements and analysis completed as part of the test program. The elements of this control scheme are as follows:

(1) *A limited control rod worth that would minimize the capability of overriding inherent shutdown mechanisms and minimize the consequences of a rod runout. This would necessitate a limited reactivity swing due to burnup during a fuel cycle.*
This may be accomplished by designing internal breeding to be as large as possible, consistent with other design, operating, and economic constraints, including those involving reprocessing. There are two main ways of increasing internal breeding. These are to harden the neutron spectrum by minimizing the amount of moderating material in the reactor, i.e., using metal rather than oxide fuel; and to increase the ^{238}U content in the core.

(2) *Inherent reactor power control achieved with changing sodium inlet temperatures.*
Eq. (1) shows that the reactivity feedbacks cause power to follow inlet temperature for a constant primary flow rate.

$$\delta P = \frac{C}{A + B}\delta T_i. \tag{6}$$

Thus, if the plant's "cold end" temperatures are allowed to naturally drop for increasing steam load demand, then the reactor power will passively follow steam load. This mode of control, called "steam load control", appears to be feasible based on (1) an analysis to determine the profile of steady state plant temperatures associated with the control scheme, section 4.1, and (2) results from a transient steam load change test, section 4.2.

(3) *Inherent reactor power control achieved with changing primary flowrate.*
Eq. (4) shows that reactivity feedbacks cause reactor

power to follow reactor flow and keep the reactor δT nearly constant. Rearranging eq. (3) to give power as a function of flow, and taking $P_o = 1$ and $F_o = 1$,

$$P = F\frac{1 + A/B}{1 + (A/B)F}. \tag{7}$$

This mode of control, called "flow control", also appears to be feasible based on (1) an analysis to determine the profile of steady state plant temperatures associated with the control scheme, section 4.1, and (2) results from flow perturbation tests, section 4.3.

These modes of control are possible in a reactor that has a relatively small power reactivity decrement ($A + B$), and in which the A/B ratio is small. This is true with metal fuel which operates relatively cool and has a smaller Doppler effect. It should be noted that large reactivity feedback due to large Doppler effects translates to large reactivity required for control, resulting in high control-rod worths. Reduction in Doppler feedback reactivity, which is an important aspect of inherent safety in response to LOFWS and LOHSWS scenarios, is also important in TOP events if a control strategy is utilized which takes advantage of low Doppler to minimize control rod worth. This benefit can be further realized by control schemes which utilize variation in primary flowrate for partial control of reactivity.

4.1. Steady state temperatures for passive control schemes

The feasibility of the steam load control and flow control schemes has been investigated by calculating the swing in plant temperatures, pressures, and flow necessary to passively control reactor power. The plant state for the normal EBR-II control scheme has also been calculated as a function of power for reference and comparison.

Referring to fig. 9, the normal control scheme in EBR-II is as follows:

1. Control reactor power (POW) either automatically or manually with control rods (control C_1)
2. Keep reactor flow (F_1) constant by controlling the pump rpm to a constant setpoint (control C_2)
3. Control (C_3) secondary flow (F_2) either automatically or manually to keep the reactor inlet temperature (T_1) constant
4. Control the steam pressure (P_1) at the throttle automatically with the throttle valve (control C_4) and/or the steam bypass valves (control C_5) for constant steam pressure
5. Keep steam drum level and feedwater temperature constant.

The calculated temperatures for the normal control scheme over a load range from 25% to 100% of 60 MWth are shown in fig. 10. As shown, the cold end temperatures of the plant are nearly constant. The hot end temperatures increase in proportion to power. The total control rod reactivity necessary to change power from 25% to 100% power was 20.5 ¢. The secondary flow decreased from 100% at 100% power to 42% at 25% power to maintain a constant reactor inlet temperature of 700°F.

