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ANALYSIS OF LOSS-OF-FLOW TRANSIENTS IN A POOL-TYPE LMFBR USING SSC-P

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ABSTRACT

In order to have a general analytical capability for the safety evaluation of any proposed LMFBR system, the USNRC is sponsoring the development and validation of computer codes for both pool- and loop-type plants. The computer code for pool-type LMFBRs is designated SSC-P. This paper is concerned with the application of SSC-P to simulate loss-of-flow accident transients in a pool-type LMFBR. The models required for dynamic plant simulation are briefly highlighted. The system response is calculated for (i) a complete loss of electric power event, with scram, leading the plant into buoyancy-induced natural circulation, (ii) a protected pipe rupture accident in the primary pump discharge line, and (iii) an unprotected loss of off-site power event. For the last case, the predicted results from SSC-P are compared with the published results of Phenix behavior by NOVATOME.

INTRODUCTION

Based on current liquid metal-cooled fast breeder reactor (LMFBR) technology, a commercial LMFBR power plant could be either of the loop-type or pool-type design. In the latter, the entire radioactive primary cooling system, including reactor, is located in a single large tank. Judging from the successful experiences with pool systems, both in the U.S. (EBR-II), and in Europe (e.g., Phenix, PFR), the pool design is a viable alternative to the loop design as demonstrated in recent EPRI design studies [1, 2].

In order to provide the U.S. Nuclear Regulatory Commission with a general capability for the safety analysis of pool-type LMFBR plants, SSC-P [3], a version in the Super System Code series [4], was developed at Brookhaven National Laboratory. SSC-P is a generalized computer program designed to analyze the system response to a malfunction anywhere in the heat transport system of the plant. This would cover a variety of normal, off-normal and accident transients where the range of real time simulation extends beyond several seconds, and where the thermal-hydraulic response of the core is tightly coupled with that of the heat transport system through flow rate and temperature of entering coolant, and the driving pressure for core flow. These transients are important for both system design and safety evaluation since the LMFBR design requires adequate heat removal without core voiding or fuel/cladding failure under all conditions (exclusive of HCDA scenarios).

This paper is concerned with the application of SSC-P to simulate loss-of-flow accident transients in pool-type LMFBRs. The essential features of the pool design are briefly described. The models used in SSC-P are then briefly highlighted. Finally, the system response is calculated for the following transients:

i. A postulated loss of electric power, (LOEP) transient, or station blackout, with scram, leading to buoyancy-induced natural circulation in the heat transport system;

ii. A protected loss of piping integrity accident in the primary system; and

iii. A loss of off-site power event, with failure to scram.

In all these events, the core coolability could be endangered and needs to be assessed. The results are discussed in terms of the dynamic behavior of important system variables.

SYSTEM DESCRIPTION

The basic configuration for the primary system being analyzed is shown in Fig. 1. The coolant exiting the core enters an open pool; during steady state operation, this open pool temperature is at approximately the reactor mixed mean outlet temperature. The sodium in this hot pool is separated from the cooler sodium in the pump suction region of the tank by an insulating barrier. In general, the liquid levels in the hot and cold pools are different; this level difference accounts for the hydraulic losses and gravitational heads occurring during flow through the IHX.

This basic configuration has been implemented in operating prototype plants, (e.g., Phenix and PFR), and will also be implemented in the Superphenix and CFR commercial plants. Furthermore, all U.S. EPRI-sponsored pool-type prototype large breeder reactor (PLBR) design studies have been based on this concept.

