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SIMULATION OF A TMI-2 TYPE SCENARIO AT THE ROSA-IV PROGRAM'S LARGE SCALE TEST FACILITY ; A FIRST LOOK

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SIMULATION OF A TMI-2 TYPE SCENARIO AT THE ROSA-IV PROGRAM'S LARGE SCALE TEST FACILITY: A FIRST LOOK

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The Three Mile Island-2 (TMI-2) type scenario in a Westinghouse (W) type four loop pressurized water reactor (PWR) was studied in preparation for upcoming tests in the ROSA-IV Program's Large Scale Test Facility (LSTF). The LSTF is a 1/48 scale simulator of a W type four loop PWR with full scale component elevation differences.

TMI-2 scenario simulation analyses were conducted to establish a pretest prediction data base for RELAP5 code evaluation purposes and to furnish a means of evaluating the LSTF's capability to simulate the reference PWR. The basis for such RELAP5 calculations and the similarities between the LSTF and reference PWR thermal-hydraulic behaviors during a TMI-2 type scenario are presented.

Keywords: PWR, LOCA, TMI-2, ROSA-IV, LSTF, Simulation, Reactor Safety

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ROSA－Ⅳ LSTF実験装置による TMI事故類似事故模擬実験の予備解析 －第１報

日本原子力研究所東海研究所安全工学部
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（1984年9月4日受理）

ROSA－Ⅳ計画によるLSTF総合実験のための予備解析の一環として，ウエスチングハウス型PWRにおけるTMI事故類似事故を模擬した実験の予備解析を行った。本解析は，計算コードRELAP5の性能評価のための実験前解析と，LSTF実験装置の性能予測とを目的とするものである。本報では，ウエスチングハウス型PWRにおいてTMI事故に類似した事象が発生するような事故条件を同定し，このような条件下でのPWRとLSTFの熱水力挙動の相似性を論ずる。

＊ 米国原子力規制委員会派遣の駐在研究員

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SUMMARY

The Three Mile Island-2 (TMI-2) accident is of great interest not only for Babcock & Wilcox (B&W) type plants, but for plants designed by other manufacturers as well. The TMI-2 type scenario in a W four loop plant is the subject addressed herein.

The Large Scale Test Facility (LSTF) is a 1/48 scale simulation of a W four loop pressurized water reactor (PWR). As such, the LSTF is being constructed as a major part of the Rig of Safety Assessment (ROSA) IV Program. Equipped with large pipe sizes (207mm diameter hot and cold legs), a 10 MWt core power capability, and the same elevation differences as a typical W four loop PWR, the LSTF is an ideal facility for investigation of the TMI-2 type scenario in which natural circulation phenomena are prevalent. In particular, the natural circulation behavior of a W type four loop PWR should be reproduced.

The LSTF pretest RELAP5 analyses are important ingredients of the ROSA IV Program. Such analyses provide insights of the LSTF transient envelope, establish a pretest prediction data base for RELAP5 code evaluation purposes and furnish a means of evaluating the LSTF capability to behave like a W type four loop PWR. Such needs motivated the study of the characteristics of the TMI-2 scenario, the study of the differences between a W type four loop PWR and a B&W TMI-2 type plant and RELAP5 calculations of TMI-2 type scenario simulations in both the LSTF and a W type four loop PWR.

The TMI-2 scenario was characterized by:

1. Continuous primary mass loss through a stuck open pressurizer power operated relief valve (PORV).
2. Little or no vessel water inventory replenishment since the operators throttled the emergency core cooling system (ECCS) flow.
3. Relatively high pressure caused by degraded primary to secondary heat transfer and,
4. Operator misinterpretation of the high pressurizer water level.

Because a W four loop plant is quite different from the TMI-2 plant in many areas, e.g., W U-tube steam generators (SG) vs the B&W once-through SGs, the presented analyses make no attempt to duplicate the TMI-2 accident chronology. Rather, the analyses seek to duplicate the four characteristics listed above.

The TMI-2 characteristics were examined to date by conducting an eleven calculation parametric study. Analyses are continuing such that the matrix may be expanded in the

future.

The thermal-hydraulic behavior of a W type four loop plant and the LSTF were examined by assuming the transient began with a loss of the main feedwater system (no auxiliary feedwater was available). Concurrently, a PORV opened.

Variations in the transient were examined by calculating the expected system behavior with both a rated area PORV and a rated flow PORV. In addition, transient behavior was calculated with the two high pressure injection (SI) pumps available, with the two SI pumps turned off when the pressurizer filled and with one SI pump available. The second case simulated the system behavior when an operator misinterpreted the meaning of a "full" pressurizer, such that the ECCS was shut off.

The analyses were conducted using RELAP5 Cy1 and Cy18 codes. The W type four loop PWR was modeled to simulate one SG and primary loop opposite three SGs and primary loops. The model loop representing three SGs also contained the pressurizer. The LSTF model was constructed with two equal loops - identical with the LSTF physical geometry.

Both the W four loop plant model and the LSTF model were initialized to rated power steady-state conditions. Thus, the nominal primary pressure was 15.6 MPa with representative hot and cold leg temperatures. However, the LSTF initial power (limited to 10 MWt) was 14% of the rated scaled value and thus the LSTF core mass flow was also low by the same factor.

The calculated transients began to depressurize immediately after time zero (due to the open PORV). Concurrently, the SG secondary water level decreased rapidly. In quick succession the reactor scrammed, the turbine throttle valve shut, the turbine bypass valves opened and the reactor coolant pumps tripped. SI pumps began to inject as the primary system pressure decreased below the pump shutoff head.

Shortly after the transient began, the pressurizer filled with water. Thus, in consideration of the operator misinterpretations present in the TMI-2 scenario, calculations were conducted with the SI pumps shutoff.

As the transient proceeded, general characteristics were found to be that the model loop without the pressurizer tended to collect steam in the SG U-tubes more quickly. Thus, flow stagnation occurred first in the loop without the pressurizer.

Flow stagnation in one primary loop was followed by core mass flow stagnation several hundred seconds later. Core flow stagnation resulted in some core heatup.

A difference observed between the W type four loop transient and the LSTF transient was in the behavior of the primary mass flow split between the two loops. The LSTF primary equal geometry showed a tendency for the fluid to oscillate from loop to loop. The plant model, with three physical loops simulated by one model loop, did not exhibit such behavior.

The TMI-2 scenario simulation analyses, conducted with the RELAP5 code have led to the following conclusions and observations:

1. The basic characteristics of a TMI-2 scenario in a W type four loop plant can best be simulated with a loss of feedwater in conjunction with an open rated flow PORV. Test boundary conditions would include SI system injection only until the pressurizer mixture level reached an agreed elevation.
2. The LSTF and (reference PWR) SI systems have sufficient capacity to replace the primary inventory lost through a rated flow PORV.
3. The TMI-2 sequence calculations have shown that the LSTF has the same qualitative thermal-hydraulic behavior as a W type four loop plant for such transients.
4. The loop to loop oscillations observed in the RELAP5 analyses of the LSTF seem to be a real phenomenon characteristic of the symmetrical two loop construction of the plant simulation coupled with a forcing function (provided by the pressurizer, for example) imposed on one loop. However, because the oscillations may be model-dependent, investigation of the phenomenon is continuing.
5. Primary loop pressure loss differences between the LSTF and the reference PWR RELAP5 models, caused by the loop resistance contributions of the recirculation pumps and the grid spacer plate may be indicative of similar differences in the facility.

1. INTRODUCTION

The Rig of Safety Assessment (ROSA)-IV Program was initiated by the Japan Atomic Energy Research Institute (JAERI) in 1980. The ROSA-IV Program was formed in response to the need for data characterizing small break loss-of-coolant accidents (SBLOCAs), abnormal and operational transients in Westinghouse (W) type four loop pressurized water reactors (PWRs). Such transients are dominated by phenomena peculiar to natural circulation. Thus, plant elevation differences are quite important.

Several test facilities have produced data relating to natural circulation over the past years, including PKL (Federal Republic of Germany), LOBI (Ispra, Italy) and Semiscale (United States of America). Although these facilities simulate typical plant elevation differences correctly, other scaling compromises combine to limit the usefulness of their data. The LOBI and Semiscale facilities have small pipe sizes which lead to atypical flow regimes and hot wall effects(1). The PKL facility is limited to test pressures below 4 MPa and rated power levels below 2 %. Thus, the PKL facility can only simulate a limited portion of most transients.

The ROSA-IV Program is international in scope. The United States Nuclear Regulatory Commission (USNRC) joined informally in 1981 and as a participant by cooperative agreement in 1984. The USNRC provides assistance in advanced instrumentation and code applications.

The Large Scale Test Facility (LSTF) is at the heart of the ROSA IV Program. The LSTF is being constructed to circumvent such compromises by having the proper plant elevation differences, pipe sizes e.g. the hot leg and cold leg diameters of 207 mm., representative primary pressure levels i.e., 16 MPa, and plant power levels sufficient to simulate the core decay heat 7 s. after scram. The LSTF volumes were scaled at 1/48 of a typical W four loop plant. The LSTF will provide data in an area currently containing limited information.

The consequence of the Three Mile Island-2 (TMI-2) accident is of great interest not only for Babcock & Wilcox (B&W) type plants, but for plants designed by other manufacturers as well. The possible TMI-2 type scenarios in a W four-loop plant are the subject addressed herein.

The objective of the study was to gather information to construct a portion of the LSTF test matrix relating to the TMI-2 scenario. The fact that some of the early events of the TMI-2 scenario can occur in a W type plant was established in an incident at the Beznau reactor in Switzerland(2). A turbine trip, combined with a turbine bypass failure ultimately caused the pressurizer power operated relief valves (PORVs) to open. One PORV jammed open. Afterwards, the pressurizer filled with water. The plant operator interpreted the transient correctly, safely shut the plant down and isolated the stuck-open PORV.

Further evidence that such transients should be studied is provided by examination of operator actions in B&W plants. Plant operators at 2 of 3 B&W plants(3) prematurely terminated the high pressure injection flow when confronted with a full pressurizer level reading following a loss-of-feedwater transient coupled with a stuck-open PORV.

The TMI-2 incident itself was characterized by some of these boundary conditions(4). In summary, the incident at TMI-2 had the characteristics shown in Table 1.1. Continuous primary system mass loss occurred through a stuck open PORV. Emergency core cooling system (ECCS) injection flow was reduced by the operator due to misinterpretation of the high pressurizer water level. The loss of main and auxiliary feedwater systems caused poor primary to secondary heat transfer and resulted in a relatively high pressure transient (Although the auxiliary feedwater system was restored after 8 minutes, the flow was throttled by the operator.).

The approach described herein was developed to provide insights for the proposed test matrix outlined in the Large Scale Test Facility conceptual design report(5). The conceptual design document(5) suggested that the LSTF test matrix contain sequences which simulate the timing and events of the actual TMI-2 scenario. Closer examination of such an undertaking(6) plus consideration of the fundamental differences between W and B&W plant designs show that major changes in test procedures must be made to LSTF boundary conditions. Differences between W and B&W plants which impact the LSTF test plan are discussed in the next section of this report.

Because the number of tests to be conducted in the TMI-2 simulation at the LSTF is relatively small(7), viz., 4, the test matrix will be examined (Section 3) from the perspective of variations of the most probable TMI-2 type scenario at a Japanese W type four loop plant (reference PWR).

The reference PWR and LSTF baseline calculations are described and compared in Section 4. Parametric calculations are discussed in Section 5 and conclusions are presented in Section 6. The results shown do not represent complete analyses of the TMI-2 type scenario as it will be conducted in the LSTF. To examine all aspects of the scenario, analyses are continuing. Thus, the following discussion should be regarded as a first look at simulating the TMI-2 type scenario in the LSTF.

TABLE 1.1: CHARACTERISTICS OF THE TMI-2 SCENARIO

CHARACTERISTIC	EXAMPLE
1. Continuous system mass loss.	a. Operator controlled letdown flow. b. Stuck open PORV.
2. Little or no vessel water inventory replenishment.	High pressure (HP) emergency core cooling system (ECCS) throttling.
3. Relatively high pressure.	Degraded primary to secondary heat transfer caused by secondary system mass loss.
4. Homogeneous two phase system.	Reactor coolant pumps left operational by operator.
5. Operator misinterpretations.	Operators misinterpret high pressurizer water level. As a result, throttle HP ECCS and activate system-letdown.

2. DIFFERENCES BETWEEN JAPANESE W FOUR LOOP AND TMI-2 CLASS PLANTS

The W type four loop reactors and the B&W TMI-2 class reactors differ in many areas. It is not the objective of this section to rigorously discuss all of their differences. The listed differences will alone assure that a Japanese constructed W type four loop plant will behave differently than the TMI-2 reactor for the same imposed initial boundary conditions. In addition, differences between standard operator procedures and the plant control systems will further impact the diverging transient behavior in the respective plants.

Major differences between the two plant types apparent from an initial study are listed in Table 2.1. Perhaps the most significant difference is the steam generator (SG) design. The once-through B&W SG design contains much less secondary fluid than the W design. Such a difference is apparent in the respective plant thermal-hydraulic behaviors during transients such as the station blackout, in which the feedwater system is inoperative. Comparisons of similar transients in the B&W TMI-2 plant and the W Zion plant(8) show SG dryout to occur more quickly in the TMI-2 plant by a factor of four at least.

Another significant difference centers on the reactor coolant pump (RCP) trip logic. Japanese built W type four loop plants trip their RCPs on low primary system pressure i.e., below 12.27 MPa., whereas B&W plants did not in 1979. Thus, the fact that the RCPs continued to operate during the TMI-2 accident caused the primary fluid to remain an unseparated two-phase mixture. However, the following calculations show the W type four loop plant to quickly become a separated two-phase system following saturation, due to early shutoff of the RCPs.

Other differences between the two plant types, apparent from a cursory study, are in the pressurizer surge line viz., TMI-2 has a loop seal whereas W four loop plants do not. B&W plants have a plant central control system which is quite different from the W four loop plants. Finally, B&W plants inject ECCS fluid into the reactor vessel through "vent valves" whereas W plants inject ECCS fluid upstream of the vessel in the hot leg.

One of the objectives of the ROSA IV Program TMI-2 simulation in the LSTF is to assess the plant behavior of a Japanese built W type four loop system given a set of initial and boundary conditions similar to those experienced at the TMI-2 reactor. Transient variations which are peculiar to the TMI-2 plant characteristics will not be examined herein.

TABLE 2.1: FUNDAMENTAL DIFFERENCES BETWEEN JAPANESE W TYPE 4 LOOP REACTORS AND THE TMI-2 REACTOR

ITEM	DESCRIPTION OF DIFFERENCE	IMPACT OF DIFFERENCE
1. Steam generators (SG).	TMI-2 has once through SGs and a W 4 loop reactor has a U-tube configuration.	Loss of offsite power calculations(8) have demonstrated that the TMI-2 SGs dried out in one fourth the time of a W 4 loop plant system.
2. Reactor coolant pump (RCP) trip logic.	Japanese built W 4 loop reactors trip their RCPs based on the safety injection signal. TMI-2 systems do not.	RCPs on or off define whether the primary system is unseparated or separated respectively.
3. Pressurizer loop seal.	TMI-2 systems have a candy cane pressurizer loop seal configuration. W 4 loop plants do not.	W pressurizer collects steam at a more rapid rate.
4. Plant control system.	B&W plants use a central control system whereas W 4 loop plants do not.	Unknown.
5. Reactor vessel vent valves.	B&W plants inject ECCS fluid into the reactor vessel through vent valves, whereas W plants inject ECCS fluid into the hot legs.	Not investigated.

3. THE CODES, MODELS AND CALCULATIONAL MATRIX

A portion of the TMI-2 scenario (especial to a W four loop plant) was simulated using the LSTF and reference PWR RELAP5 models(9). Calculations were conducted assuming the initiating event to be a main feedwater system trip. Thereafter, plant boundary conditions were determined by combining Japanese built W four loop plant characteristics and responses with the known events at the TMI-2 plant. For example, the reactor coolant pumps were tripped off at primary pressures less than 12.27 MPa. Standard W four loop plant operator procedures require the reactor coolant pumps to be tripped between 8.3 and 9.7 MPa.

The models (and codes), the initial conditions, boundary conditions and the calculational matrix are discussed in the following three subsections respectively. Because the RELAP5/MOD 1 Cy18 model was a modified version of the Cy1 reference PWR model(9,10), subsections 3.1 and 3.2 will discuss both the Cy1 model and the Cy18 model (i.e. modified Cy1 model) development and initial conditions respectively.

3.1 RELAP5 Code and Model Development

Most of the calculations shown in the following sections were conducted using RELAP5 Cy1(11) together with the LSTF and reference PWR models described in past analyses(9,10). However, the initial responses of the reference PWR were calculated using Cy18 and a modified RELAP5 model.

3.1.1 Cycle 1 model

The models used for the parametric calculations were developed and discussed in the initial loss-of-feedwater transient scoping studies report(9). The reference and current models are identical, with the exception of a junction change in the upper downcomer which routed fluid to the upper end of volume 21 instead of the lower end of volume 1 in the vessel upper head (see Figures 3.1 and 3.2). The nodalization change was made to enhance model stability under saturated conditions.

It should be noted that none of the calculations are complete. During the course of conducting the Cy1 calculations, the effort was plagued with water packing problems. These shortcomings deserve investigation.