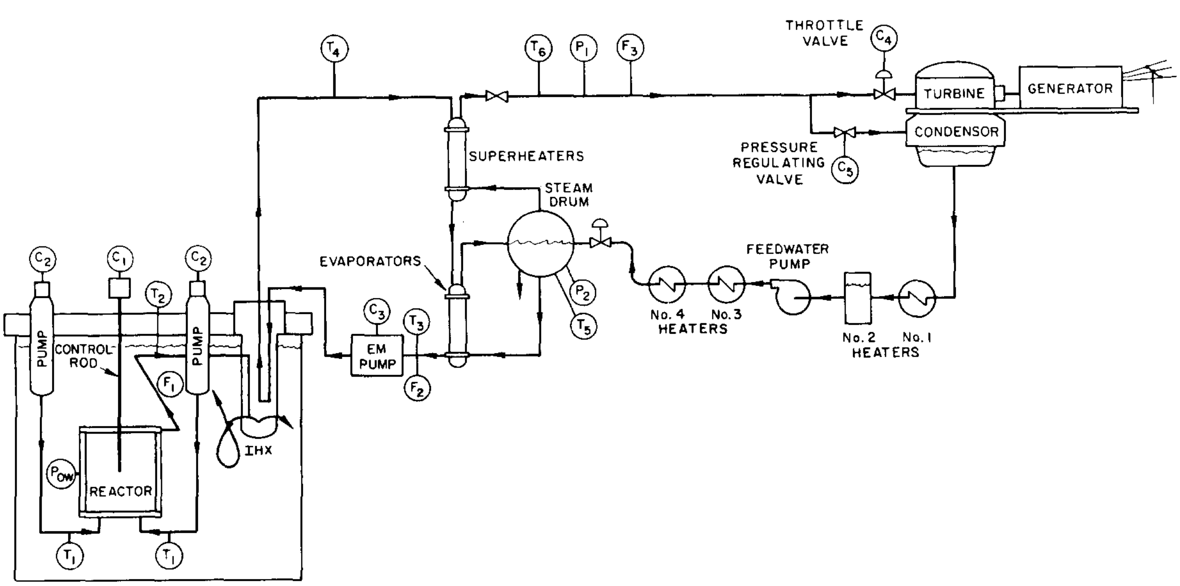

Fig. 9. EBR-II process, monitoring, and control diagram.

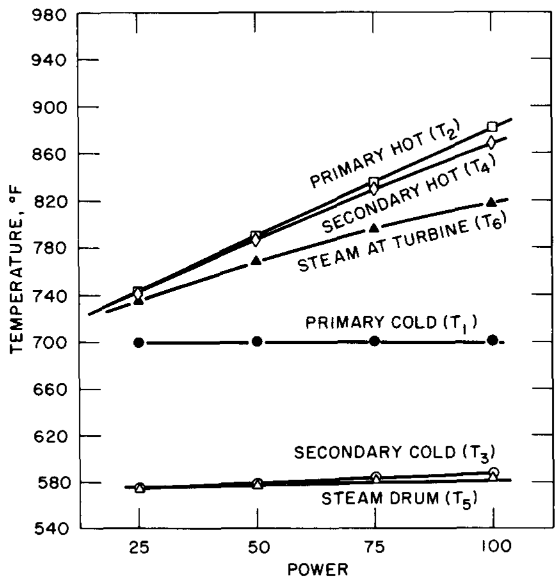

Fig. 10. Temperature as a function of power for normal control in EBR-II.

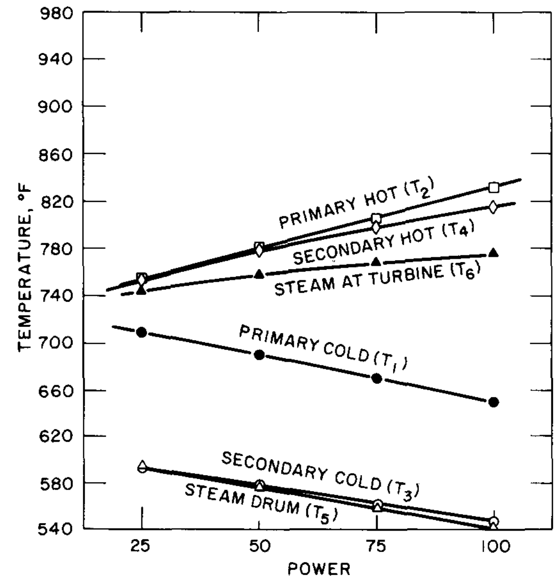

Fig. 11. Temperature as a function of power for steam load control in EBR-II.