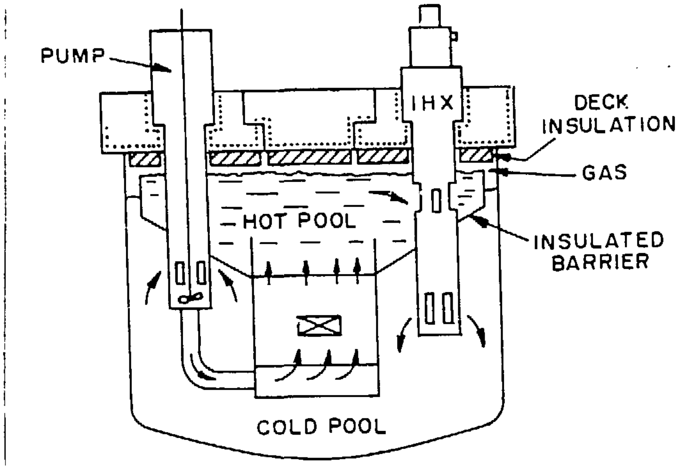

Fig. 1 Primary system configuration for a pool-type LMFBR

MODELS

Primary Coolant Dynamics

In the pool-type designs under discussion, where both hot and cold pools are free surfaces, there is direct mixing of the coolant with these open pools prior to entering the next component. Thus, two different flow segments have to be modeled to characterize the coolant dynamics of the primary system. During steady-state operation, the two flow rates are related by a simple algebraic equation. During a transient, however, the flow in the up-leg from the pump would respond to the pump head and losses in that circuit (including losses in the core); the IHX flow would respond to the level difference between the two pools, as well as losses and gravity gains in the unit. The gravity gain could be significant for low-flow conditions, particularly if the IHX becomes overcooled due to a mismatch of primary and secondary flows.

For symmetric transients, such as a loss-of-electric-power (LOEP) event, all parallel components can be expected to behave identically, and only one flow equation needs to be modeled for each set of parallel components. However, in cases of asymmetric events, it is necessary to distinguish between the components that are directly affected by the postulated accident from those that are not. Examples of such events are a pipe rupture in a pump discharge line to the reactor, a single pump malfunction or a malfunction in an intermediate circuit causing the affected IHXs to behave differently from the others.

Liquid levels in the pools are determined when mass balance equations for the pools, accounting for the combined effect of coolant thermal expansion, and flow mismatches, are solved simultaneously with the primary flow equations.

Due to the tight hydraulic coupling of the reactor inlet with the rest of the primary system, the solution of the system flow equations, as well as coolant dynamics in the core, requires that the reactor inlet pressure be known at all times. This is obtained by combining mass conservation at the reactor inlet with momentum balance for the channel flows [3].

The thermo-hydraulic modeling of the intermediate circuit is essentially unchanged from that in SSC-L, with the exception of a pump tank model and some allowance for branching.

Pumps

The sodium pumps used in pool-type LMFBRs are vertically mounted, variable speed, centrifugal units. In the pump model, the impeller behavior is characterized by homologous head and torque relations encompassing all regions of operation. The homologous characteristics were derived from independent model test results with a centrifugal pump of specific speed (N_S) equal to 35 (SI units), and are applicable to LMFBR pumps in general [5]. Transient pump speed is obtained from an equation describing the torque balance of the pump.

Stratification in Pools

Since the hot pool forms a link in the primary flow circuit, it is necessary to predict the pool coolant temperature distribution with sufficient accuracy to determine its contribution to the net buoyancy head. It is also needed for the computation of the inlet temperature conditions for the components in the circuit.

During a normal reactor scram, the heat generation is reduced almost instantaneously while the coolant flow rate follows the pump coastdown. This mismatch between power and flow results in a situation where the core flow entering the hot pool is at a lower temperature than the temperature of the bulk pool sodium. This temperature difference leads to stratification when the decaying coolant momentum is insufficient to overcome the negative buoyancy force.

Currently, the stratification of core flow in the hot pool is represented by a two-zone model, based on the model for mixing in the upper plenum of loop-type LMFBRs in SSC-L [6]. Compared to the upper plenum representation of SSC-L, the hot pool analysis is more complicated because the pool cross-sectional area is not uniform. The volumes of the lower and upper zones, as well as the areas for heat transfer with the thermal barrier, are evaluated during the transient.

In the cold pool, perfect mixing of the IHX flow with the pool sodium is currently assumed. Allowances are made in the formulation for a user-specified fraction of the IHX flow to directly stream into the adjacent pumps without any mixing with the cold pool sodium.

Energy Balance in Pools

Energy balance in the hot pool includes heat transfer between the pool sodium and the barrier and other structures as well as with the cover gas, and energy additions due to core flow entering and IHX flow leaving the pool. Energy balance in the cold pool includes heat transfer to the barrier and cover gas. It also includes energy additions due to IHX and external bypass flows entering and pump flow leaving the pool. The overall heat transfer coefficients for the barrier are evaluated assuming a multi-plate configuration with a stagnant (or active) medium in between plates. More details on film coefficients and other modeling features can be found in [3].

Intermediate Heat Exchanger

The intermediate heat exchanger in pool-type LMFBRs is identical in function, and very similar in design, to that of loop-type designs. The only difference arises from the different configuration, where the IHX draws primary coolant from an open pool and discharges it to another open pool.