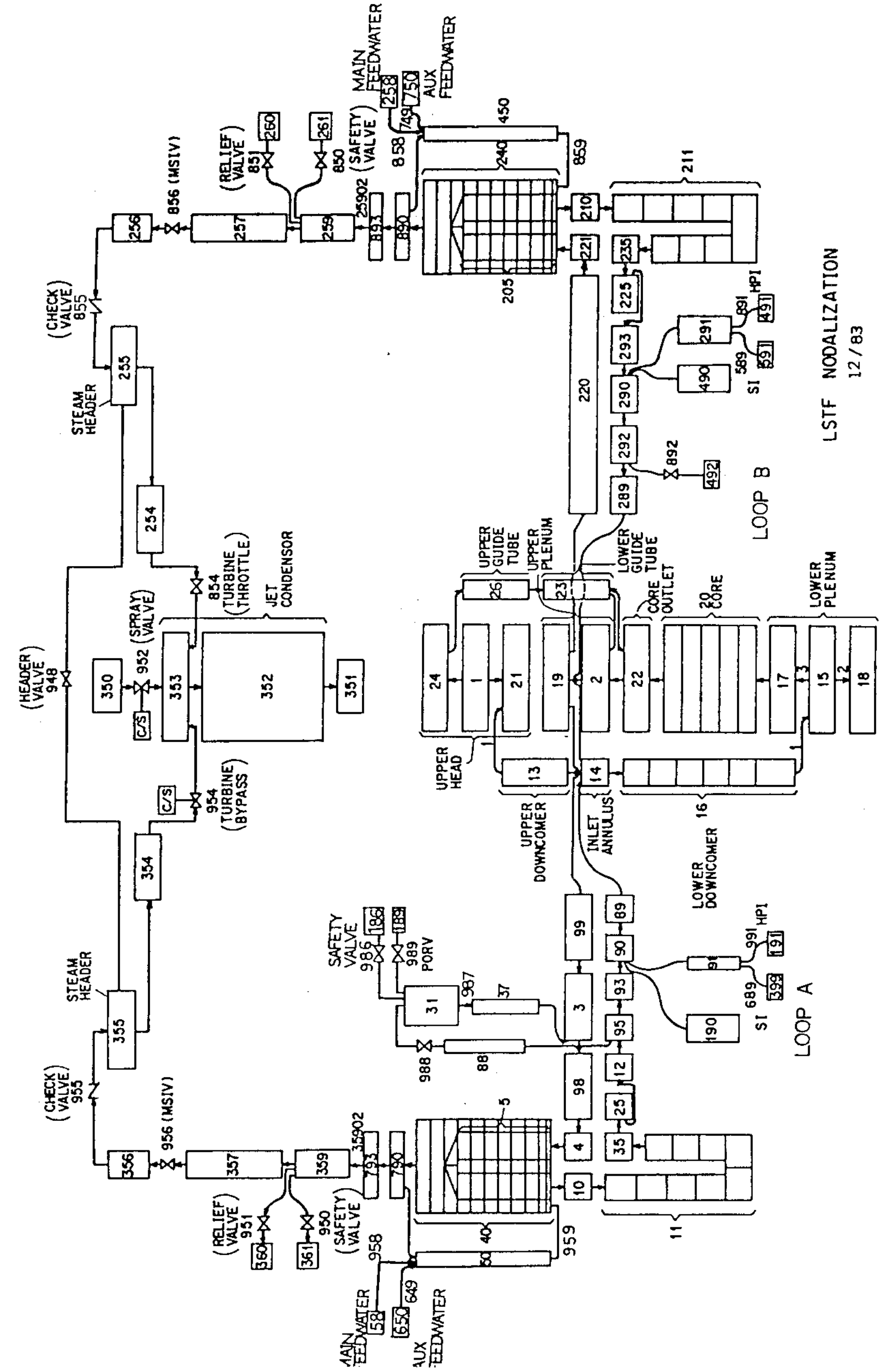

Figure 3.1 The Large Scale Test Facility (LSTF) RELAP5 model nodalization

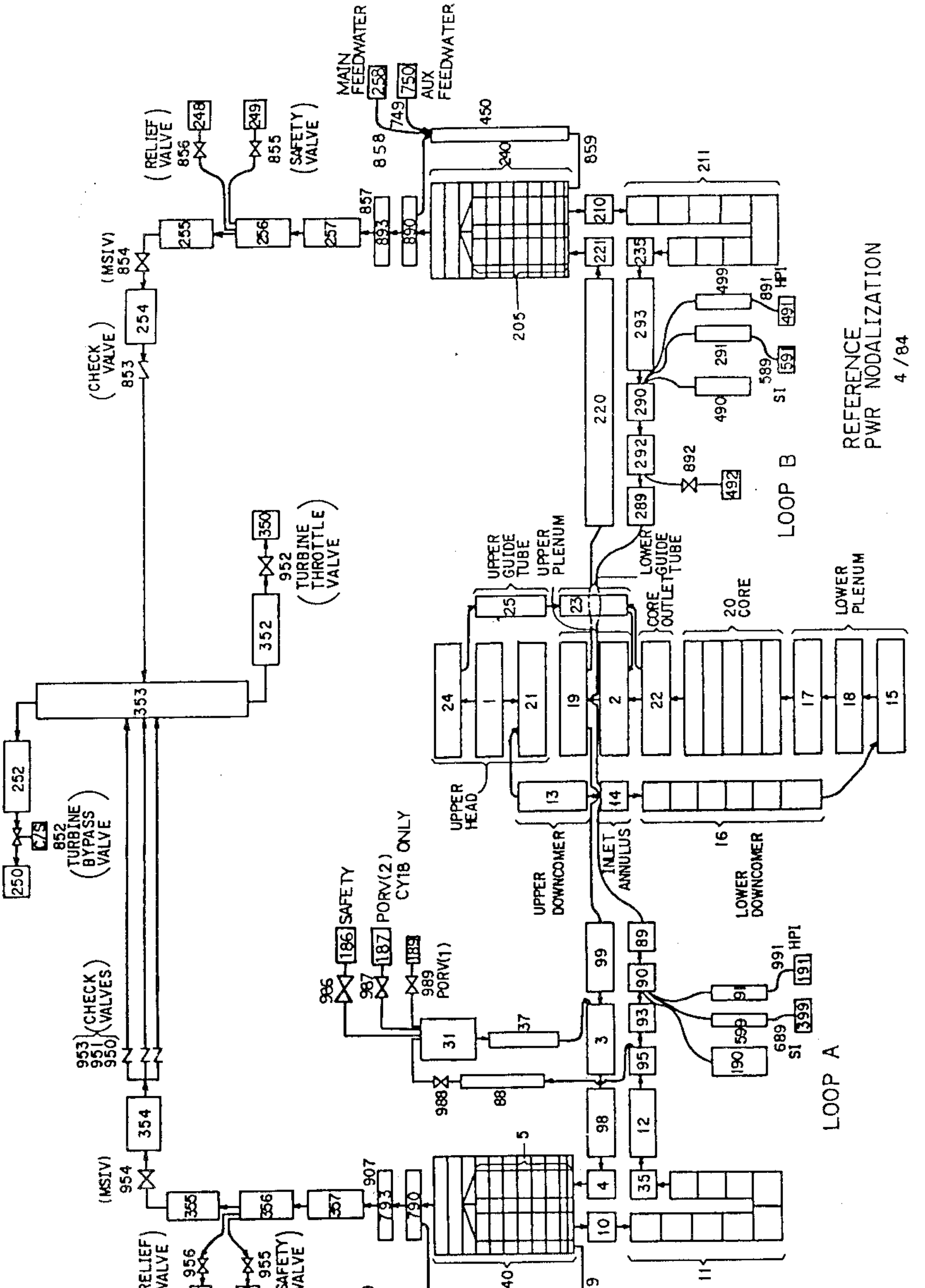

Figure 3.2 The reference PWR RELAP5 model nodalization

3.1.2 Cycle 18 model

Several important additions distinguish the Cy18 model from the Cy1 model. Because the Cy18 model was used to examine the initial response of the reference PWR (i.e. prior to scram), pressurizer heaters, pressurizer spray logic and the pressurizer wall material were added to the model. Further, the SG downcomer level measurement control logic was modified to more accurately represent the existing differential pressure tap locations. Finally, junction 987 was added to model the presence of two normal pressurizer power operated relief valves (PORVs). Junction 989 was left intact to model a PORV that opened on demand, but remained open thereafter.

3.1.3 Differences between the RELAP5 LSTF and reference PWR models

Past reports(5,9) have summarized the differences between the LSTF and a W type four loop reactor system. However, another factor must be considered when assessing the similariites of LSTF and reference PWR model predictions i.e., the differences between the LSTF and reference PWR models. An exhaustive comparison between the LSTF and reference PWR models will not be described herein. However, some of the differences between the models, which affect the calculations shown in the following sections, are listed. Specifically, items 3, 4 and 5 are differences between the reference PWR and LSTF models. However, items 3 and 4 have the potential to become physical differences between the LSTF and the reference PWR. An exhaustive comparison will be conducted at a later date.

In order to obtain LSTF event chronologies as close as possible to the reference PWR, the LSTF2 model(Defined in ref. 4) was used for the TMI-2 type scenario calculations. Thus, the following differences between the models were present:

(1) Steam generator (SG) initial water level - As discussed in Reference 9, the initial SG water level was lower in the LSTF model than in the reference PWR because the LSTF is only capable of producing 14 % of the plant scaled rated power. However, the SG secondary inventory was scaled to be 1/24 of the reference PWR i.e., the correct value.

(2) Reactor coolant pump (RCP) head at rated

conditions - Whereas the reference PWR is equipped with RCPs which contribute a 0.63 MPa head at rated conditions, the LSTF's RCPs only contribute a head of 0.01 MPa. The anomaly in pump steady-state conditions is related directly to the LSTF 10 MWt maximum power output capability i.e., 14 % of a 1/48 scaled core. Thus, at steady-state conditions, the LSTF can maintain a substantial percentage of the rated flow rate with only the natural circulation contribution. The difference in RCP rated heads for the reference PWR and LSTF is important during the portion of any transient when the RCPs have been shut off and the pump impellors are decreasing to zero rotational velocity. The pump coastdown period is estimated to be 250 s.

(3) RCP pressure loss - The fluid pressure loss through the RCPs is important throughout the transient simulation. The LSTF RELAP5 model contains a pump component with 14 % more loss than the reference PWR at a stationary condition.

(4) Core support plate/inlet orifices - The LSTF RELAP5 model contains an area expansion at the core inlet which produces a pressure loss 17 times greater than the reference PWR. Such a difference is an attempt to model the pressure loss sustained as the vessel flow passes from the downcomer and between 1064 electrical cables in the LSTF lower plenum.

(5) Elevation differences - The current LSTF model, based on preliminary designs, contains an elevation difference of 16.78 m between the core bottom and the top of the SG U-tubes. The corresponding dimension in the reference PWR model is 18.05 m.

(6) SG secondary pressure level - Initially, the SG secondary pressure level in the LSTF was maintained at 7.14 MPa (corresponding reference PWR pressure = 5.71 MPa) to limit the heat transfer to the core rated value(9).

The above six items all have an impact on the LSTF behavior insofar as the reference PWR is simulated. Items 2 and 6 will influence the first hundred or so seconds of any transient as the heat transfer is limited from the primary to the secondary by high sink temperatures (item 6) and low flow (item 2). Further, the primary to secondary heat transfer will be affected during the natural circulation portion of the simulated transients by a friction and expansion pressure loss component that is 12 percent larger in the LSTF than in the reference PWR (items 3 and 4). The

LSTF RELAP5 model natural circulation flow capability is further influenced by the net elevation differences between the core and SG U-tubes (item 5). Finally, if the correctly scaled SG secondary water inventory is used to simulate the SG dryout times, the appropriate low and low-low trips must be estimated from outside sources, since the LSTF initial water level is less than the reference PWR secondary (item 1).

3.2 Boundary and Initial Conditions

The initial conditions used in the Cy1 and Cy18 models were different. The Cy1 model conditions were unchanged from those previously reported(9). However, the Cy18 reference PWR model had an increased SG secondary mass (see Table 3.1) based on more accurate initial conditions provided by the USNRC Reactor Training Center for a W type four loop plant.

All calculations assumed the loss of the main feedwater system at the start of the transient. The reference PWR models and the LSTF model were assumed to be at 100 % rated power as the transient began (see Table 3.1 - Note that 100 % rated power for the LSTF is 14 % of 1/48 scaled reference PWR rated power). System trip logic was triggered as listed in Table 3.2. The trip logic listed specifically for the Cy1 model was unchanged from previous calculations(10). However, pressurizer spray and heater trip logic was used for the Cy18 model as listed. Also, the two operational PORVs were programmed to behave as listed.

3.3 Calculational Matrix

The calculational matrix (see Table 3.3) was constructed to examine the effect of variations in the PORV flow area i.e., junction 989, combined with variations in the available emergency core cooling system (ECCS) flow i.e., junctions 589,689,891 and 991. These calculations assumed the PORV to fail open at the beginning of the transient.

Previous calculations were conducted using an open PORV with the nominal geometrical area(10). However, the listed geometrical area passes nearly double the rated PORV steam flow rate (26.4 kg/s at a 16.18 MPa stagnation pressure). Thus, to provide a transition between previous calculations and yet provide a more realistic simulation of the subject transient, the calculations were conducted with the PORV at both the rated area (RA) and at the rated flow (RF).

Variations in the ECCS flow were examined by changing the injection quantity from the rated conditions for two high head safety injection (SI) pumps to none. The calculations conducted with zero ECCS flow allowed the two SI pumps to inject upon demand, but the flow was shut off when the pressurizer level reached the top. Such an operational sequence was designed to simulate an operator who responded to a full pressurizer i.e., "solid-system" by shutting off the ECCS (as in TMI-2(4)).

For the purposes of this report, the baseline calculation has two SI pumps i.e., the high pressure charging pumps are failed and a stuck-open RA PORV.

TABLE 3.1: LSTF AND REFERENCE PWR INITIAL CONDITIONS

	LSTF (Cy1)	PWR (Cy1)	PWR (Cy18)
System pressure (MPa)	15.59	15.60	15.60
Cold leg temperature			
A loop (K)	562.18	562.2	562.2
B loop (K)	562.14	562.22	562.22
Hot leg temperature			
A loop (K)	598.15	598.22	598.22
B loop (K)	598.15	598.22	598.22
Core flow rate (kg/s)	48.37(b)	16516.	16516.
Core power (MW)	10.0(b)	3423.	3423.
Steam generator secondary			
A loop			
Pressure (MPa)	7.14(c)	5.71	5.71
Mass (kg)	1658.9	117043.	129000.
B loop			
Pressure (MPa)	7.14(c)	5.71	5.71
Mass (kg)	1652.0	39068.	43000.

(a) Based on intact loop data.
(b) Both the initial flow rate and core power are 14% of the full-scaled values based on a system scale factor of 1/48.
(c) The initial SG secondary pressure is higher in the LSTF than in the PWR to reduce the steady-state heat transfer to 10 MWt.

TABLE 3.2: RELAP5 MOD1 (CYCLES 1 & 18) TRIP LOGIC

No	Action	Setpoint
1.	PWR scram, SG throttle valve closure, turbine bypass valve opens(a).	P (12.97 MPa(b), P) 16.46 MPa(b) or SG downcomer level (25%(c).
2.	Trip coolant pump and initiate high pressure injection and safety injection.	P (12.27 MPa(b) or P (4.235 MPa(d)
3.	LSTF core power trip(e).	Scram signal plus 7.1 s.
4.	Auxiliary feedwater begins to inject. However, the sytem failed in these calculations.	On loss of main feedwater(f).
5.	Main steam isolation valve closes.	P (4.235 MPa(d).
6.	SG relief valve	
	open	P = 8.05 MPa(d)
	close	P = 7.82 MPa(d)
7.(g)	Pressurizer spray	Spray turns on at P(b)) or = 15.68 MPa. Spray quantity directly proportional to system pressure.
8.(g)	Pressurizer heaters	
	Proportional	Heaters on P(b)(15.614 MPa
	Backup	Heaters on at P(b)(15.341 MPa
9.	Pressurizer PORVs	
	Failed open	time) 0.
	Cy1	P(b)) or = 16.2 MPa
	Cy18	Open P) or = 16.2 MPa; Close P (16.07 MPa.

(a) If the primary, mean temperature is above 564.9 K (plus a 2.78 K delay), the turbine bypass valve opens, otherwise the valve remains closed. Note: the turbine throttle valve was programmed to close at P) or = 16.2 MPa in the Cy18 case.
(b) Pressurizer pressure.
(c) Collapsed liquid level in upper SG downcomer less than 25% of full scale in the reference PWR.
(d) SG secondary steam dome pressure.
(e) The PWR core power decreases to 14% power level in 7.1 s after scram. The LSTF steady state power level corresponds to 14% of the reference PWR. The LSTF core power was tripped at 20 s based on the reference PWR calculations(9).
(f) As a simplifying assumption, and until more information is obtained concerning the reference PWR, no delay in the auxiliary feedwater startup time was assumed.
(g) Cy18 model only.

TABLE 3.3 TMI-2 SCENARIO SEQUENCE MATRIX: A FIRST LOOK

CALCULATION	PORV MODEL	AVAILABLE(a,b) ECCS EQUIPMENT	COMMENTS
1	RATED AREA (RA)	2 SI	Baseline calculation
2	RA	2 SI(c)	ECCS reduced to zero at 200s as pressurizer level peaked.
3	RA	1 SI	
4	RATED FLOW(RF)	2 SI	Not conducted for the LSTF.
5	RF	2 SI(d)	ECCS reduced to zero at 400s as pressurizer level peaked.
6	RF	1 SI	

(a) The ECCS equipment available is:
1. Two high head safety injection pumps (SI): Shutoff head = 10.7 MPa.
2. Two centrifugal high pressure injection charging pumps (HPI): Shutoff head = 18.2 MPa.
3. Two residual heat removal pumps (RHR): shutoff head = 1.3 MPa.
4. Four accumulators: Injection pressure = 4.5 MPa.

(b) Primary pressures in the calculations shown never decreased to the RHR pump shutoff head or the accumulator injection pressure.

(c) Only operative to 200s.

(d) Only operative to 400s.

4. RESULTS

The LSTF was designed to simulate a W type four loop reactor transient response after scram has occurred. However, the transient initiators, the effect of transient initiators and the reactor system behavior when the core power level is greater than 14% of the rated value cannot be simulated.

An initial study of the reference PWR behavior during the first seconds of a loss of feedwater transient has been conducted and is discussed in subsection 4.1. A portion of the long term transient is discussed in subsection 4.2. Subsections 4.2.1 and 4.2.2 are structured to give a detailed description of the baseline calculation (Number 1 - see Table 3.3) for the reference PWR and the LSTF. A comparison between the LSTF and reference PWR is given in subsection 4.2.3. Thereafter, calculations 2 through 6 are summarized in Section 5.

4.1 Study of the Initial Transient

The initiating events of the B&W TMI-2 accident quickly led to a turbine trip with loss of the main feedwater system. Three seconds after the turbine trip, the pressurizer pilot operated relief valves opened. The failure of a pilot operated relief valve to close was an event of major importance in the accident scenario.

The study of the initial transient immediately following the loss of the main feedwater system in a W type four loop plant was undertaken to examine the circumstances which would cause the pressurizer power operated relief valves (PORVs) to open. To do so, calculations were conducted to each scram setpoint to determine whether the PORVs had opened. During the course of the calculations, the pressurizer spray was found to be influential in the timing of the primary system pressure rise and the timing of scram under some conditions. Thus, four individual transients were examined, based on the following equipment assumptions:

1. The "low" steam generator (SG) water level scram, i.e. 25%, together with a steam-feedwater flow mismatch greater than 40%, was assumed operative.
2. The "low-low" SG water level scram, i.e. 10%, was assumed operative. The "low" scram trip was assumed failed.
3. The high primary system pressure scram was assumed operative, but the SG level trips and the pressurizer spray were assumed inoperative.
4. The high primary system pressure scram and the

pressurizer spray system were assumed operative. The SG level trips were assumed inoperative.