In the steam load control mode, the throttle (control C_4) was assumed to be opened and the secondary pump was controlled (control C_3) to give decreasing steam drum pressure (P_2) and temperature (T_5) and decreasing reactor inlet temperature (T_1) as a function of increasing load. Fig. 11 shows the calculated temperature profile over a load range from 25% to 100% for one variation of this type of control scheme. The 60°F decrease in tank temperature passively controls the reactor over a range of 25% to 100% power.

In this particular scheme the throttle and secondary pump were operated, somewhat arbitrarily, to decrease the steam drum saturation temperature by about 52°F – about the same as the reactor inlet temperature decrease. An optimum control scheme would likely combine both cold-end temperature swings, steam pressure swings, and flow control of the primary and secondary pumps.

The profile of plant temperatures for the flow control scheme were calculated based on the following assumptions:
1. Primary flow (F_1) was ramped (control C_2) in direct proportion to the desired load.
2. Secondary flow (F_2) was controlled (control C_3) to maintain a constant reactor inlet temperature.
3. The throttle valve was controlled (control C_4) to maintain constant throttle pressure (P_1).
4. Control rods were adjusted (control C_1) to maintain constant reactor ΔT (T_1 and T_2). This variation to absolute flow control (in which power passively responds to flow changes per eq. (7)) was made to simplify the NATDEMO calculations.

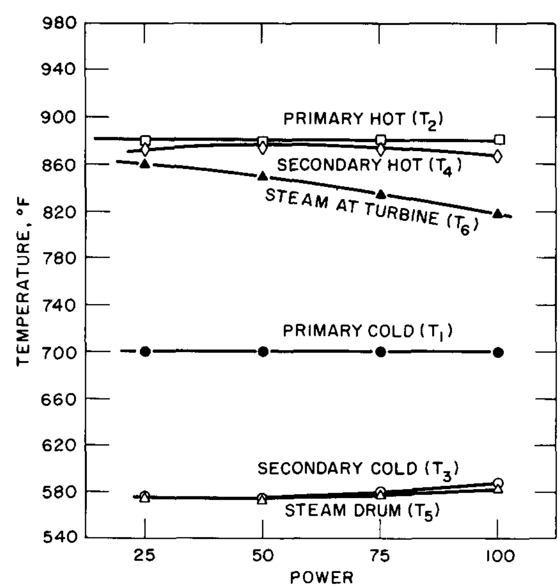

Fig. 12. Temperature as a function of power for flow control in EBR-II.

The results of the calculations are shown in fig. 12. As is seen, the sodium temperatures in both the primary tank and secondary loop are nearly constant. The steam temperature at the throttle droops about 40°F with load. This is about 1/2 of the steam temperature change for the normal control scheme as shown in fig. 10. Only about 1.7 ¢ of control rod reactivity was used to keep the core ΔT constant. The secondary flow was essentially proportional to the load.

In conclusion, these calculations show that plant states that support inherent control can be achieved in EBR-II. The two passive control schemes, steam load control and flow control, result in steady state temperatures within the EBR-II design envelope and result in essentially no required control rod reactivity to maneuver from 25% to 100% power.

4.2. Steam load perturbation tests

The ability of the plant and reactor to passively follow transient steam load demands is proved by the results of a series of steam pressure perturbation tests. Three of these tests were conducted from a normal 75% power level with initial steam header pressure of 1265 psig. The tests were controlled by automatic controllers operating the bypass pressure regulating valve (control C_5 in fig. 9). The three tests involved (1) automatically reducing the steam header pressure 100, 200, or 400 psi on linear ramps over a 2 min period, (2) automatically holding the pressure constant until a steady state evolved (about 45 min), (3) automatically ramping the pressure back to its initial value and (4) holding until steady state reactor power was attained. Primary and secondary flow were held constant, reactor power was allowed to freely respond to reactor inlet temperature changes, and steam drum level and feedwater temperature were controlled to a constant setpoint.

Figs. 13, 14 and 15 show how an increase in steam power is matched passively by increased reactor power. The data on the figures are for test B402 in which the steam header pressure was reduced from 1265 to 865 psig over a 2 min period. This resulted in a steam drum pressure decrease of 367 psig and a reduction in steam drum saturation temperature of about 40°F. The decrease in steam drum temperature is directly reflected in a decrease in sodium temperature at the evaporator outlet, the IHX secondary inlet, and the reactor inlet, as shown in fig. 13. The 40°F decrease in steam drum saturation temperature results in an approximate 40°F decrease in secondary loop cold leg temperature. The IHX, with a primary to secondary flow ratio of almost 2, along with the reactivity feedbacks result in a reactor inlet temperature decrease of about 12°F. Reactor power increases, and the temperatures at the core outlet, the IHX secondary outlet, and the superheater steam outlet increase as shown in fig. 14. The initial decrease in steam temperature is caused by the increased steam flow.