The energy equations are written using nodal heat balance with the 'donor cell' differencing approach. The overall heat transfer coefficients include the resistances due to film, wall, and fouling. Integration is performed using a single-layer fully implicit scheme.

The energy equations for the piping are written in the same manner.

Solution Procedure

During a transient the system hydraulic equations and equations for energy balance within the primary tank are solved together by a fifth-order predictor-corrector scheme. The thermal equations for the IHX and piping are solved in a marching fashion in the direction of flow.

The combined solution is made possible by judiciously taking advantage of the properties of liquid sodium: the time-dependent energy and momentum equations can be decoupled since the effect of pressure on subcooled liquid sodium properties is considered negligible. The energy equations have only a weak influence on the momentum equations through the sodium properties. This allows the hydraulic equations (along with the energy balance equations in the primary tank) to be solved first, using coolant properties from the IHX and pipes as boundary conditions, evaluated at the previous timestep.

INPUT DATA

The analyses presented in this paper have used Phenix data to a significant extent. Thus, while the results may not necessarily be representative of expected Phenix behavior, they should still provide useful trend comparisons.

Table 1 shows the values for major global variables used in the current simulations. These results were generated by the steady state portion of the SSC-P code.

The Phenix reactor core consists of 103 fuel and 90 radial blanket assemblies, producing nearly 563 MW of power. For the transients reported in this paper, the reactor core is modeled by three channels. The first is the fuel hot channel and represents one fuel assembly. The second models the remaining 102 fuel assemblies. The third channel models all of the radial blanket assemblies, together with the control and shielding assemblies. Table 2 presents the power and flow fractions used in the three-channel model. The coolant temperature rise values in the last column were obtained from a steady-state run using SSC-P. The power fraction in the fuel hot channel during steady-state is obtained from the radial peaking factor for Phenix [7] and the flow fraction is assigned to achieve a steady-state temperature rise of 235.2K [8]. The power fractions for the remaining two channels were obtained from decay heat data on fuel and blanket assemblies during refueling operations in Phenix [9], while the flow fractions were estimated from available information on the average flow per assembly [10].

The pump inertia and frictional torque were evaluated to yield the required pump coastdown as reported in the literature [8,11].

RESULTS

For the first transient, a total loss of electric power was assumed to occur, causing all coolant pumps to trip at time zero, followed by reactor scram from full power at 0.7 seconds. Figure 2(a) shows the predicted core flow decay, and the hot channel outlet temperature response for the first 10 minutes following transient initiation. The temperature is seen to reach a maximum of 919.2K at t = 240s. The core flow reaches a minimum value of 2% at about the same time, before natural circulation is established.

The behavior of pool levels is shown in Fig. 2(b). The level difference is seen to drop sharply from the steady-state value of 66.4 cm to about 21 cm in the first 30 seconds. Thereafter, the level changes are very gradual as the flows in the primary system begin to stabilize.

For the second transient case, a pipe rupture (with break area equal to 1.5 times the pipe cross-sectional area) was postulated to occur in the pump discharge line, 1.5 m upstream of the reactor inlet. The system response predicted by SSC-P is shown in Fig. 3. The hot channel outlet temperature reaches a maximum of about 1000K. The predicted impact of this accident for a pool design, as shown in Fig., 3, is less severe than the predicted consequences for a loop-type LMFBR. This is partly because, for a given reactor inlet pressure, the driving pressure for flow through the break is less in the pool design, leaving more flow available to cool the core, which leads to significantly lower core sodium and cladding temperatures.

For the third transient case, the predicted results from SSC-P for an unprotected loss of off-site power event are compared to the behavior of Phenix as reported by Freslon, et.al. [8] based on studies by NOVATOME. As seen in Figure 4, the agreement is quite close, with SSC-P predicting the occurrence of boiling in the fuel hot channel at 49 seconds, as compared to the published results which show boiling to occur at 44 seconds.

CONCLUSIONS

A variety of loss-of-flow transients in a pool-type LMFBR plant were simulated using the SSC-P computer code. From the results presented, and subject to the design parameters used, the following conclusions may be drawn:

1. The decay heat following a complete loss of forced cooling can be adequately removed via natural circulation.

2. A protected pipe rupture accident in a single pump discharge line is less serious for the pool vs. the loop design, leaving a significant margin to boiling in the fuel hot channel.