The pressurizer pressures for the above calculations are shown in Figure 4.1 as a function of transient time. The calculations were conducted until scram occurred in each case using the reference PWR model and RELAP5/MOD1 Cy18. The transient began with the total loss of the main feedwater system at time zero. In addition, the auxiliary feedwater system failed.

4.1.1 Initial transients: Scram on SG water level

As the transient began (see Fig 4.1), the primary pressure was unaffected for the first 5s. However, after 5s the primary pressure began to increase as the primary to secondary heat transfer decreased.

Given that the reactor system trip logic behaved as designed, the calculation indicated scram at 5.8s (see curve 1 - Fig 4.1) as the SG downcomer water level reached the 25% level (combined with a steam-feedwater flow mismatch exceeding 40%). No PORVs were opened.

If the "low" SG water level trip failed, the SG water level would continue to decrease and the primary system pressure would increase (see curve 2 - Fig 4.1). The second calculation indicated that scram occurred at 8.5s as the SG downcomer water level reached 10%. However, the primary system pressure did not increase to the PORV setpoint.

4.1.2 Initial transients: Scram on high primary system pressure.

Given that both SG water level scram trips failed, the system was calculated to behave as shown in curves 3 or 4, depending on the availability of the pressurizer spray. These calculations have the same primary system pressure behavior as shown in curve 2 until 8.5s. Thereafter the primary system pressure continued to increase as the primary to secondary heat transfer decreased.

If the pressurizer spray were unavailable, the PORVs were calculated to first open at 21s as the pressure exceeded 16.2 MPa (see Table 3.2). The PORV steam flow rate was sufficiently large to depressurize the primary and maintain the system pressure below the high pressure scram setpoint. However, primary to secondary heat transfer was degraded to such a degree that the system continued to pressurize to the

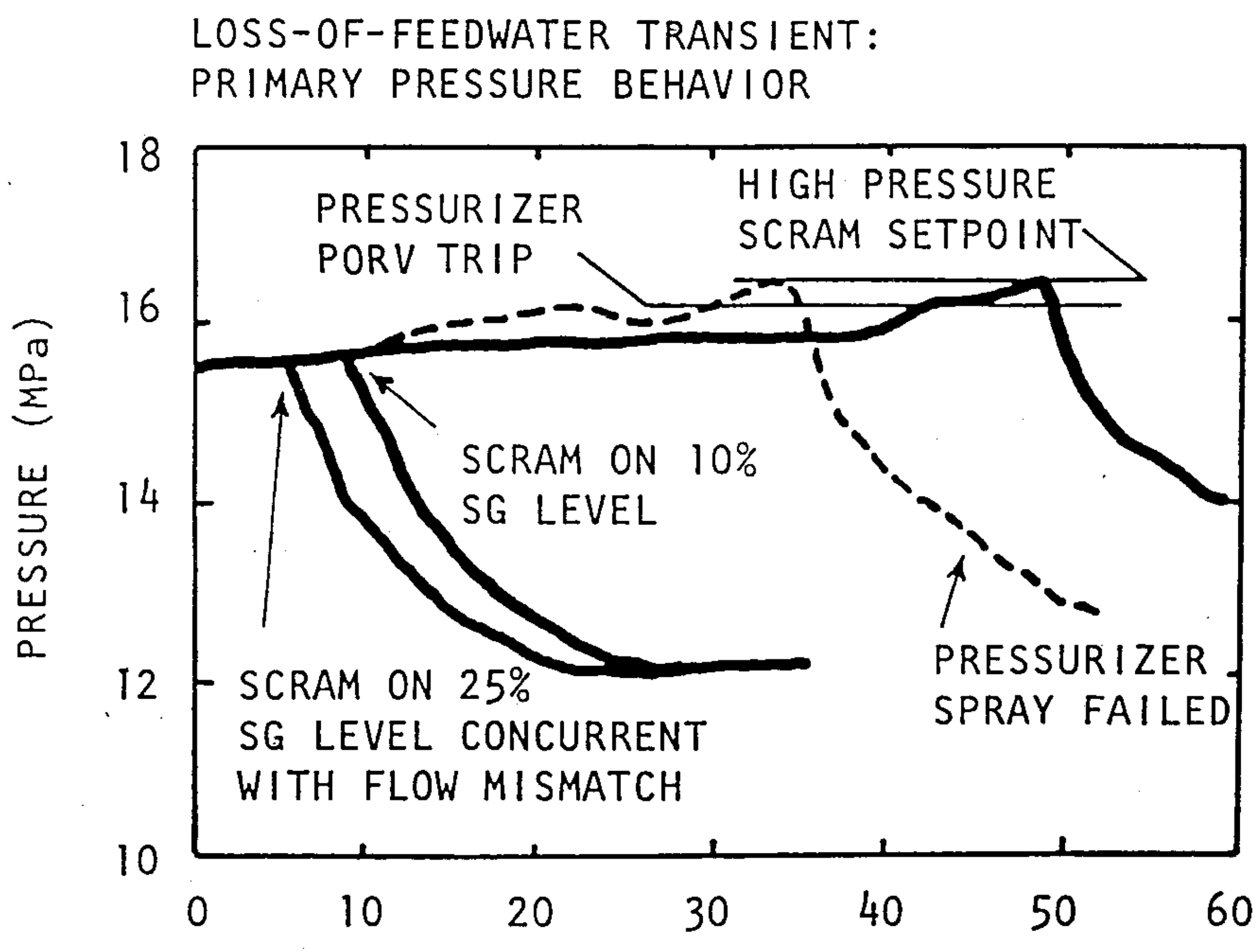

Figure 4.1 Reference PWR loss of feedwater transient: upper plenum pressure behavior

scram setpoint.

Transient events calculated to occur if the pressurizer spray were available were similar, but on a different time scale. The presence of the pressurizer spray resulted in a slow, but steady pressure rise. The PORVs opened at 42 s. But because of degraded heat transfer in the SGs, the pressure continued to rise, but at a reduced rate until the system scrammed on high pressure.

4.1.3 Synopsis

It should be noted that if the "low" SG level trip failed, a 15% uncertainty limit on the "low-low" SG level trip would delay the trip to 11 s. In addition, if a 3% uncertainty limit were applicable to the PORV setpoint, the lower setpoint bound would be exceeded by 10 s. Thus, certain combinations of failures and equipment uncertainty bands could result in a PORV challenge. Even so, these calculations indicate that the PORVs in a W four loop plant are not likely to open following a loss of the main feedwater system initiated transient.

4.1.4 Limitations of the initial transient calculations

Limitations of the calculations are defined by several shortcomings: (1) nuclear core reactivity feedback is not modeled, (2) the pressurizer water level is most likely below the rated value, (3) the turbine throttle valve was programmed to shut at pressurizer pressures above 16.2 MPa (see Table 3.2); thus, the primary to secondary heat transfer was not calculated realistically at pressures above the PORV setpoint and (4) the plant control systems are not modeled (thus the realistic response of the turbine throttle valve, the reactor coolant pumps, control rods, etc., are unknown at this writing). As additional information becomes available, the calculation can be updated.

4.2 Study of the Long Term Transient Baseline Calculation

The baseline calculation was conducted assuming a nominal PORV open area (without regard to the calculated flow rate). Further, the high pressure emergency core cooling system (HP ECCS) charging pumps and the auxiliary feedwater systems were assumed failed throughout the transient.

4.2.1 Reference PWR thermal-hydraulic behavior

The transient began as the main feedwater system failed and the feedwater flow quickly became zero (see Table 4.1). Concurrently, a PORV opened and remained open throughout the transient. The auxiliary feedwater system received a signal to start by 0.01 s, but was inoperative. Thus, the SG secondary water level began to decrease rapidly (see Figure 4.2) as the primary system power remained at the steady state value.

As the SG water level continued to decrease, the reactor scrammed (at 11.3 s), the turbine bypass valve opened and the turbine throttle valve received a signal to shut as the SG water level reached the 25 % full elevation. Following scram, the primary to secondary heat transfer rate decreased rapidly causing the SG downcomer water level to quickly decrease as the secondary voids collapsed. Because the only available primary heat source was the core decay and stored heat, the primary loop average temperature quickly decreased below 564.9 K and the turbine bypass valves shut by 14 s. In addition, the turbine throttle valves (3 s closure time) were fully closed by 14.4s. Thus, by 14.4 s, the secondary mass loss became zero.

The primary system depressurized (see Figure 4.3) quickly during the first few seconds of the transient. The pressure decreased below 12.27 MPa by 17.4 s which tripped off the reactor coolant pumps (RCPs). With the RCPs switched off, the primary mass flow rapidly decreased and the primary to secondary heat transfer decreased at an increasing rate.

Concurrent with the RCP trip, the high pressure (HP) ECCS received a signal to inject. However, because the high pressure charging pumps were unavailable, the remaining HP ECCS was only capable of injection at primary pressures below 10.7 MPa. Thus, ECCS injection was delayed as the system continued to depressurize.

The primary pressure reached 10.7 MPa and ECCS injection began at 35 s. Thereafter, cold i.e., 293 K water was injected in the system's cold legs. The primary mass inventory (see Fig 4.4) heretofor decreasing as mass was exhausted through the PORV, began to increase as ECCS water began to enter the system.

During the first 35 s of the transient, as the primary system pressure decreased (see Fig 4.3) and the core power level rapidly decreased, the secondary pressure also decreased (see Fig 4.5) as the primary to secondary heat transfer decreased. Secondary depressurization was caused by secondary steam exhaust through the turbine bypass and throttle valves. However, by 25s, as the core decay power

TABLE 4.1: TRANSIENT CHRONOLOGY FOR THE LSTF AND REFERENCE PWR UP TO ECCS INJECTION - KEY EVENTS

No	Event	TIMING(a) (s) LPWR CALCULATIONS 1-3	4-6	LSTF CALCULATIONS 1-3	4-6
1.	Loss of feedwater, PORV opens, auxiliary feedwater receives signal to operate.	0.	0.	0.	0.
2.	Feedwater flow = zero.	0.01	0.01	0.01	0.01
3.	Indicated SG water level = 25%. Reactor scrams	11.3	11.3	18.2	18.2
4.	Turbine throttle valve receives signal to shut.	11.3	11.3	11.1	11.1
5.	Indicated SG water level = 10%	14.5	14.5	NA	NA
6.	RCPs trip, high pressure ECCS pumps trip on.	17.4	35.0	38.3	63.8
7.	ECCS injection begins.	35.0	82.0	57.0	115.0

(a) Calculation numbers correspond to Table 3.3 listing.
NA = Not applicable.

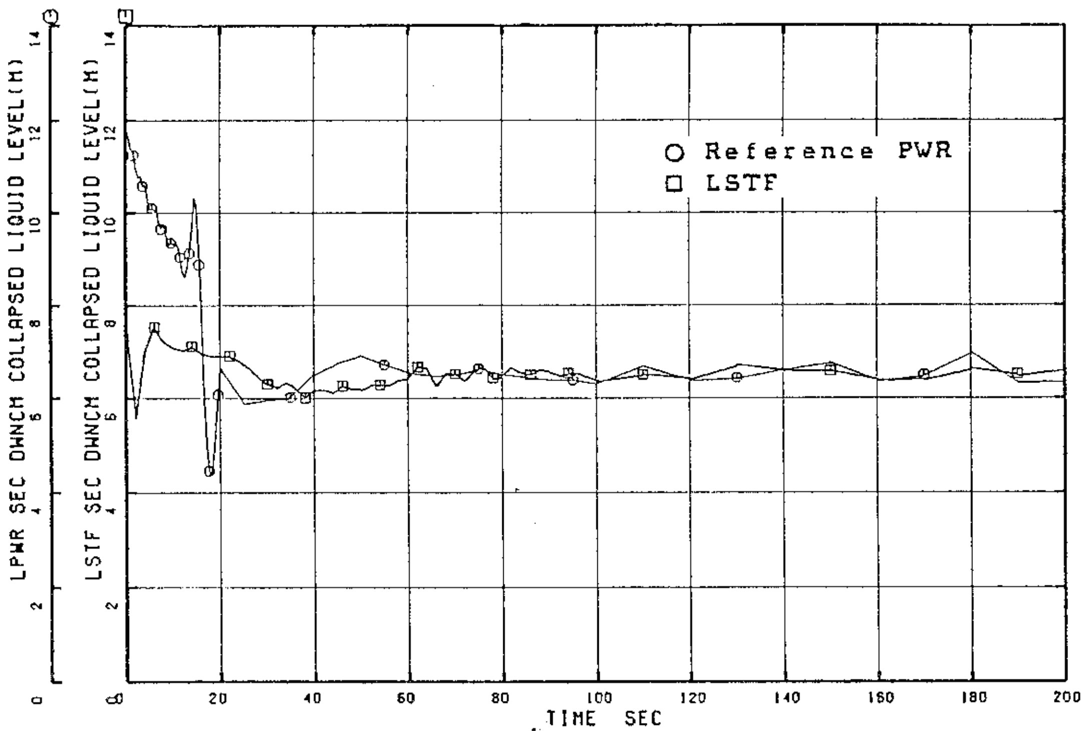

Figure 4.2 Comparison of the LSTF and reference PWR secondary downcomer water level: calculation 1 (first 200s)

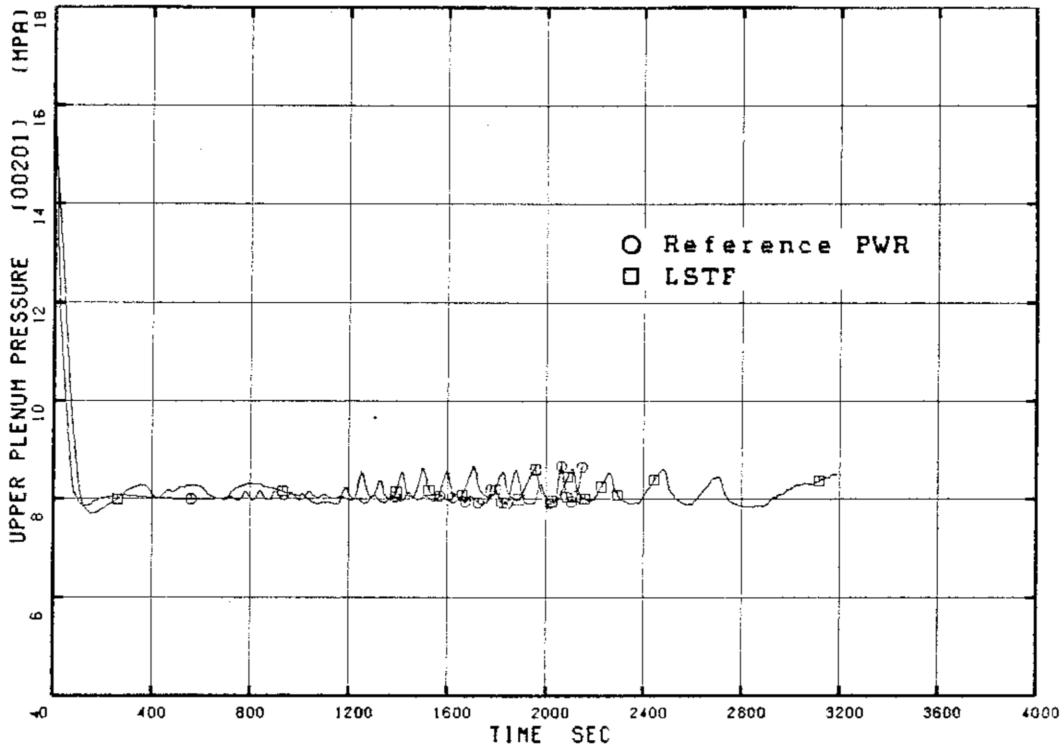

Figure 4.3 Comparison of the LSTF and reference PWR upper plenum pressure: calculation 1