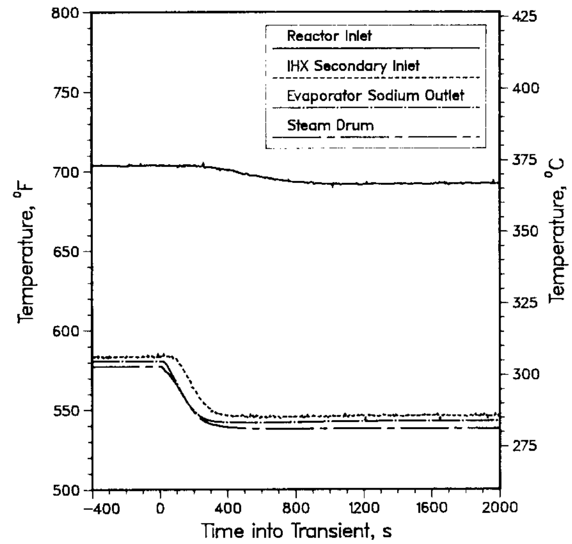

Fig. 13. Cold end temperature transients in EBR-II for a steam load increase. Test B402.

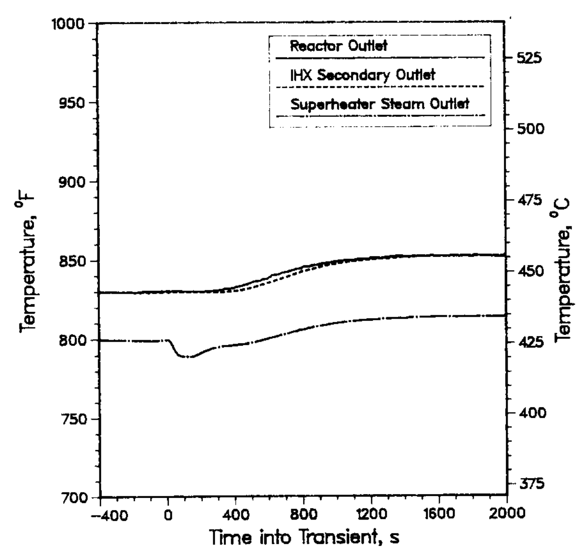

Fig. 14. Hot end temperature transients in EBR-II for a steam load increase. Test B402.

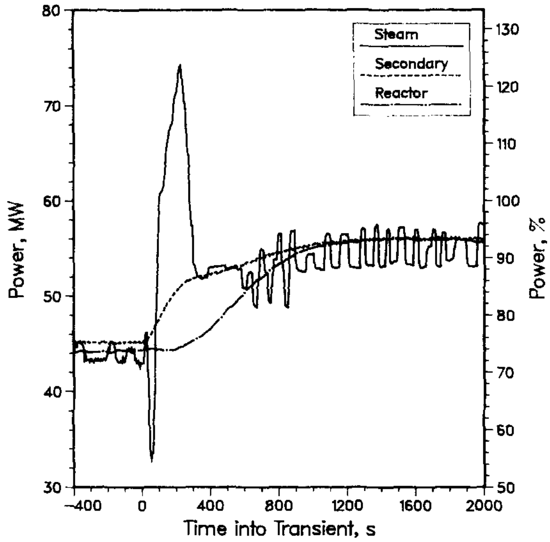

Fig. 15. Load transients in the EBR-II steam generator, secondary loop and reactor for a steam load demand. Test B402.