3. Boiling occurs in the fuel hot channel following an unprotected loss of off-site power event.

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Table I

Pretransient Phenix Global Variables

Parameter		SSC-P
Reactor Power	MW	563.
Total Core Flow	Kg/s	2760.0
Primary Heat Transport System		
Number of Intermediate Heat Exchangers	--	6
Number of Primary Sodium Pumps	--	3
Sodium Flow per IHX	Kg/s	460.0
Sodium Flow per Pump	Kg/s	1012.0
Reactor Inlet Temperature	K	673.13
Reactor Outlet Temperature	K	831.44
Hot Pool Average Temperature	K	828.30
Cold Pool Average Temperature	K	671.82
Coolant Level in Hot Pool	m	8.672
Coolant Level in Cold Pool	m	8.00912
Intermediate Heat Exchanger Temperatures		
Primary Sodium Inlet	K	828.30
Primary Sodium Outlet	K	668.08
Intermediate Sodium Inlet	K	623.15
Intermediate Sodium Outlet	K	822.46
Log-mean Δ T	K	25.34
Intermediate Heat Transport System		
Number of Loops	--	3
Sodium Flow per Loop	Kg/s	737.0

Table II

Steady-State Power and Flow Fractions

Channel No.	Number of Assemblies	Power Fraction	Flow Fraction	Coolant Temp. Rise, K
1	1	0.01092	0.0074	235.20
2	102	0.85408	0.8376	161.43
3	90	0.135	0.155	137.76