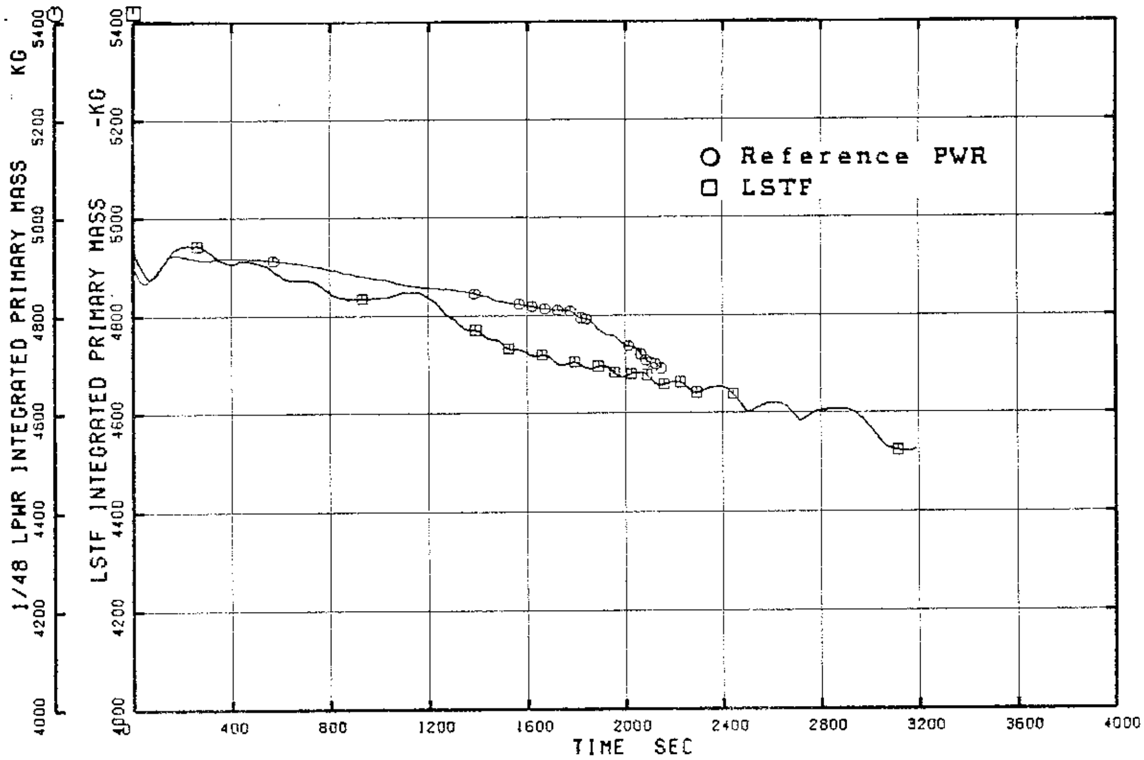

Figure 4.4 Comparison of the LSTF and reference PWR primary mass: calculation 1

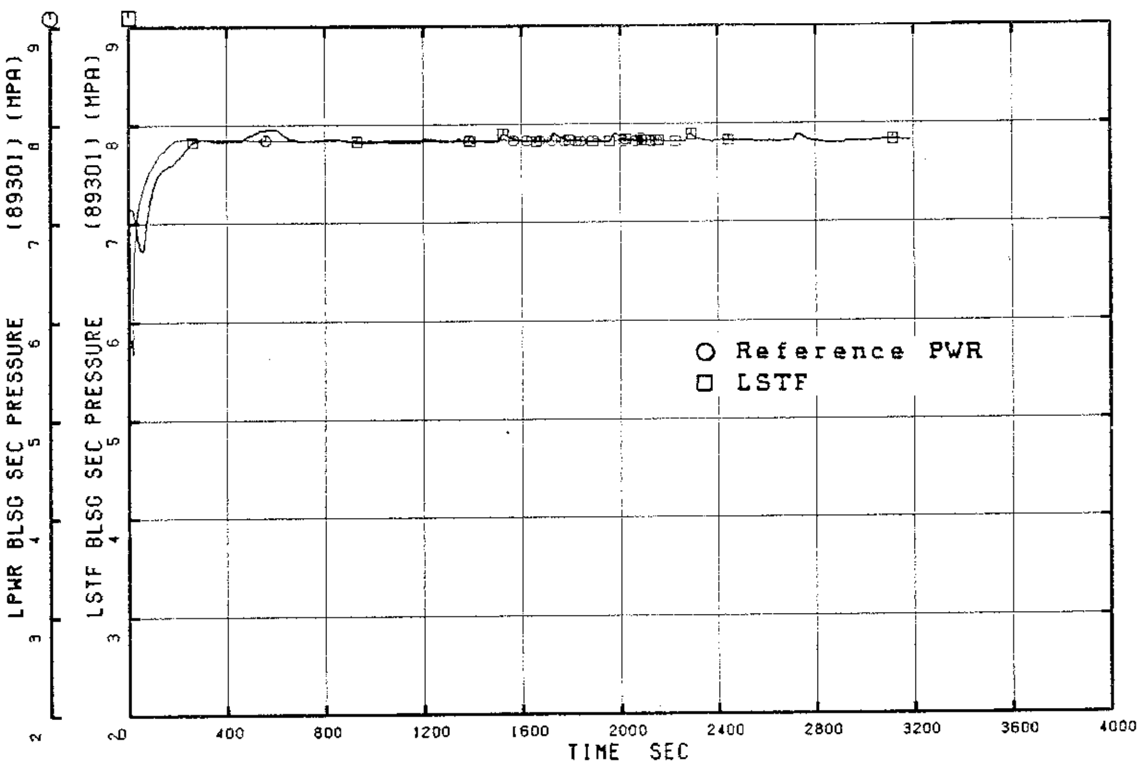

Figure 4.5 Comparison of the LSTF and reference PWR secondary pressure: calculation 1

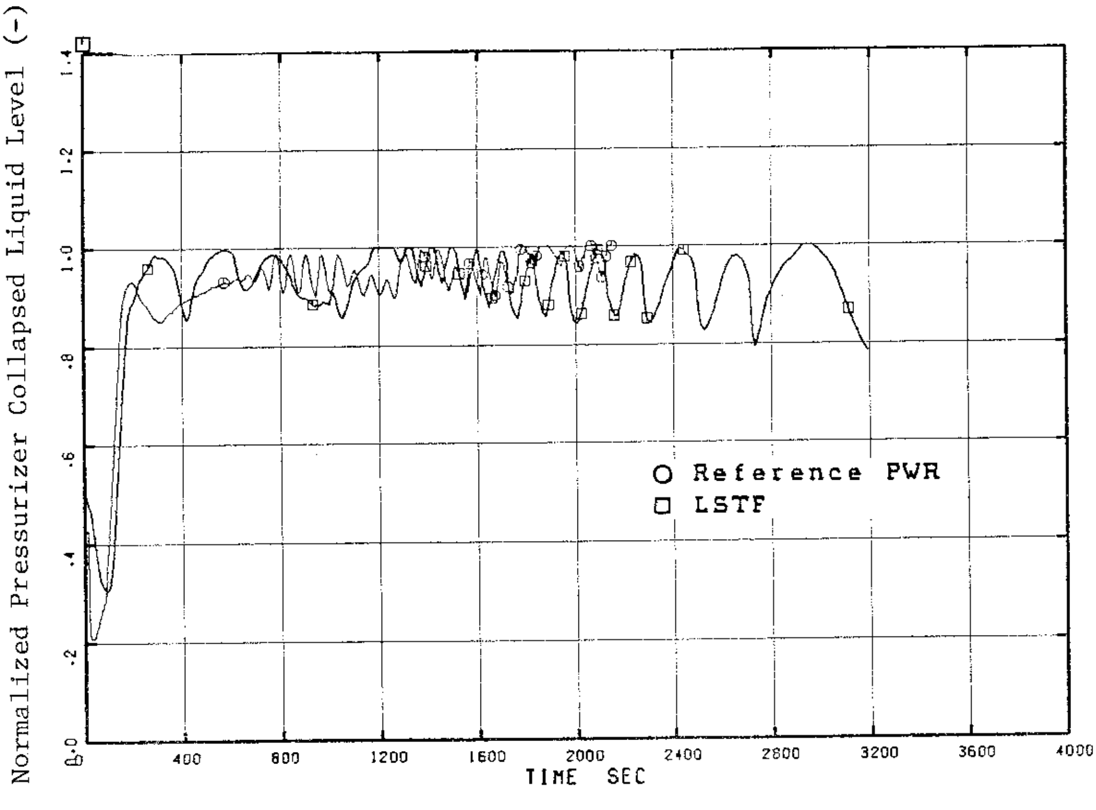

Figure 4.6 Comparison of the LSTF and reference PWR pressurizer collapsed liquid level: calculation 1

stabilized (and the primary system energy loss through the PORV continued to decrease), the secondary pressure reached an inflection point and then began to increase. The secondary system pressure increase served to decrease the primary to secondary heat transfer even further.

Reduced heat transfer from the primary to secondary and system depressurization i.e. flashing hastened primary steam formation. Voids first appeared at the core exit at 100 s. Shortly thereafter, the primary began to repressurize in conjunction with the secondary. Voids, formed in the core, moved into the hot leg with a fraction of steam entering the pressurizer surge line. Thus, the mixture level increased as both liquid and steam entered the pressurizer through the surge line (see Fig 4.6). The pressurizer collapsed mixture level reached the 92% full level* at 170 s. The arrival of the
pressurizer mixture level at the vessel top resulted in two-phase flow exhaust from the PORV for the remainder of the transient. Thus, the primary mass exhausted from the PORV became greater than the mass injected by the ECCS (see Fig 4.4). The primary mass decreased for the remainder of the calculated transient.

As the primary pressurized (after 120 s), the secondary continued to pressurize. The atmospheric relief valve (ARV) open setpoint (P=7.8 MPa) was reached at 175 s. Thereafter, the secondary pressure hovered between 7.85 and 7.96 MPa as the modulating ARV adjusted the flow area to maintain the system at a relatively constant pressure.

The system behavior for the remainder of the transient was sluggish. No additional reactor system components were available or capable of affecting the primary or secondary loops. Thus, the reactor system behavior was defined by a balance between the core decay power level, the fluid loss rate through the PORV, the ECCS injection rate, the primary to secondary heat transfer rate and the ARV flow rate. The combination of these variables, determined the primary and secondary flow movement and flow regimes as the transient progressed.

The open secondary ARVs fixed the secondary pressure (the SG secondary temperature) and established a stable boundary condition with respect to the primary. Thus, the primary pressure increased to a average value of 8 MPa (see Fig 4.3). Stable primary and secondary system pressures were achieved by 250s.

As the transient proceeded, the core was cooled by natural circulation as the RCP coastdown was completed at 268 s.

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*The reactor is designed to scram when the pressurizer water level reaches 92 % full. Thus, for definition purposes, a 92 % full pressurizer is assumed to be "full".

Core flow (see Fig 4.7) continued to decrease smoothly until an oscillatory behavior was induced by core and SG heat transfer and void formation at 600 s. The formation and collapse of voids in the core is shown as the core collapsed liquid level increased and decreased after 600 s (see Fig 4.8). Such behavior continued for the duration of the transient.

The behavior of the core mass flow was similar to that of the two loops (see Fig 4.9) for the first 1000 s. The left loop (defined to represent three plant loops and containing the pressurizer) passed about three times more flow than the right loop. However, as the transient progressed and the core steam generation rate increased, the left and right loop flow behaviors diverged. Much of the steam moving in the left loop was removed from the loop volume by passage into the pressurizer surge line. As the core decay power level decreased and the loop mass flow rate decreased steam accumulated at the top of the SG U-tubes in the right loop at a larger rate than in the left loop.

By 1050 s, significant voids became present at the higher elevations of the right loop SG U-tubes such that the right loop flow occassionly stagnated, whereas the left loop flow remained positive. Finally, primary voids were sufficiently large (by 1700 s) to cause intermittent core flow stagnation (see Fig 4.7).

Throughout the calculated transient, the core remained well cooled. Core temperatures at the bottom, the midpoint and the top of the core are shown in Figures 4.10, 4.11 and 4.12. At all three elevations, the core temperatures mirrored the temperature of the adjacent fluid. Thus, the core temperatures reflected the smoothly changing core mass flow behavior up to 600 s. Thereafter, the rod temperatures increased and decreased with the presence and quantity of voids. Also, the minimum temperatures calculated in the core i.e., the nadir of the core temperature fluctuations was seen to decrease as the transient proceeded. Such behavior can be seen at all of the core elevations. In particular, the most pronounced temperature decrease was calculated at the core bottom. Thus, the bottom core rod temperature decreased to 555 K at 1590 s, but at 1815 s to 548 K. This effect is explained by the tendency of the SI water to move down the cold leg and into the core region. As the primary mass flow stagnated, the lower end of the core (and lower plenum) was filled with increasingly cool water.

4.2.2 LSTF thermal-hydraulic behavior

The baseline transient in the LSTF was calculated to behave

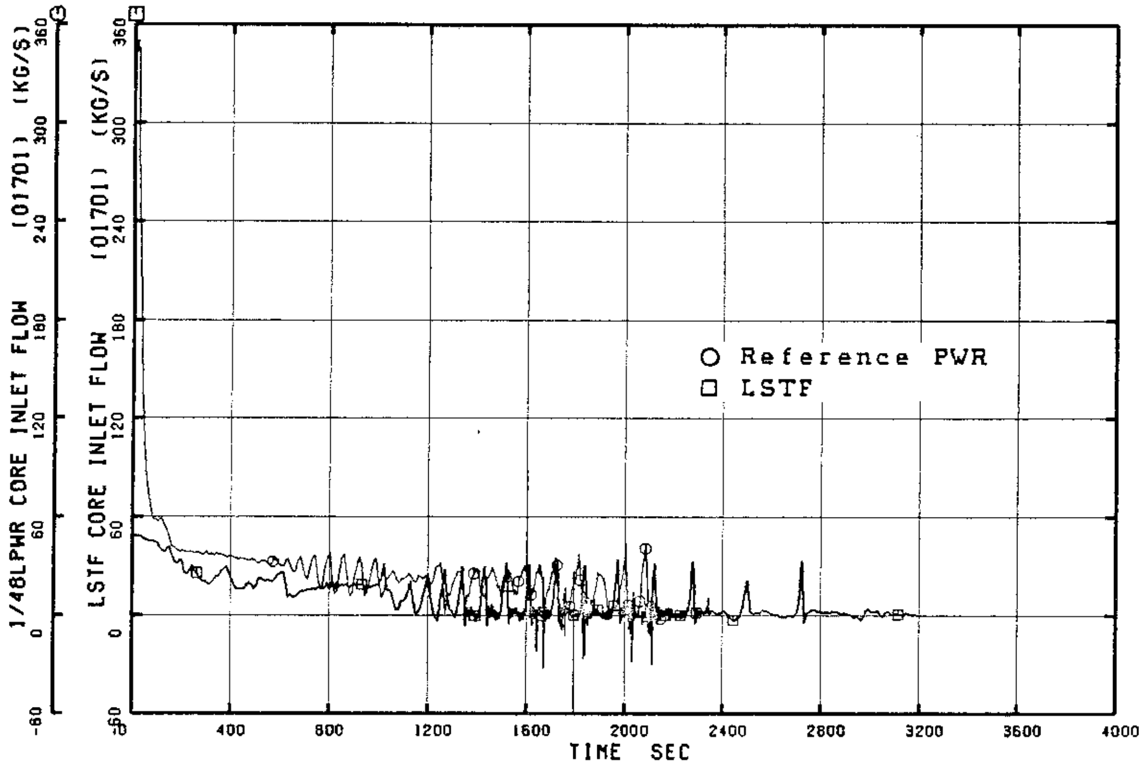


Figure 4.7 Comparison of the LSTF and reference PWR core inlet mass flow: calculation 1

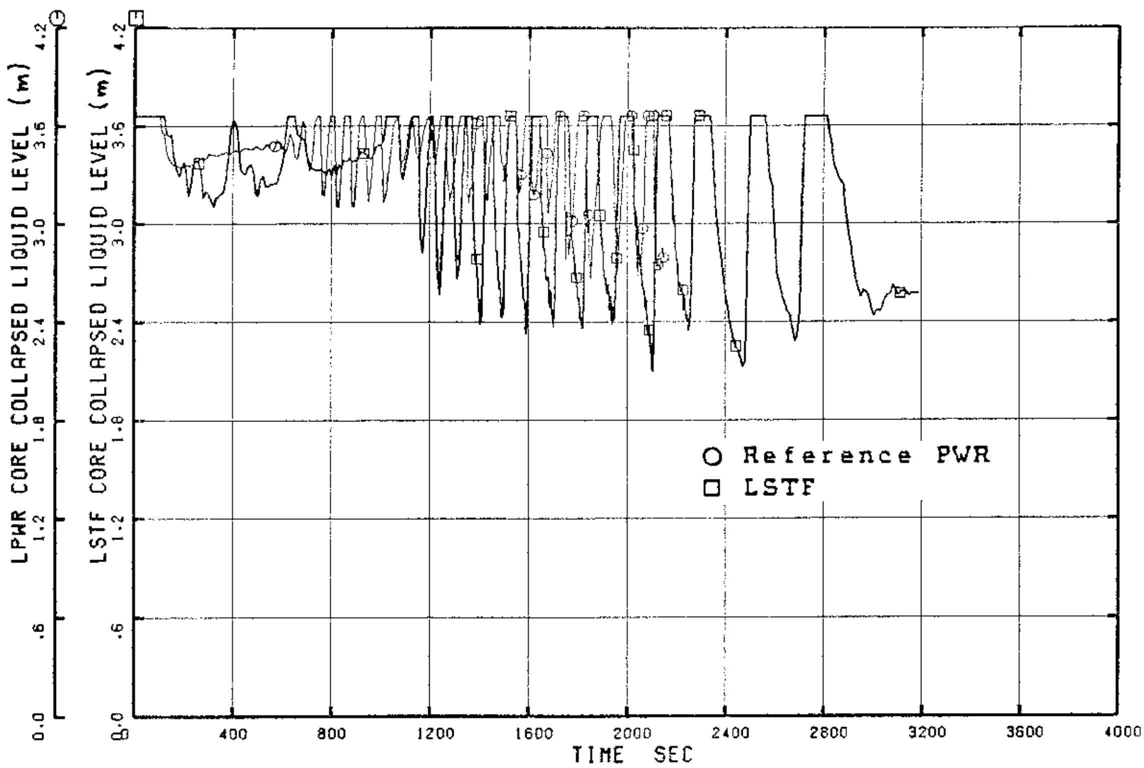


Figure 4.8 Comparison of the LSTF and reference PWR core collapsed liquid level: calculation 1

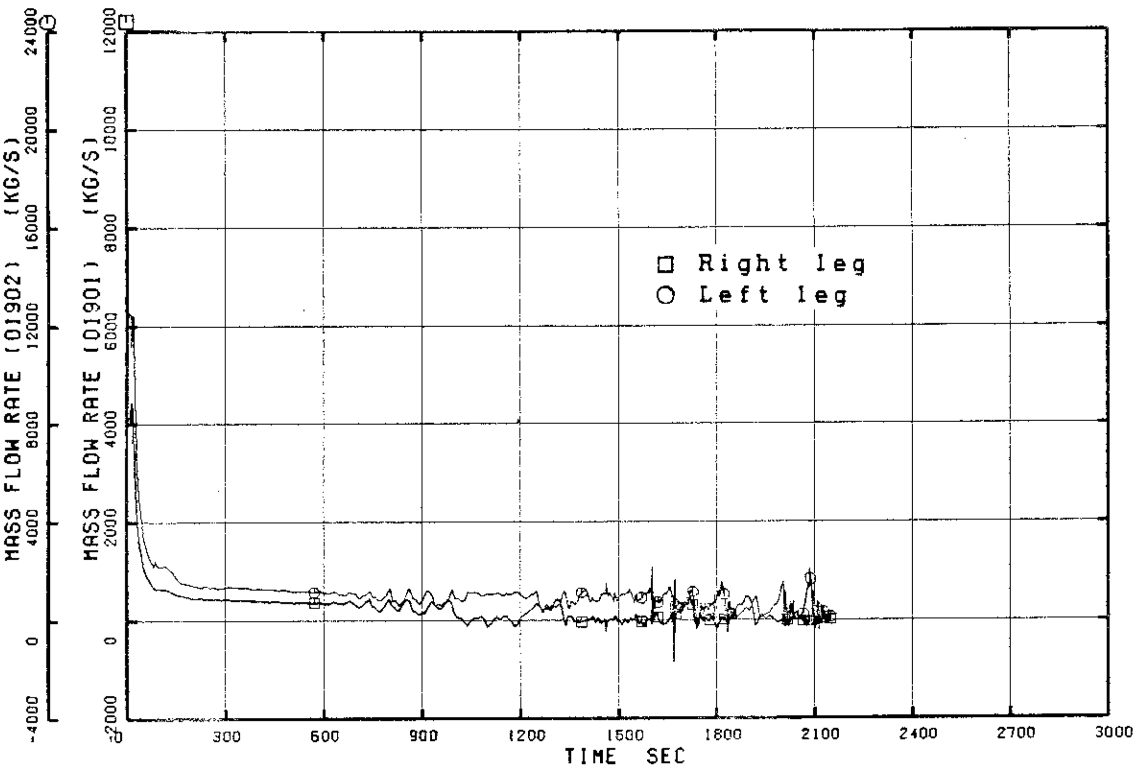


Figure 4.9 Comparison of the reference PWR right and left primary loop mass flow: calculation 1