The transient load-following capability of the plant is illustrated with steam generator power, secondary loop power and reactor power as shown in fig. 15. The steam generator power is calculated with steam pressure and temperature measured at the exit of the superheater, and feedwater flow and temperature measured at the entrance to the steam drum. The secondary loop power was calculated with measured flow and temperatures. The calculations were made on-line with the plant computer using algorithm normally used for steady state calorimetric calculations. Therefore, these powers, while accurate in the steady state are approximate for the transient and must be properly interpreted. For example, steam flow is assumed to equal feed flow in the calculation. This plus the cycling of the feedwater regulating valve as the transient is initiated, leads to the anomalous dip in calculated steam power at the start of the transient. Notwithstanding the approximations, the results illustrate the time phasing and magnitude of the rate of change of load. As shown in fig. 15, opening of the pressure regulating valve leads to large transient steam generator powers with rates of change exceeding 20% per minute; however, the reactor responds in a slow, deliberate way with no overshoot and with a rate of change of reactor power that is less than 1% per minute.

In summary, the measurements from the steam load perturbation tests show that an increase in steam load results in a decrease in the sodium temperature at the reactor inlet and a passive increase in reactor power. For an increase in steam power from 75% to 93%, the reactor inlet temperature decreased 12°F and reactivity feedback passively drove up reactor power from 75% to 93% to balance the steam load demand. The reactor response was stable and well behaved.

4.3. Flow perturbation tests

Flow perturbation tests demonstrated the ability of the reactor power to passively follow transient changes in flow. Measurements from one of these tests (run from 70% power and 70% flow) are discussed in [4].

The control scheme for these tests was as follows:

1. Primary flow was ramped to 130% of the initial value and held until a steady state evolved (about 17 min) and then returned to its initial value.
2. Control rods were not used. Reactor power was allowed to freely respond to changing coolant temperatures.
3. Secondary flow was controlled to keep the primary cold pool (reactor inlet) temperature constant.
4. The steam plant was controlled to constant setpoint for steam header pressure, steam drum level and feedwater temperature.

As shown in [4], the measured reactor power follows flow smoothly and predictably. For a 130% flow increase, the power increased to about 127%. Secondary flow was also increased to about 130% to keep the tank (reactor inlet temperature) constant.

The sodium and steam plant temperature changes are small for this type of control scheme as shown in figs. 16 and 17. The hot-end sodium temperature change in the primary and secondary systems was less than 30°F. The cold-end temperature swing was less than 10°F.

In summary, the measurements from the flow perturbation tests show that reactor power passively responds to transient changes in flow in a smooth, predictable way. The primary, secondary and steam flows were increased by about 130% of their original value (from 70% to 91%) and reactor power passively responded (via the reactor feedback coefficients) to increase power by about 127% (from 70% to 80%). The accompanying reactor plant temperature transient was mild. It is significant that the hot-end temperatures *decrease* with increasing power. This suggests a very safe situation in which equipment failures that would lead to transient overpower situations also lead to *lower* reactor temperatures, which are less challenging to core and plant structures.

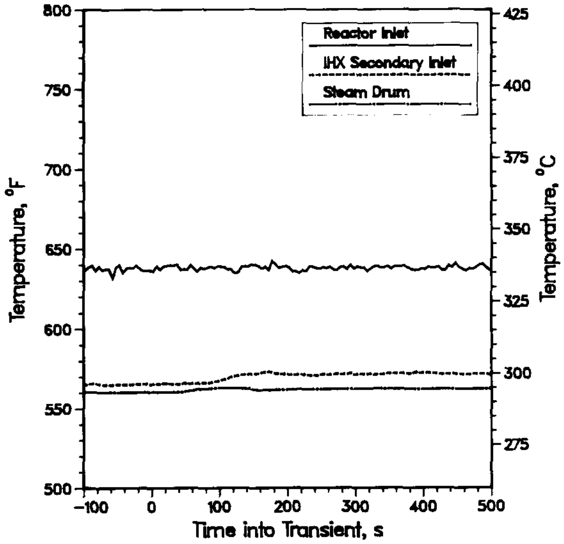

Fig. 16. Cold end temperature transients in EBR-II for an increase in primary, secondary and steam flow. Test 25.

4.4. Safety considerations with inherent control

The preliminary analysis and experimental measurements suggest that an inherent control scheme which does not use control rods is feasible, and that with this scheme, overpower from operational incidents such as rod runout, would be eliminated or reduced to a trivial magnitude. It may be possible to limit hypothetical sources of reactivity insertion, such as core compaction from a seismic event, by core design so that the consequences are not a safety problem. However, the emphasis in this paper is on operational accidents.