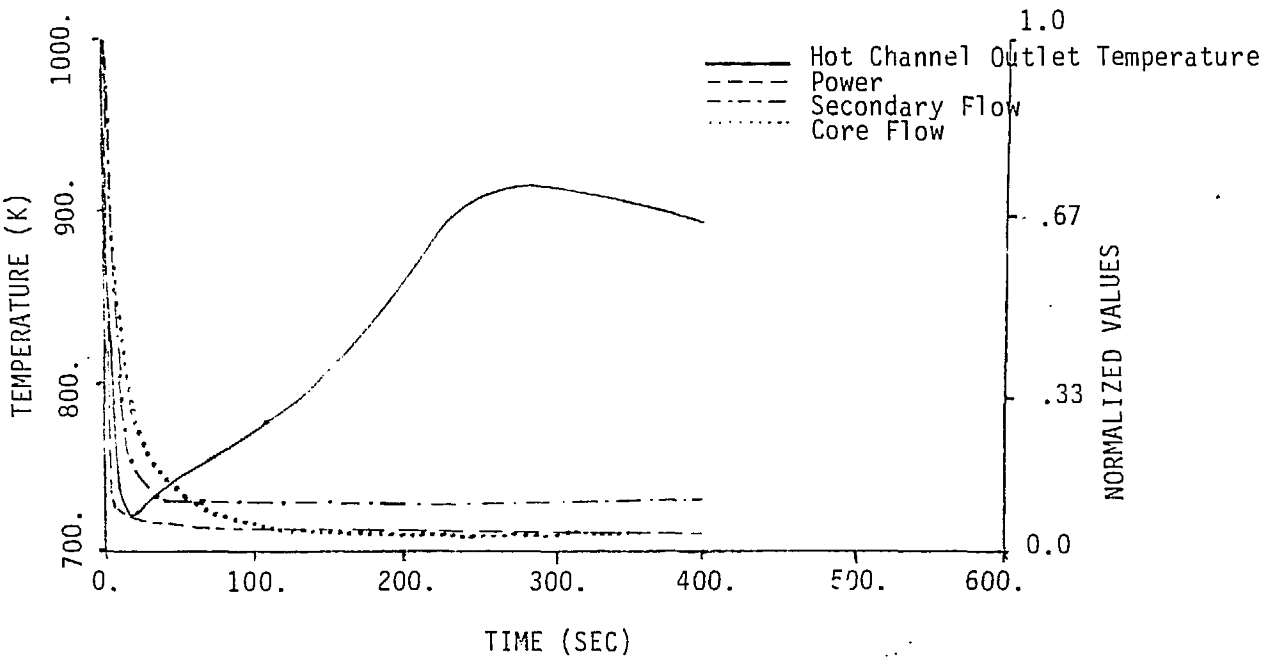

Fig. 2a - System response to a complete loss of electric power event, with scram

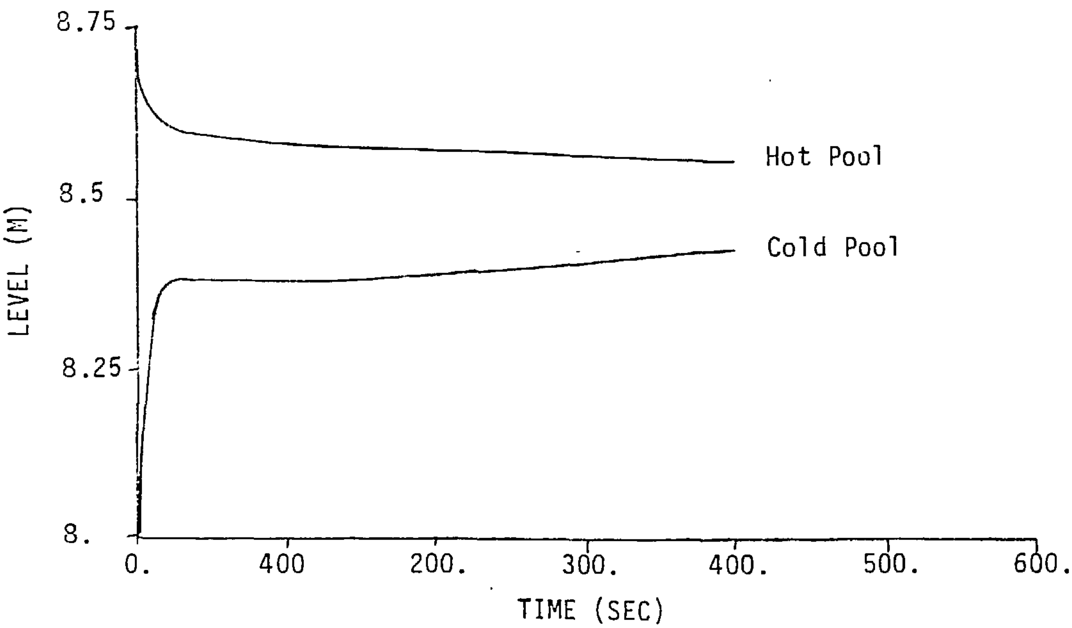

Fig. 2b - Sodium level in pools following a complete loss of electric power event, with scram