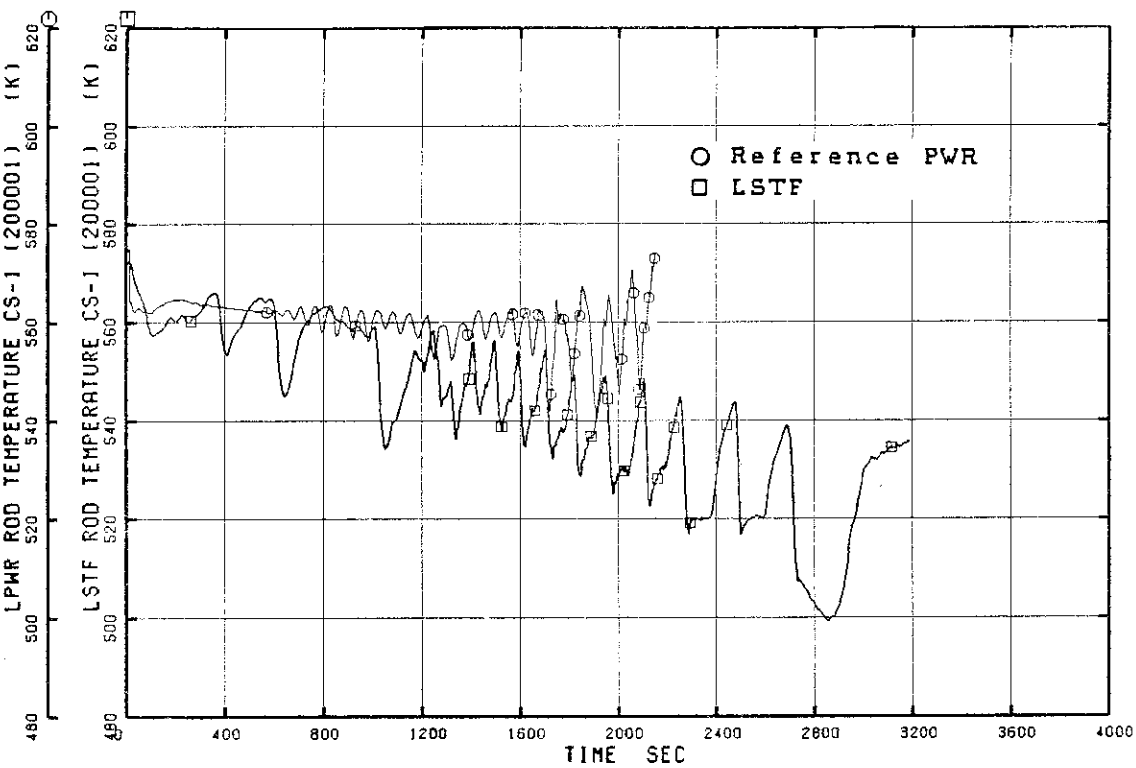


Figure 4.10 Comparison of the LSTF and reference PWR rod surface temperature: calculation 1 - bottom elevation

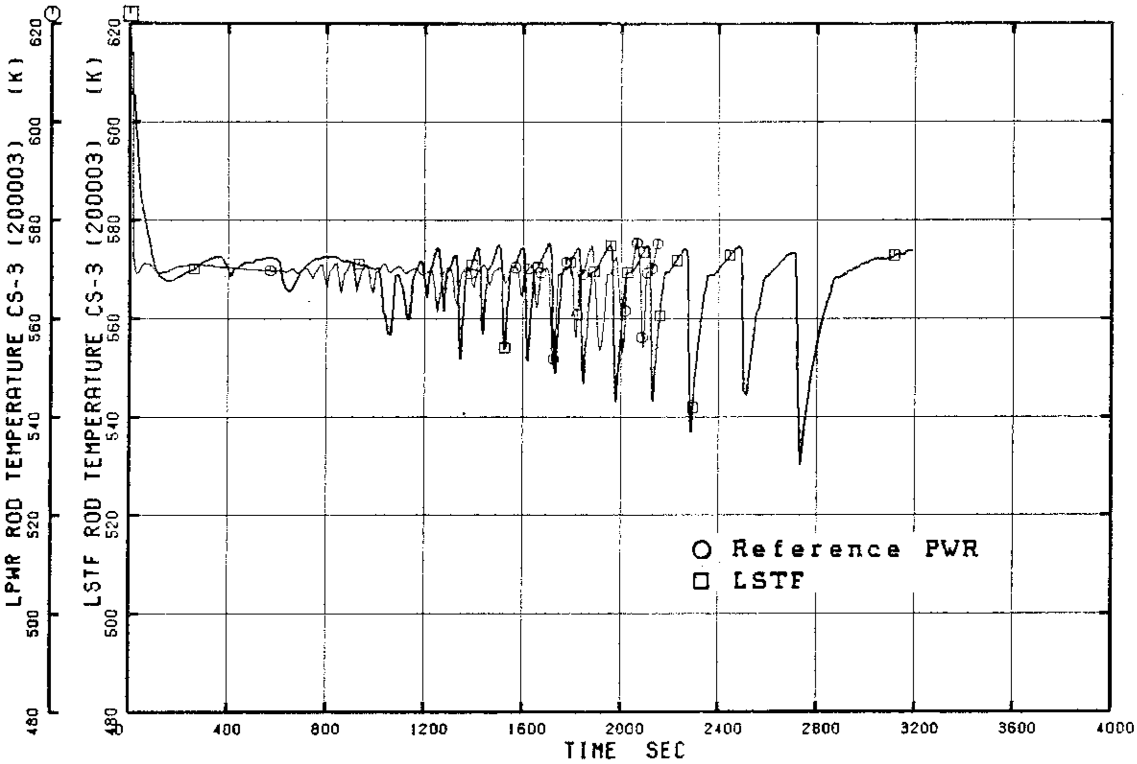


Figure 4.11 Comparison of the LSTF and reference PWR rod surface temperature: calculation 1 - midplane

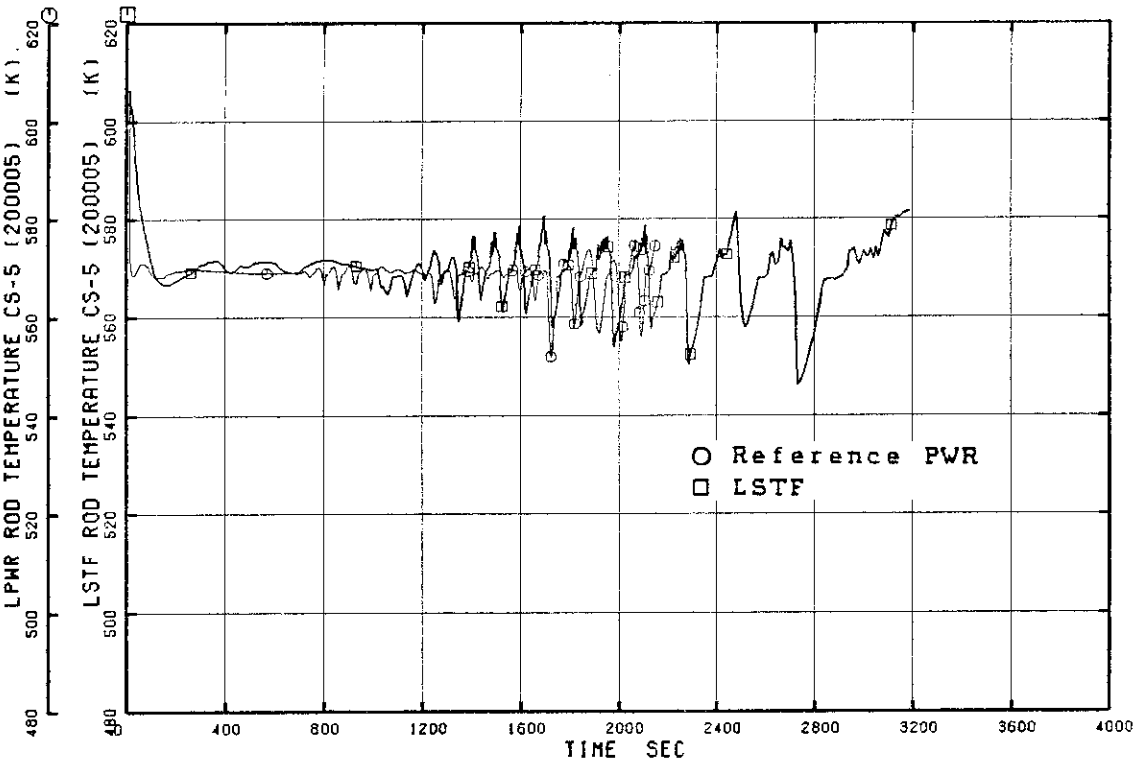


Figure 4.12 Comparison of the LSTF and reference PWR rod surface temperature: calculation 1 - upper elevation

qualitatively the same as the reference PWR. The transient began with a loss of feedwater (see Table 4.1) concurrent with an open PORV.

The SG secondary inventory quickly decreased (see Fig 4.2) as the primary heat load remained at the steady-state value. The turbine throttle valve (see Fig 3.1) was programmed to shut at 11.1 s in agreement with the reference PWR calculated timing. Concurrently, the turbine bypass valve opened. The LSTF power supply was programmed to remain at the rated value until the reference PWR post scram power level reached 14 % i.e., 18.2 s. Thereafter, the LSTF power supply provided power to the LSTF at the same normalized level as the reference PWR.

As the SG secondary water inventory decreased, the primary system pressure decreased due to the mass lost through the stuck open PORV (see Fig 4.3) and the decreasing core power level. The primary system depressurized continuously during the early portion of the transient. By 38.3 s, the primary system pressure was less than 12.27 MPa. Thus, the RCPs tripped off and the high pressure ECCS received a signal to inject. However, ECCS injection did not begin until 57 s, when the cold leg pressure became less than the SI system shutoff head viz., 10.7 MPa. The primary system mass continuously decreased during the first 57 s as shown in Fig 4.4. However, as the LSTF ECCS began to inject fluid into the cold leg, the net primary system mass began to increase.

The LSTF secondary system behaved very similar to the reference PWR (see Fig 4.5). First the secondary depressurized as the primary to secondary heat transfer decreased in combination with the open turbine bypass valves and then increased as the secondary system mass stabilized following closure of the turbine bypass valves at 62s. The increasing secondary pressure (and saturation temperature) caused a reduction in the heat transferred to the secondary. As a result, and because the primary was depressurizing, steam began to form at the core outlet by 120 s. Steam moving from the vessel upper plenum to the SG began to enter the pressurizer surge line. The resulting movement of the pressurizer mixture level was sufficient to cause a two phase mixture to exit the PORV by 160 s (see Fig 4.6). As increased quantities of liquid were exhausted through the PORV, the primary system mass began to decrease again at 200 s (see Fig 4.4). Thus, at 260 s, the primary and secondary system behaviors were governed by the PORV mass flow rate, the ARV flow rate, the primary to secondary heat transfer, the ECCS injection rate and the core power level. The core was cooled totally by natural circulation (after 285 s) as the RCP coastdown was completed and the pumps remained stationary for the duration of the transient. Thereafter, the mass inventory of the secondary decreased continuously. Although the primary system mass decreased on the long term (see Fig 4.4), the primary system mass intermittently increased as the PORV flow cycled between void fractions of zero and a value between zero and one (see Fig 4.6). Thus, the ECCS injected flow was not sufficient to maintain the primary system mass when the PORV effluent was totally liquid.

The primary to secondary heat transfer continued to degenerate such that voids were apparent at the core midplane by 210 s. At 260 s the secondary pressure exceeded the ARV setpoints. Thereafter the secondary pressure was maintained at approximately 7.8 MPa by the modulating ARVs.

As the pressurizer mixture level began to oscillate (by 300 s) to allow first a two phase mixture followed by single phase liquid effluent, an oscillatory loop flow behavior (see Fig 4.13) began. The left and right loop flows moved out of phase with one another. It is thought that the oscillatory loop flows were induced by the oscillatory PORV discharge.

The LSTF core mass flow continued to decrease (see Fig 4.7) and stagnated briefly at 1150 s. Thus, the core steam production rate increased as shown by the core collapsed water level (see Fig 4.8). The core collapsed water level decreased markedly at 1195 s. Comparison between Figures 4.7 and 4.8 shows the core collapsed water level to decrease at low or zero core flows such that as the core oscillated the core steam formation rate also oscillated.

Continued core stagnation caused the volume of the saturated core fluid to increase with time. By 1800 s, almost 75% of the core contained saturated fluid.

The oscillatory behavior of the core flow caused the heater rod temperatures to oscillate (see Figs 4.10, 4.11 and 4.12). Correlation between the core mass flow behavior and the rod surface temperatures shows that as the core flow decreased, the rod temperature increased. Such behavior was observed throughout the core in the calculation.

Comparison of the rod temperatures at the three elevations shows the average rod temperature at the core bottom and midplane to be decreasing with time. Such behavior is indicative of the SI fluid moving from the injection point to the pressure vessel. Thus, even though the primary system sustained a net mass loss after 200 s (since the PORV mass flow exceeded the SI injected flow), the LSTF core remained well cooled throughout the transient.

4.2.3 The reference PWR and LSTF calculations compared

The objective of the following discussion is to relate the

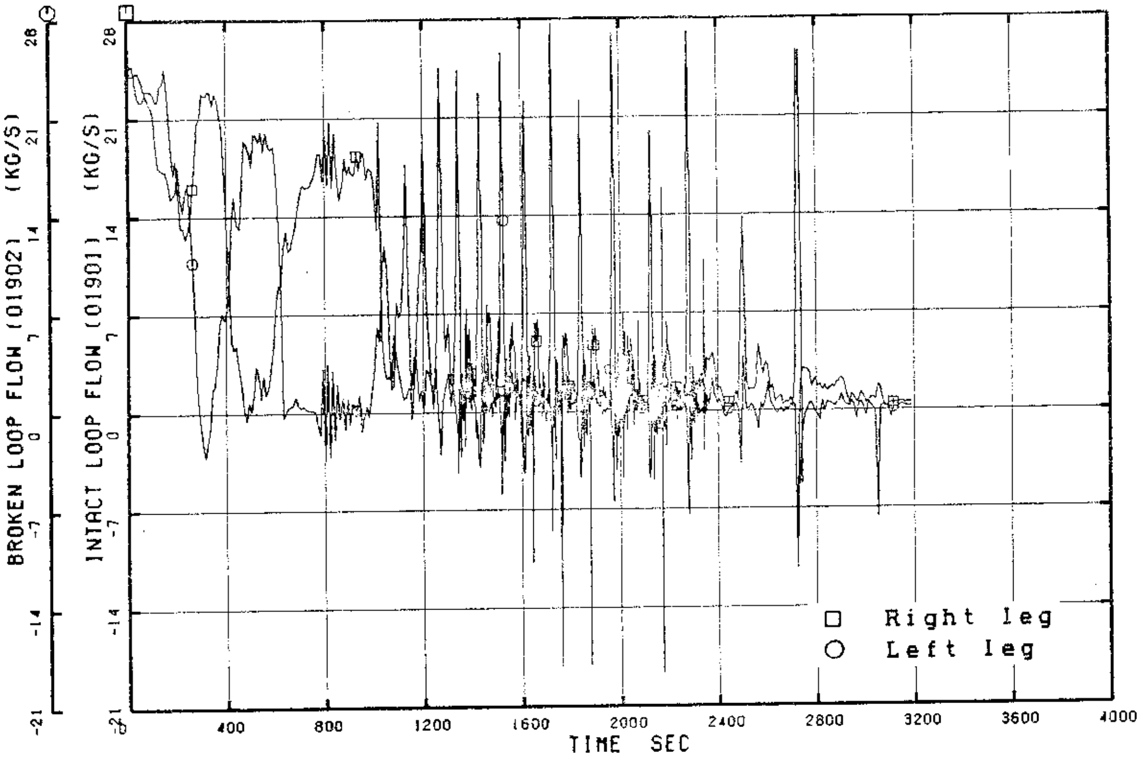


Figure 4.13 Comparison of the LSTF right and left primary loop mass flow: calculation 1

differences between the reference PWR and LSTF calculations shown in Figures 4.2 through 4.13 to both the physical and model differences between the plant and the test facility.

4.2.3.1 Initial conditions and the early portion of the transient

The initial SG secondary water levels of the LSTF and the reference PWR were different (see Table 3.1 and Fig 4.2) even though both systems had the same mass inventory (on a scaled basis). The difference is directly related to the quantity of steam present in the secondary. The LSTF only had the capability of simulating 14 % of the plant scaled power. Thus, the LSTF secondary steam formation rate was lower than the plants and the mixture level also was correspondingly lower. Such a difference was only apparent at steady state conditions and only affected the transient from the perspective of plant trip times i.e., low and low-low SG secondary water levels. For this reason, the trip times listed in Table 4.1 for the LSTF are flagged as not applicable (NA).

Other differences between the reference PWR and the LSTF are listed in subsection 3.1.3. The fact that the LSTF only provided 14 % of the plant rated scaled power also influenced the RCP operation and secondary initial pressure. The LSTF RCPs only needed to provide a rated head of 0.01 MPa (a factor of 63 less than the reference PWR). Further, to maintain the desired primary conditions, the secondary pressure was maintained at 7.14 MPa (1.43 MPa greater than the reference PWR rated condition).

The above factors influence the first portion of any transient. For the TMI-2 scenario, as the main feedwater system failed, the SG secondary quickly decreased. But the LSTF water level was reduced at a lower rate than the reference PWR since less core power per unit secondary inventory mass was transferred (see Fig 4.2). As the transient proceeded, the reference PWR SG water level trip elevation was reached at 11.1 s. As the reactor scrammed, the steam production rate decreased rapidly. Concurrently, the secondary circulation flow rate decreased rapidly. These factors caused a rapid decrease in the downcomer indicated water level. Neither of these factors were present in the LSTF system.

Following scram, the RCPs in the LSTF and the reference PWR continued at the steady-state value until the primary pressure decreased to 12.27 MPa (see Table 3.2). Because the reference PWR pumps pumped a factor of 7 more primary flow, and because the primary to secondary temperature difference in the reference PWR was greater than in the LSTF (due to the high LSTF secondary pressure), the reference PWR would depressurize more quickly than the LSTF regardless of the open PORV. Superior primary to secondary heat transfer also resulted in the reference PWR turbine bypass value closure at 14s (as the primary average loop temperature became less than 564.9 K) compared to closure at 62s for the LSTF (see Fig 4.5). Thus the reference PWR primary system depressurized to 12.29 MPa by 17.4 s whereas the LSTF reached the same pressure level at 38.3 s (see Fig 4.3)

4.2.3.2 The transient during RCP coastdown

During the portion of the transient when the RCPs were shut off and coasted to zero impellor velocity, the reference PWR had substantially more primary flow than the LSTF due to the pump head differences. As such, the reference PWR again had a greater primary to secondary heat transfer rate than the LSTF. Consequently, the reference PWR secondary repressurized more rapidly than the LSTF such that the ARV setpoint was reached in the reference PWR by 175 s. The LSTF ARV setpoint was reached at 280 s.