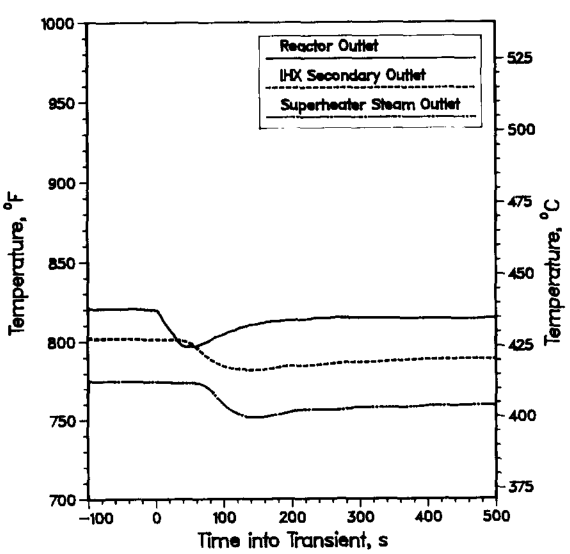

Fig. 17. Hot end temperature transients in EBR-II for an increase in primary, secondary and steam flow. Test 25.

If a primary pump increases flow excessively, then the passive feedbacks would cause reactor power to nearly follow flow (see eq. (7)). This type of accident could be caused by failure of instruments, failure of control systems, or by human error. With the use of fault-tolerant control systems this type of accident could be very improbable. However, it appears possible to design the plant to passively limit the flow runout consequences and therefore passively accomplish the safety function of protection against overpower.

The temperature transient for a pump runout is significantly different from a transient overpower accident driven by control rods. In a rod runout, flow is constant, so sodium temperature increases linearly with power. In the pump runout, power follows flow so the sodium temperature decreases with increasing power. *Thus the margin to the sodium boiling temperature increases* with increasing power, as shown in eq. (4). The amount of power can be inherently limited by pump capacity. In the long term the overpower is also limited by heat exchanger and steam generator capacity.

If the steam power demand increases by a large magnitude, then the reactor inlet temperature coefficient could tend to increase reactor power excessively. A spurious steam power demand could be caused by failure in the turbine or steam bypass valve controls, or by inadvertent opening of a steam relief valve. Other large transient steam demands are also conceivable – for example consider a pipe break in a steam generator system and the consequent drum/module blowdown, or consider an operator error which results in suddenly bringing a cold secondary loop of sodium on line. Both events, depending on details of the design, could cool the inlet of the reactor and appear as a load demand.

It is likely that fault-tolerant adaptive control systems could be used to make the excessive steam load demand accident very improbable. However, it appears that a plant can be designed so that the consequences of the excessive load demand is inherently limited without relying on adaptive control to perform a safety function. For example, the maximum long term steam power demand can be set by limiting the capacity of throttle and pressure regulating valves and feed pumps. The consequences of large transient power demands are limited by heat capacitance of sodium and structure in the cold pool.

4.5. Design and operational aspects of passive control

The passive power control schemes outlined have the possibility of (1) greatly simplifying the safety-related control and protection systems, and (2) allowing uninhibited deployment of state of the art automatic controls for the balance of plant system.

The experimental results have shown passive shutdown for loss of flow and loss of heat sink events. Analysis and test results discussed in previous sections suggest a similar inherently safe response for transient overpower events. The analysis also indicates this response could be expected from large LMR plants. These results suggest that the principal first line of safety protection could be passive shutdown and that the active scram systems could be backups. This could greatly simplify the instrumentation, logic, and scram system design, maintenance, and operation. It could also help reduce spurious scrams.

Plant automation has not yet been fully implemented in U.S. nuclear plants, in part because of concerns for reliability and licensability. The designers of such systems are reluctant to accept the open-ended risk of failures in nuclear equipment. If reactor power is controlled passively and the plant is designed to passively accommodate failure of any controllers, then the controllers do not need to be safety grade. Consequently there could be rapid deployment of many computer-based state of the art control schemes for control of flow, control of steam load, and control of various other components in the balance of plant. The modular plants now being designed [1,2], however, will benefit greatly from automation, and the technology is being seriously considered. In fact, both the SAFR and PRISM plants are to be totally automated in order to enhance operating reliability and to reduce operating costs.