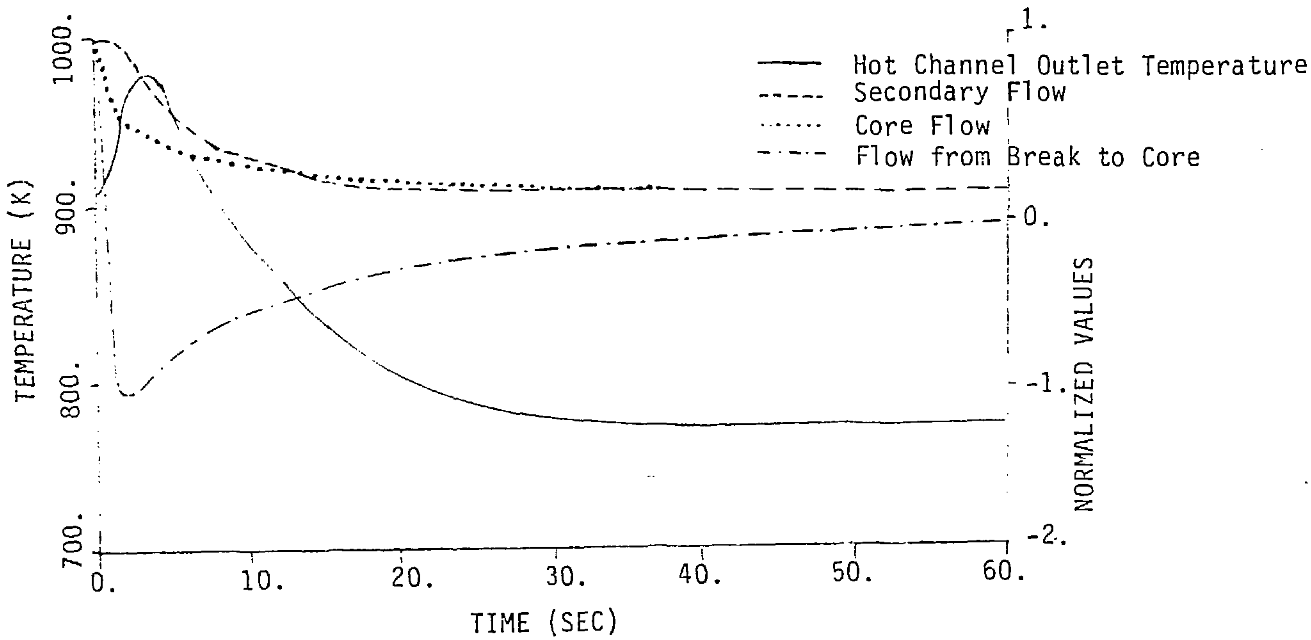

Fig. 3 - System response to a protected pipe rupture accident in a primary pump discharge line

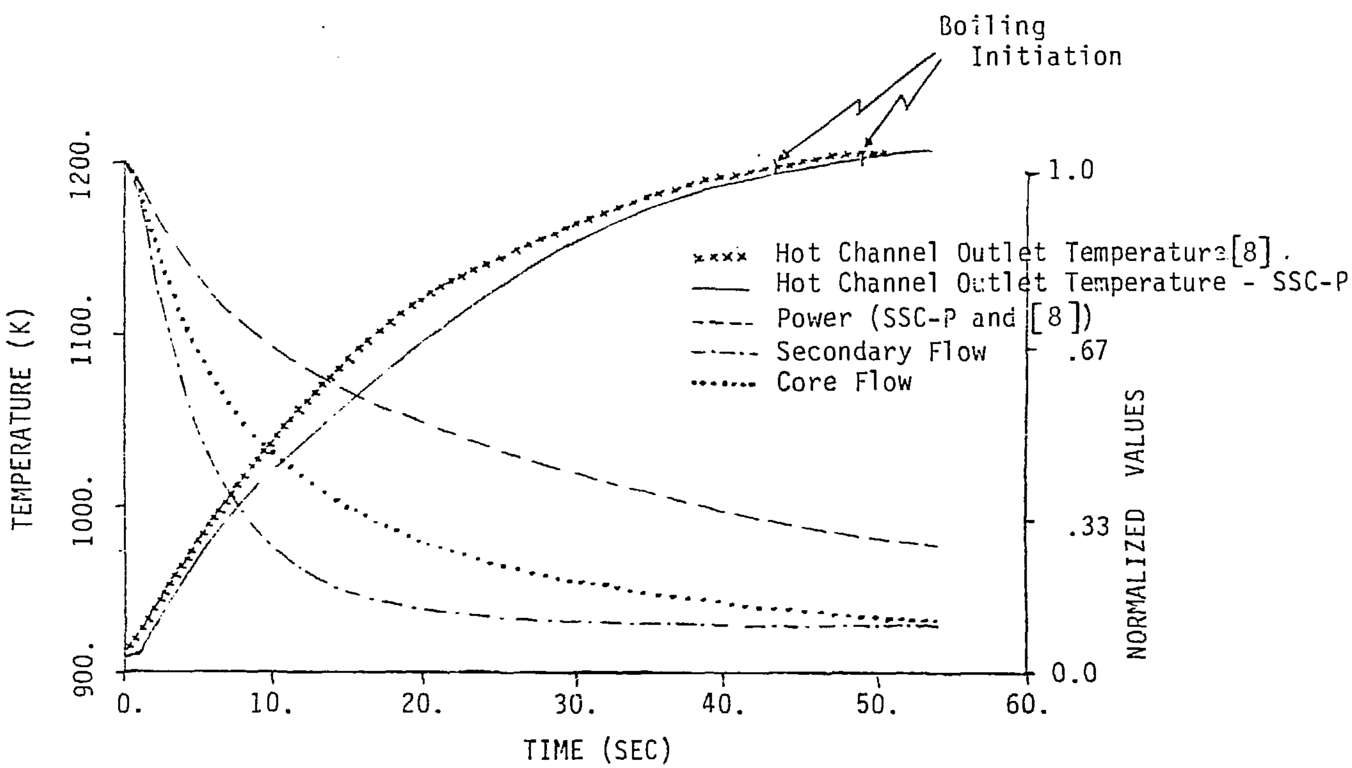

Fig. 4 - System response to an unprotected loss of off-site power event