The difference in primary flow rate between the systems (see Fig 4.7) resulted in the LSTF primary steam production rate to exceed that of the reference PWR on a scaled basis. Thus, the lower core mass flow rate calculated for the LSTF resulted in core steam formation from the midplane to the exit by 210 s. The reference PWR didn't experience core steaming at the midplane until 600 plus seconds.

The difference in core mass flow rate between the LSTF and reference PWR was not due solely to the RCP head. Other factors which influenced the disparity in the core mass flow rates between the two systems are listed in subsection 3.1.3 i.e. a 14 % larger pump loss, a larger grid spacer loss and a 16.78 m core bottom to SG top elevation difference in the LSTF. The importance of these factors increased with time as the RCP coastdown neared completion.

4.2.3.3 The transient after RCP coastdown

The fact that the LSTF produced more steam (scaled value) than the reference PWR caused more severe pressurizer water level behavior to occur in the LSTF (see Fig 4.6). As the transient proceeded, the LSTF primary mass flow rate decreased noticeably more rapidly than the reference PWR. In turn, the LSTF core steam production rate increased more rapidly than that of the reference PWR.

The baseline LSTF model calculation was conducted with the pressurizer 50 % full initially. The reference PWR initial pressurizer level was 42 %. The initial pressurizer levels, combined with the larger core steam production rate caused the LSTF pressurizer to fill completely by 300 s such that liquid water was exhausted from the PORV.

Rapid changes in the pressurizer water level and thus the PORV discharge void fraction possibly induced the primary flow oscillatory behavior observed in the LSTF (see Fig 4.13). In contrast, the loop to loop primary flow behavior in the reference PWR was quite different (see Fig 4.9). Although analysis has not been conducted to investigate the relative tendency of the two equal loop LSTF system to oscillate, compared with the reference PWR geometry, the authors believe the available system damping in the plant four loop design will tend to preclude similar oscillations from occurring in the reference PWR.

As the transient proceeded, the LSTF core mass flow briefly reached zero at 1150 s. The reference PWR core mass flow did not stagnate until 1700 s.

5. PARAMETRIC CALCULATIONS

In addition to the baseline analyses, nine other calculations were conducted to explore the system behavior of a W type four loop PWR and the LSTF as a W four loop PWR simulator (see Table 3.3). The content of this section briefly discusses the remaining nine calculations and their relationships to the baseline calculations.

5.1 The Baseline Transient With ECCS Shutoff at 200s: Calculation 2

One of the TMI-2 scenario characteristics (see Table 1.1) was caused by operator action. Due to misinterpretation of the pressurizer water level, the operators decreased the plant injection flows (both ECCS and charging flow) when the pressurizer became full. Such a boundary condition was the basis for conducting calculation 2. As discussed in Section 4.2.1, the reference PWR pressurizer collapsed water level reached the 92 % full mark at 170 s (see Fig 5.1). Allowing 30 s for the operator to respond, the two operational SI pumps were switched off at 200 s.

To duplicate the action of the reference PWR system operator, the LSTF operator was also assumed to switch off the two operational LSTF SI pumps at 200 s. The LSTF pressurizer collapsed water level was 81 % at 170 s.

Calculation 2 for both the reference PWR and the LSTF was identical to calculation 1 respectively for the first 200 s of the transient.

5.1.1 Reference PWR

Until 200 s, the SI injection flow exceeded the system mass loss rate through the PORV. As the ECCS injection was terminated, the primary system mass immediately began to decrease (see Fig 5.2). In addition, the core average temperature increased. Thus, the average loop temperature exceeded 567.7 K and the turbine bypass valve reopened. In addition, the atmospheric relief valves opened intermittently as the secondary pressure exceeded 7.78 MPa. Thus, the secondary mass loss through the bypass valves and ARVs maintained the SG secondary pressure between 7.2 and 7.85 MPa (see Fig 5.3) for the transient duration. Since the turbine bypass valve open area was set as a function of the loop mean temperature, the secondary pressure oscillated as a function of the loop temperature. Thus, the primary temperature (and the pressure - see Fig 5.4) also oscillated

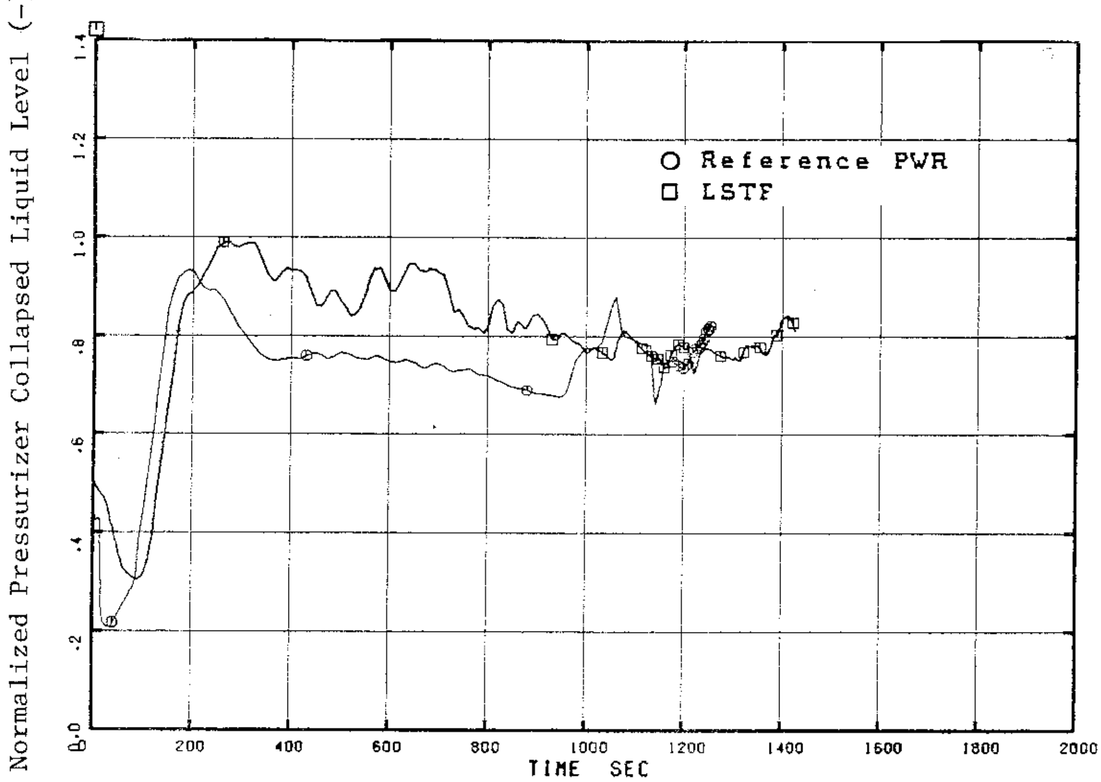


Figure 5.1 Comparison of the LSTF and reference PWR pressurizer collapsed liquid level: calculation 2

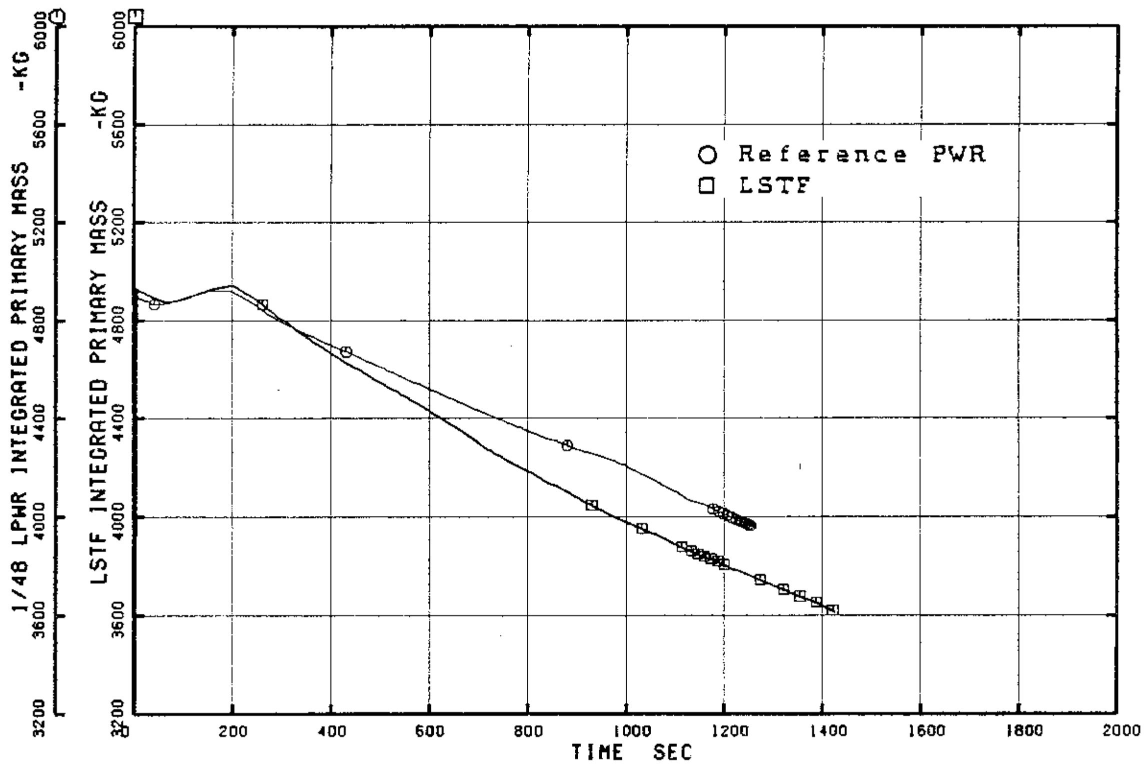


Figure 5.2 Comparison of the LSTF and reference PWR primary mass: calculation 2

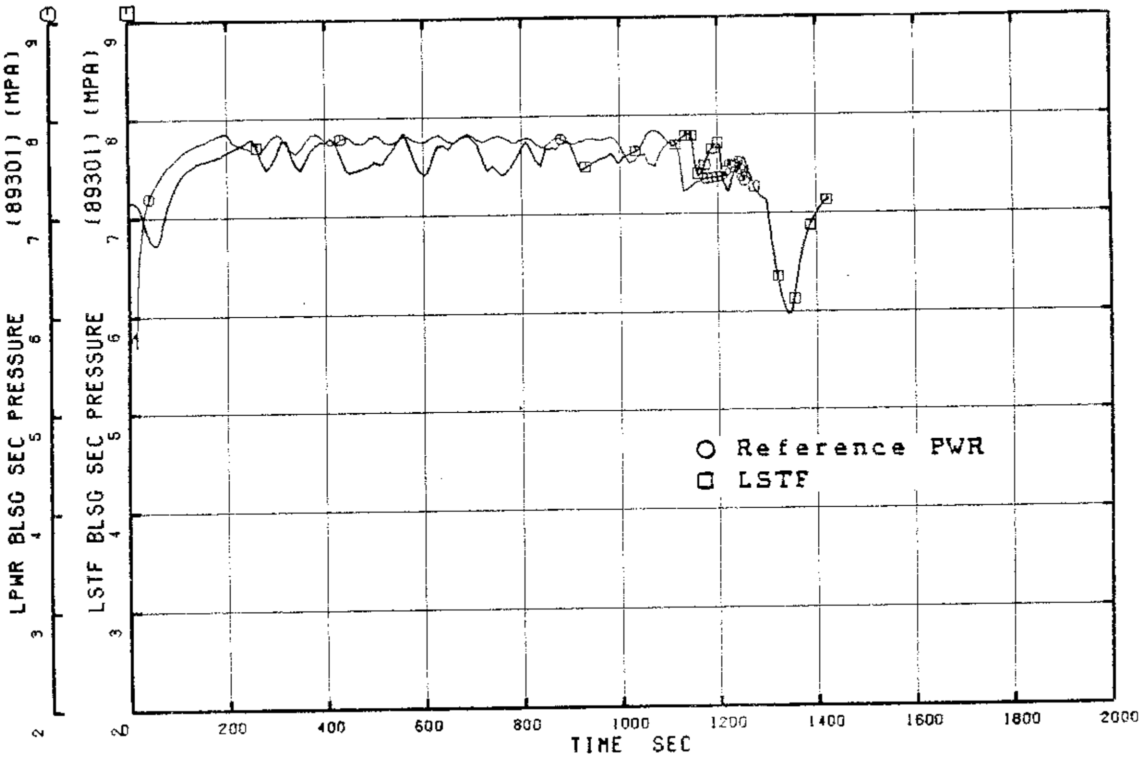


Figure 5.3 Comparison of the LSTF and reference PWR secondary pressure: calculation 2

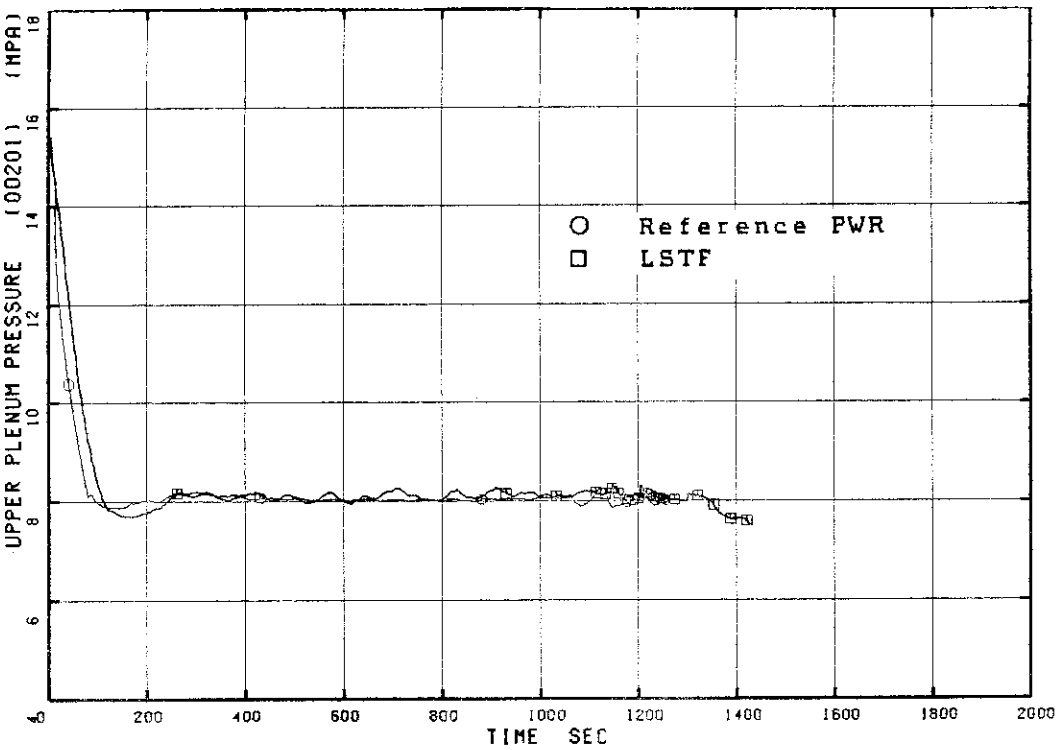


Figure 5.4 Comparison of the LSTF and reference PWR upper plenum pressure: calculation 2

as the primary to secondary heat transfer first increased and then decreased.

As the transient proceeded, the PORV mass flow remainded two phase (see Fig 5.1) and the core mass flow slowly decreased (see Fig 5.5). The left and right loop flow rates remained at a ratio of 3 to 1 (see Fig 5.6) until 800 s when the right loop flow decreased and then stagnated at 820 s. The core inlet mass flow began to stagnate intermittently at 1050 s.

The effect of decreased core mass flow was apparent immediately in the behavior of the core collapsed liquid level (see Fig 5.7). Whereas the core collapsed liquid level remained at a relatively constant value for most of the transient, the level decreased first to 2.6m (at 1000 s) and then to nearly 2.3m (at 1110 s), when the inlet flow stagnated.

The fuel rod temperatures followed the adjacent fluid temperatures. Thus, the rod temperatures throughout the core (see Figs 5.8, 5.9 and 5.10) remained at a relatively constant value i.e., 569 K, due to the action of the turbine bypass and ARVs. But after 1000 s, the core rod temperatures began to increase intermittently (see Fig 5.10), as the core collapsed water level reached minimum values. However, the reference PWR core remained well cooled for the calculated transient.

5.1.2 LSTF

The LSTF behaved qualitatively the same as the reference PWR. However, the pressurizer collapsed liquid level contiued to rise after ECCS injection ceased as a steam and liquid mixture moved into the pressurizer volume (see Fig 5.1). Thus, the LSTF PORV exhausted a lower quality fluid than the reference PWR during most of the transient. Consequently, the LSTF primary system mass decreased more rapidly than the reference PWR (see Fig 5.2).

Since the LSTF secondary pressure was lower than the reference PWR at 200 s (see discussion - subsection 4.2.3), the LSTF primary to secondary heat transfer was adequate to maintain the LSTF primary loop average temperature below the turbine bypass valve setpoint until 240 s. Thereafter, the LSTF turbine bypass and ARVs opened to maintain a relatively constant primary loop average temperature and to maintain the secondary pressure below 7.78 MPa. As a result, the secondary pressure (see Fig 5.3) oscillated between 7.4 and 7.8 MPa for most of the transient.