EBR-II is being modified to support testing and demonstration of advanced control and diagnostic systems appropriate to the advanced reactor designs [1,2[as well as to space-nuclear power systems. This work is especially relevant when it can be coupled to the passive-control approach described earlier. Already, much has been accomplished. A computer-driven automatic control-rod drive system has been in operation for several years, allowing for shaped power-transients at rates of power change up to 10%/sec. Also, a major effort is underway to develop and apply formal methods of analysis to the fault-tolerant aspects of a C.S. Draper computer designed for a safety-system at EBR-II. The formal-analysis techniques have a potentially large area of application to design in general, taking advantage of many years of development of automated reasoning software at ANL. Other major efforts include installation of microcomputers for most balance-of-plant control functions, capable of being networked to a central-control computer. Also installed at EBR-II is a new large computer for data acquisition and a network including two VAX-11/750's, SUN computer, and IBM AT/PCs, for engineering development and testing of advanced control and diagnostic systems. This network coupled with the EBR-II plant will allow comprehensive tests for a wide range of control strategies as well as for the underlying technology.

5. Summary and conclusions

The data from the EBR-II inherent safety demonstration tests and the extrapolation to larger commercial plants suggest a fundamentally different approach to reactor control and reactor safety. In this approach, the central control and safety function – maintaining a balance between heat generation in the reactor and heat removal from the reactor – would be accomplished passively by inherent physical processes.

The EBR-II test series has shown inherent safety for a wide range of loss of flow without scram (LOFWS) and loss of heat sink without scram (LOHSWS) accidents. In each of the final LOFWS and LOHSWS tests, the reactor passively shut itself down from 100% power. There was no operator intervention or automatic action of the shutdown system. Core temperatures were mild, and no fuel was damaged. There was good agreement between pretest predictions and measurements of in-core temperatures. This strongly suggests that the important phenomena governing the passive shutdown have been identified and adequately modeled in the NATDEMO and HOTCHAN computer codes.

Inherent safety is also suggested for overpower accidents. A passive method for reactor power control is being developed to eliminate the need for control rod reactivity additions for load control or for burnup. Reactor power would be controlled inherently by primary flow and steam load. The final objectives of the ongoing testing and analysis is to show that the effects of any likely equipment failure would be inherently mitigated so that overpower would not be a safety problem. The analysis and results presented are encouraging and support the feasibility of inherent control.

Analysis also indicates that the characteristics which support inherent safety and inherent control are also applicable, with smaller margins, to larger metal fueled LMRs, at least up to 1000 MWe (see tables 1 and 2).

Therefore, it seems clear that core disruption as a result of any of the large class of loss-of-flow accidents or loss-of-heat-sink accidents can be eliminated from any reasonable safety consideration. Preliminary results indicate that operational overpower accidents can also be eliminated as HCDA initiators. The test results therefore strongly support the position that HCDAs are not design basis accidents. Further, these results, together with other safety development delineated by Till and Chang [15] strongly suggest that third level design margins are not necessary to accommodate the effects of an HCDA.

The implications of the tests are perhaps even further ranging. The effectiveness of the passive shutdown and decay heat removal measured in EBR-II suggests that the inherent processes could be the basis for the design of the first and second levels of safety and that a much simplified automatic reactor shutdown system could be a diverse backup for the third level of safety and accommodate extremely unlikely or hypothetical equipment failures. This approach is based on two experimental observations as follows: (1) pump coastdown time can be selected to keep peak reactor temperatures for LOFWS less than limits for normal operational transients, and (2) peak reactor temperatures for a LOHSWS were lower than the temperatures during normal operation. A similar situation is expected for the TOPWS.

Continued emphasis on inherent safety and inherent control could also bring about important improvements in plant operability and economics. The demonstrated ability to passively accommodate a loss of the normal heat sink and to passively remove shutdown heat with natural circulation strongly supports current design approaches. These designs rely on natural circulation for safety related shutdown heat removal. The loss-of-heat-sink-without-scram measurements show the secondary loop and balance of plant are not necessary for any safety function and therefore support design choices of a non-safety-grade balance of plant. By deleting safety-grade pony motors, power supplies, and other complex active equipment and by using a non-safety-grade balance of plant designs can be simpler and less expensive to construct, maintain, and operate.

The end result of current inherently safe development and design efforts could therefore be a plant that is not only safer, but is simpler, more easily operable, more readily accepted, and more economically competitive.

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