Since the PORV flow void fraction changed over a relatively

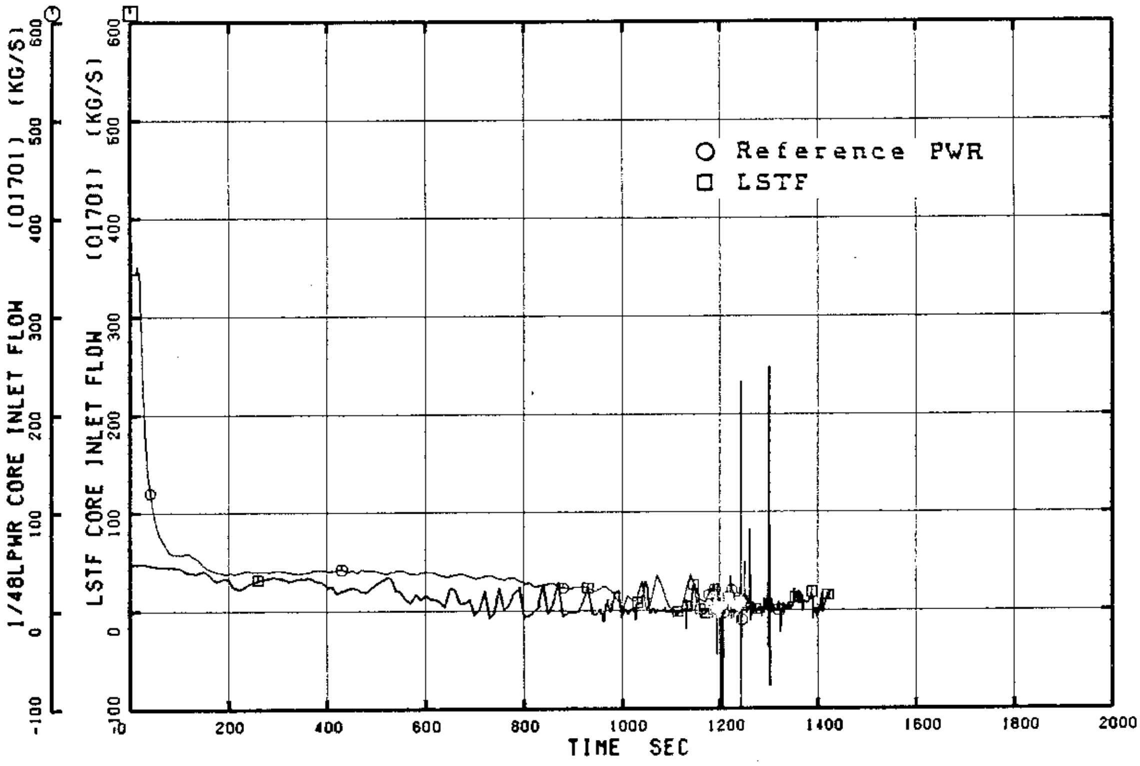


Figure 5.5 Comparison of the LSTF and reference PWR core inlet mass flow: calculation 2

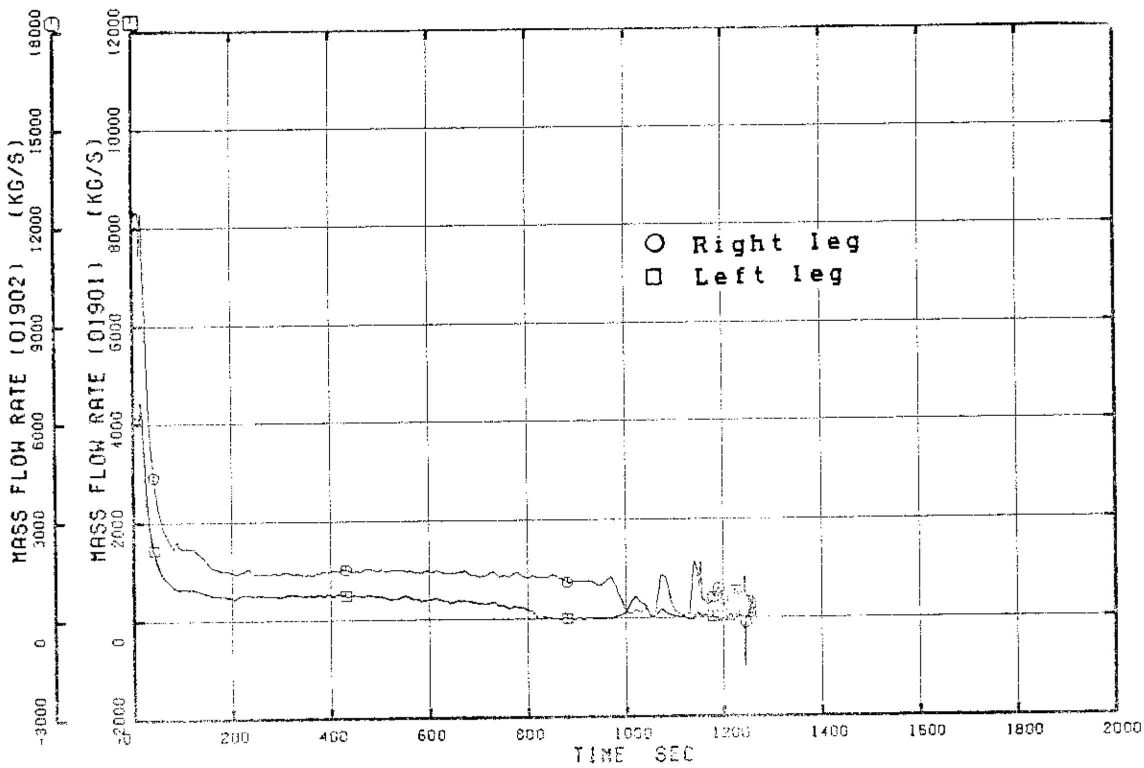


Figure 5.6 Comparison of the reference PWR right and left primary loop mass flow: calculation 2

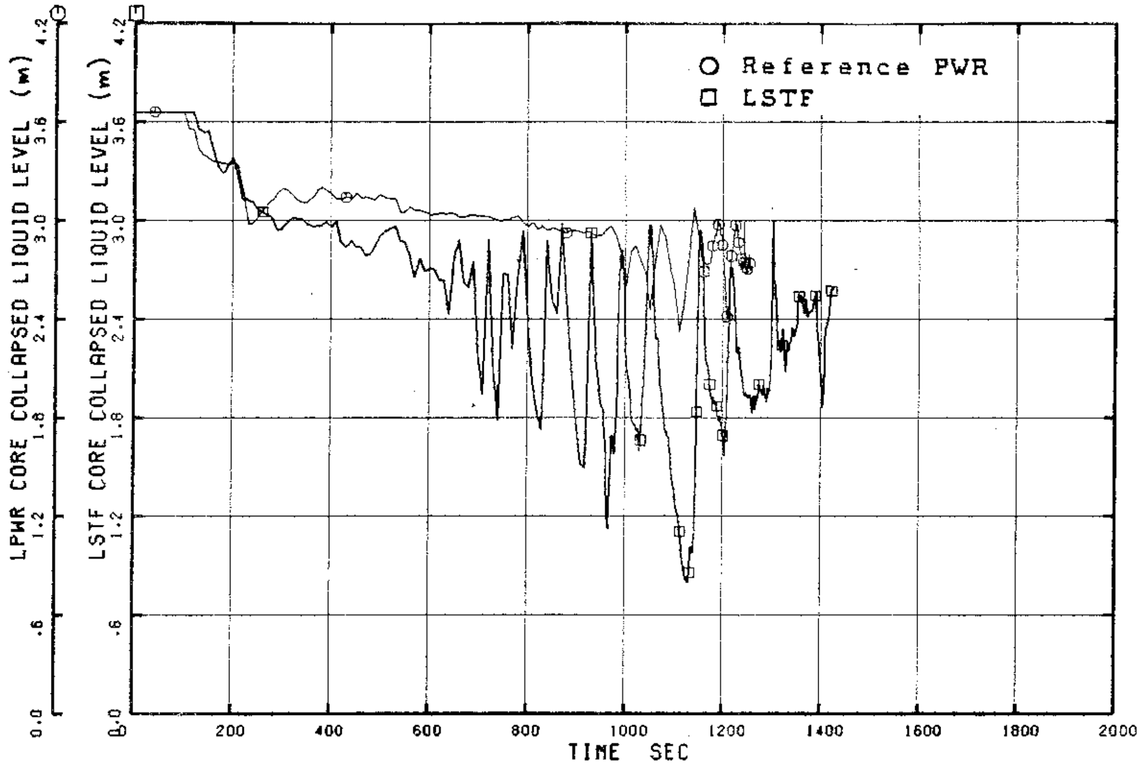


Figure 5.7 Comparison of the LSTF and reference PWR core collapsed liquid level: calculation 2

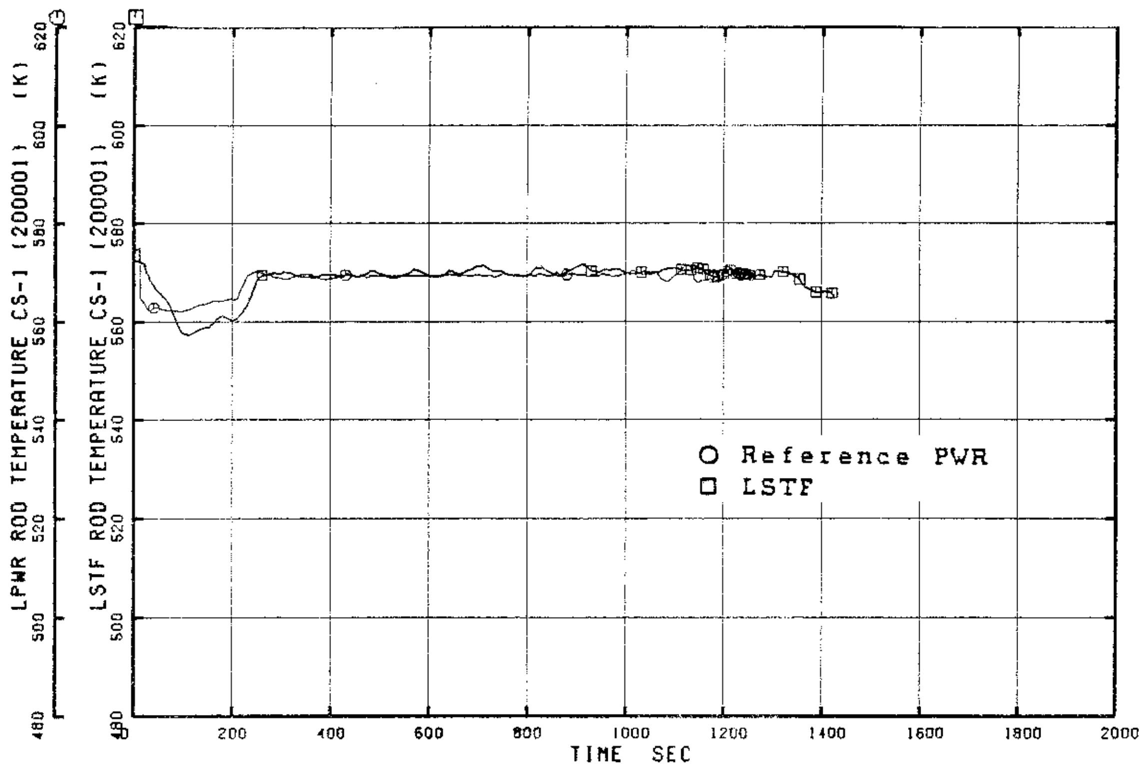


Figure 5.8 Comparison of the LSTF and reference PWR rod surface temperatures: calculation 2 - lower elevation

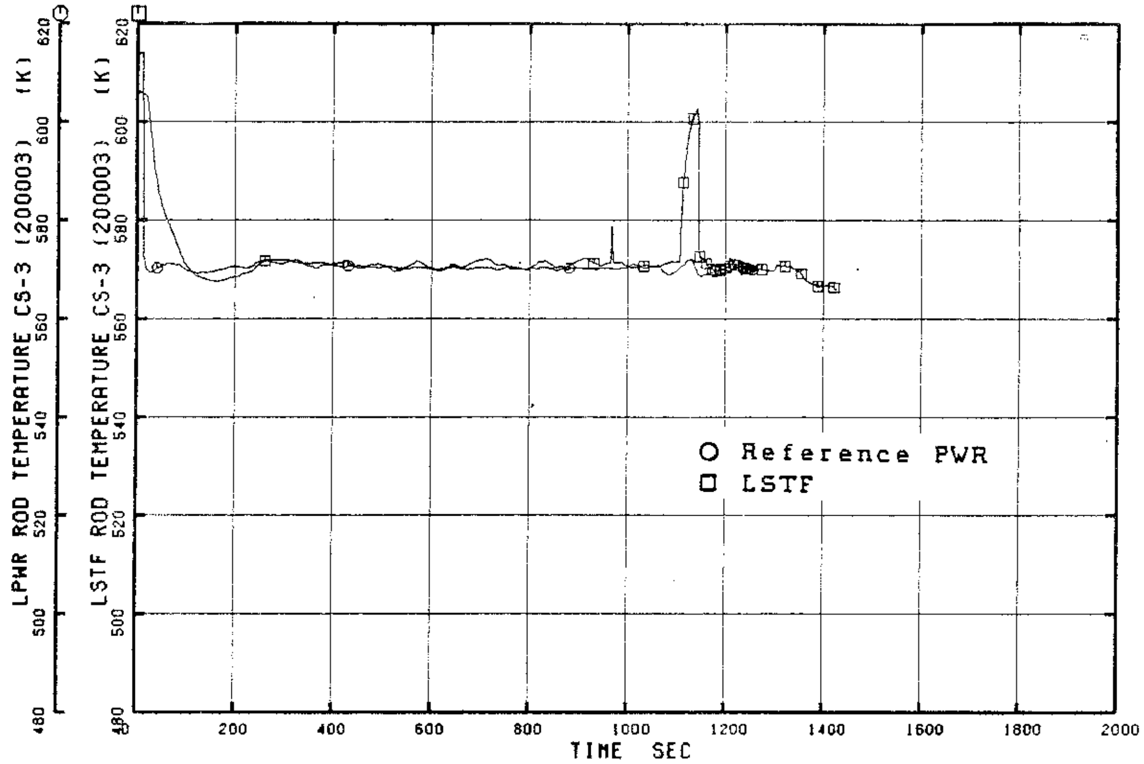


Figure 5.9 Comparison of the LSTF and reference PWR rod surface temperatures: calculation 2 - midplane

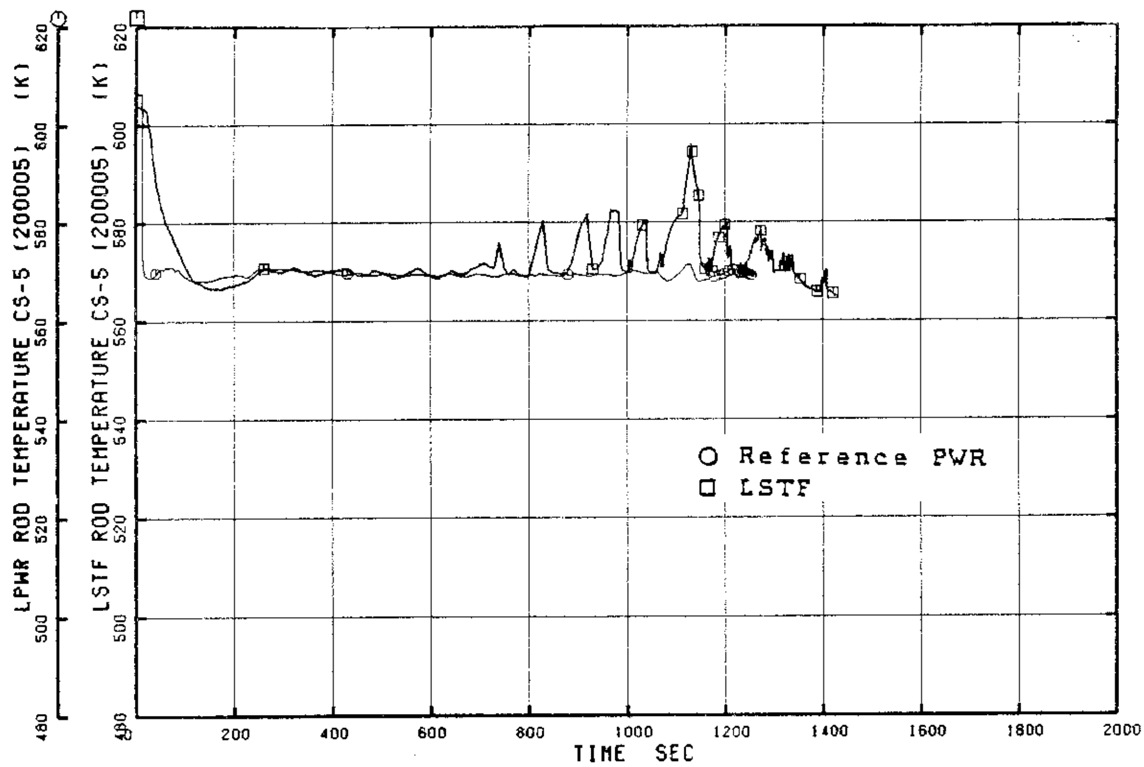


Figure 5.10 Comparison of the LSTF and reference PWR rod surface temperatures: calculation 2 - upper elevation

wide range, the PORV discharge flow oscillated over a relatively wide range. The left and right loop flows oscillated (see Fig 5.11) similar to the behavior observed in calculation 1.

The LSTF primary pressure (see Fig 5.4) also oscillated as it was influenced by the pressurizer exhaust flow, the oscillatory loop to loop flow and the secondary turbine bypass and ARV behavior. However, the LSTF pressure was maintained at a nominal value i.e., 8.1 MPa, equivalent to the reference PWR for the transient duration.

The LSTF core mass flow was lower than the reference PWR throughout the transient (see Fig 5.5). The core flow stagnated frequently after 700 s. Thus, core void formation was enhanced and the core collapsed liquid level, greater than 2.4 m previously, began to decrease to below the midplane by 740 s.

Although the core remained well cooled at the bottom (see Fig 5.8) throughout the transient, the core elevations from the midplane to the top (see Figs 5.9 and 5.10) experienced sporadic heatup between 700 s and the end of the calculation. The maximum core heatup was calculated to be 605 K at the core midplane at 1130 s. as the core collapsed water level decreased to 0.8 m.

Core mass flow stagnation, combined with rising heater rod temperatures caused the primary average fluid temperature to rise markedly. The turbine bypass valve, sensitive to the primary average fluid temperature, opened wide at 1300s to increase the primary to secondary heat transfer.

5.2 The Baseline Transient With One SI Pump: Calculation 3

A variation of the baseline calculation with less available ECCS equipment was investigated to complete the parametric study with an open nominal area PORV. Calculation 3 has exactly the same boundary and initial conditions as calculation 1, except only one SI pump was available.

5.2.1 Reference PWR

The reference PWR behavior predicted in calculation 3 followed the same initial path as the baseline until 35 s, when ECCS injection was initiated. Thereafter, the calculation 3 transient differed from the baseline, only as a result of the primary mass balance (as influenced by the available SI equipment).

The primary system pressure (see Fig 5.12) stabilized at 8.1 MPa (at 200 s). By then, the primary system mass was

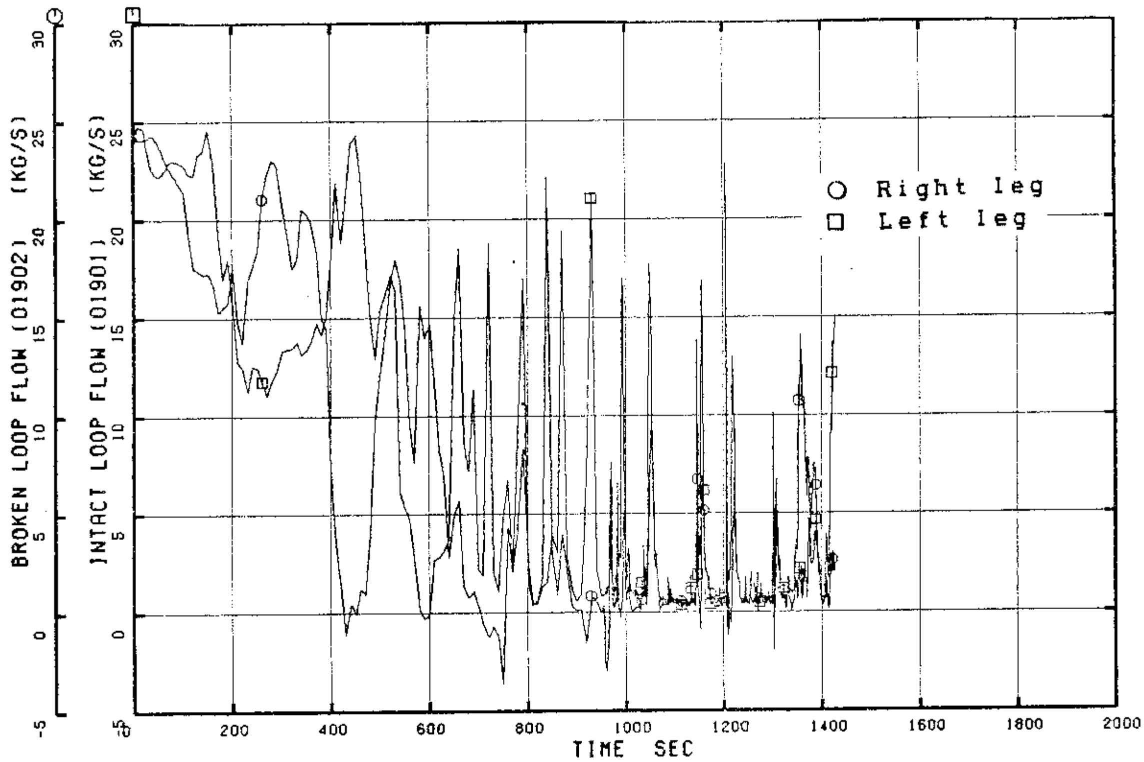


Figure 5.11 Comparison of the LSTF right and left primary loop mass flows: calculation 2

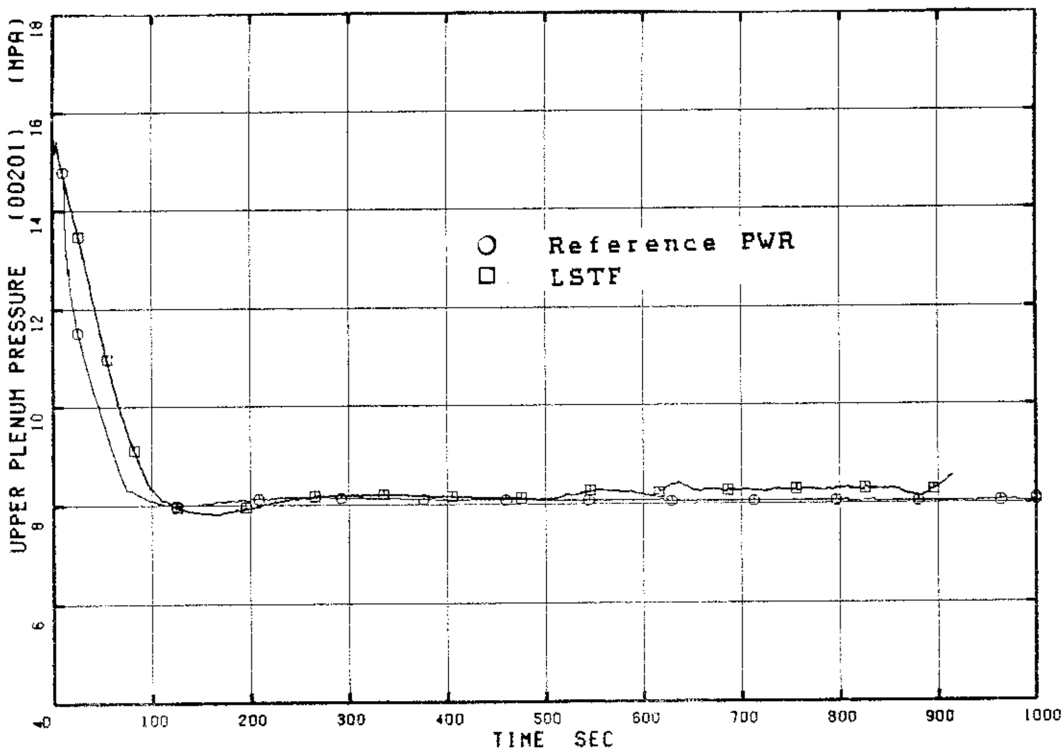


Figure 5.12 Comparison of the LSTF and reference PWR upper plenum pressure: calculation 3

decreasing (see Fig 5.13) as the single SI pump was unable to match the PORV exhaust flow. The left and right loop mass flows (see Fig 5.14) remained at a 3 to 1 ratio until 1060 s when the right loop flow stagnated. The core mass decreased throughout the transient (see Fig 5.15), but did not stagnate. Consequently, the core collapsed liquid level remained above the 3.15 m elevation (see Fig 5.16) and the core remained well cooled (see Fig 5.17).

5.2.2 LSTF

The predicted calculation 3 LSTF behavior duplicated the baseline transient for the first 57 s (when ECCS injection was initiated). Thereafter, the calculation 3 transient differed from the baseline only as a result of less SI injected flow.

The primary system pressure stabilized at 8.2 MPa by 300 s (see Fig 5.12). The primary system mass loss was greater than the reference PWR since the PORV exhausted lower void fraction fluid (see Fig 5.13). Loop to loop oscillatory flow was experienced (see Fig 5.18) for most of the calculated transient. The LSTF calculated core mass flow was less than the reference PWR (see Fig 5.15) and stagnated first at 625 s and then again at 880 s. Although the core collapsed water level (see Fig 5.16) remained above 2.9 m prior to 600 s, the effect of core flow stagnation was to decrease the collapsed water level first to 2.35 m (at 625 s) and then to 2.05 m (at 910 s). Thus, the core began to experience limited heatup (see Fig 5.17). The maximum calculated core temperature was 580 K at 910 s.

5.3 The Calculations With Rated Flow PORVs

The TMI-2 scenario calculations were also conducted with a rated flow PORV using the same variations in ECCS injection equipment as discussed in the previous section (see Table 3.3). Because calculations 4,5 and 6 have a rated flow PORV, combined with rated flow ECCS components, the system predicted behavior will be more realistic. Calculations 4,5 and 6 are presented and discussed briefly in the following three subsections.

5.3.1 Rated flow PORV transient: Calculation 4

Calculation 4 differed from the baseline only in the PORV flow area. The PORV area which exhausted plant rated steam flow was approximately half the size of the rated flow area.

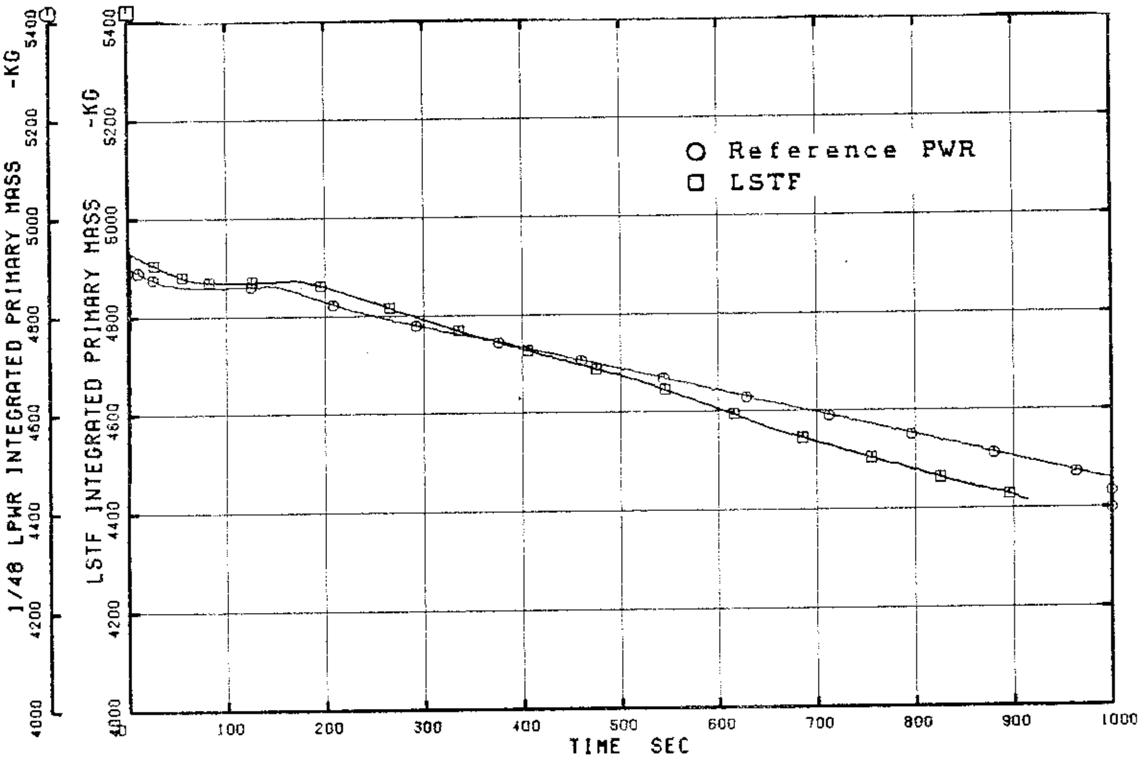


Figure 5.13 Comparison of the LSTF and reference PWR primary mass: calculation 3

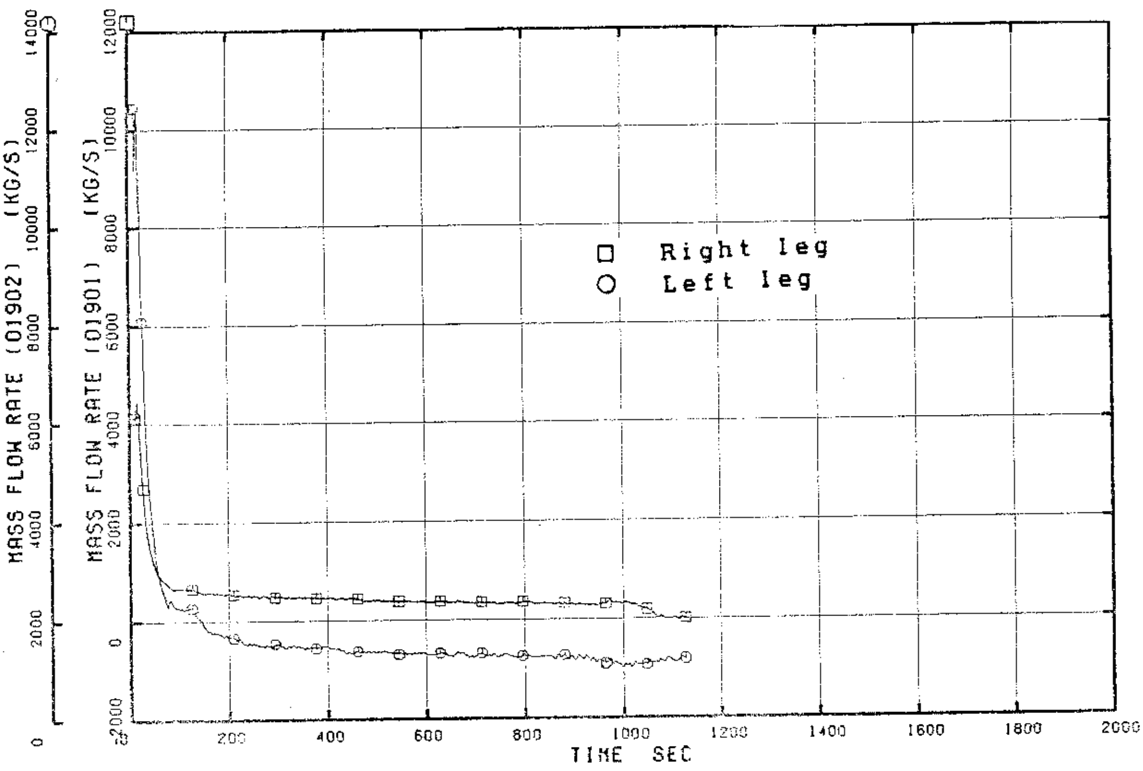


Figure 5.14 Comparison of the reference PWR right and left primary loop mass flows: calculation 3

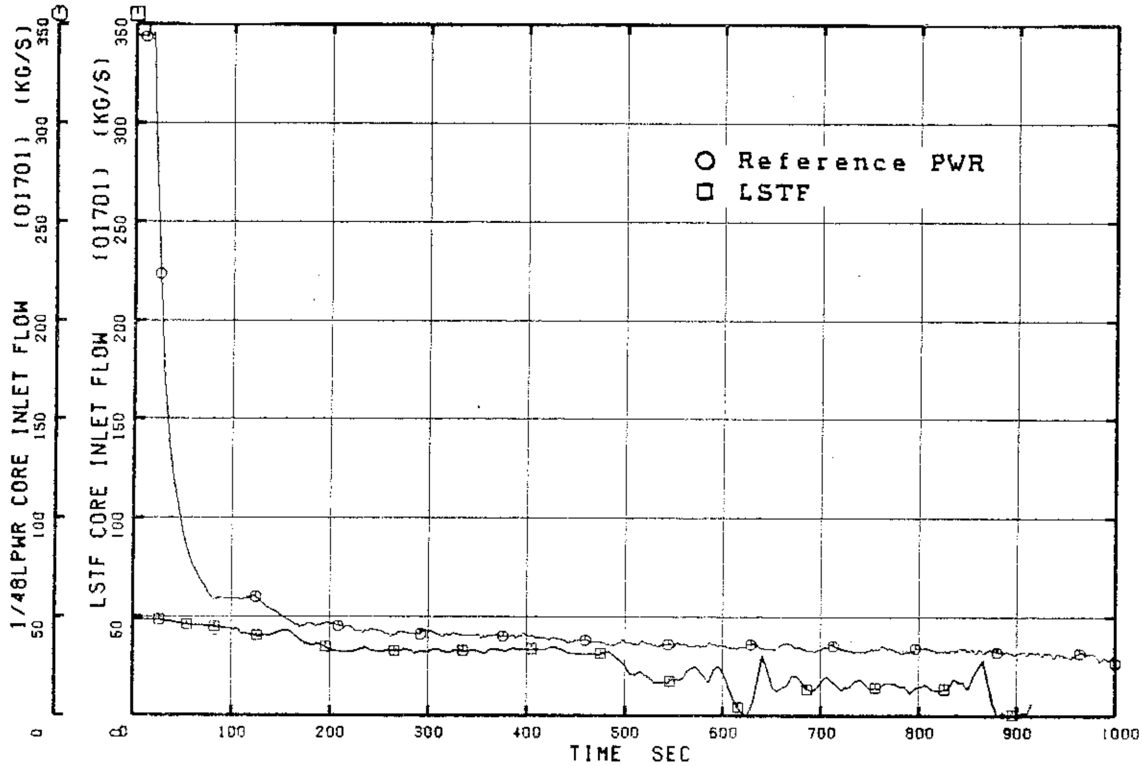


Figure 5.15 Comparison of the LSTF and reference PWR core mass flows: calculation 3

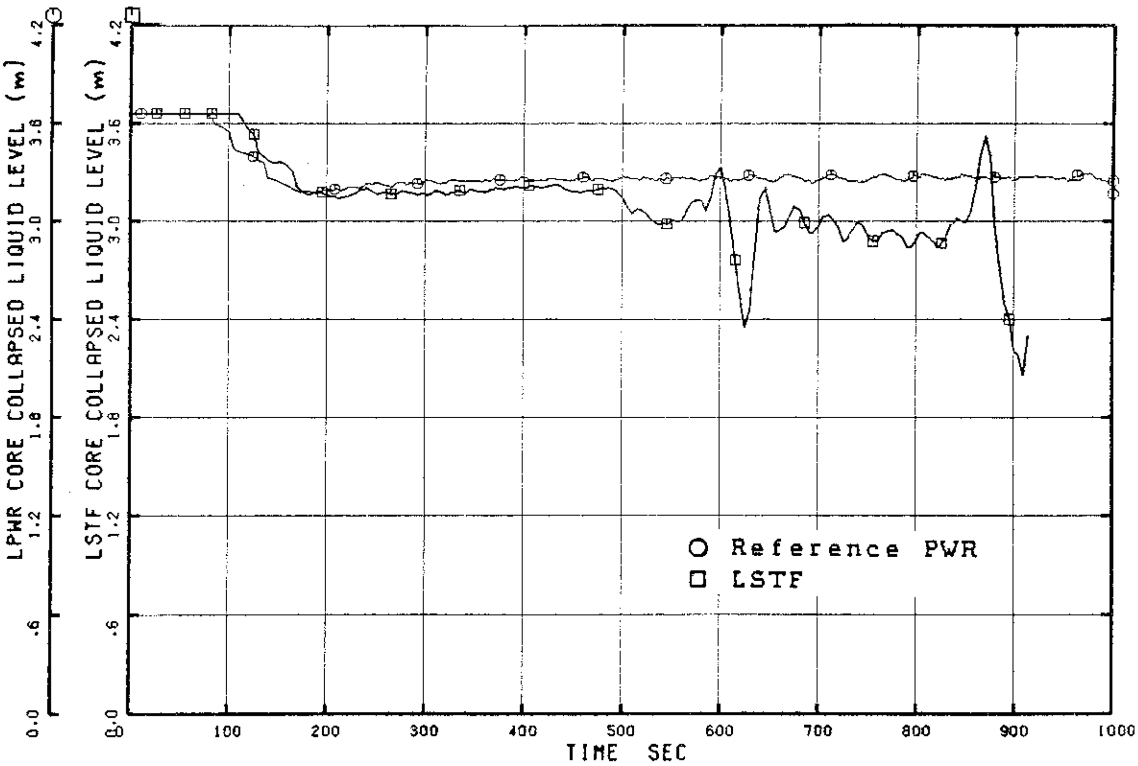


Figure 5.16 Comparison of the LSTF and reference PWR core collapsed liquid level: calculation 3

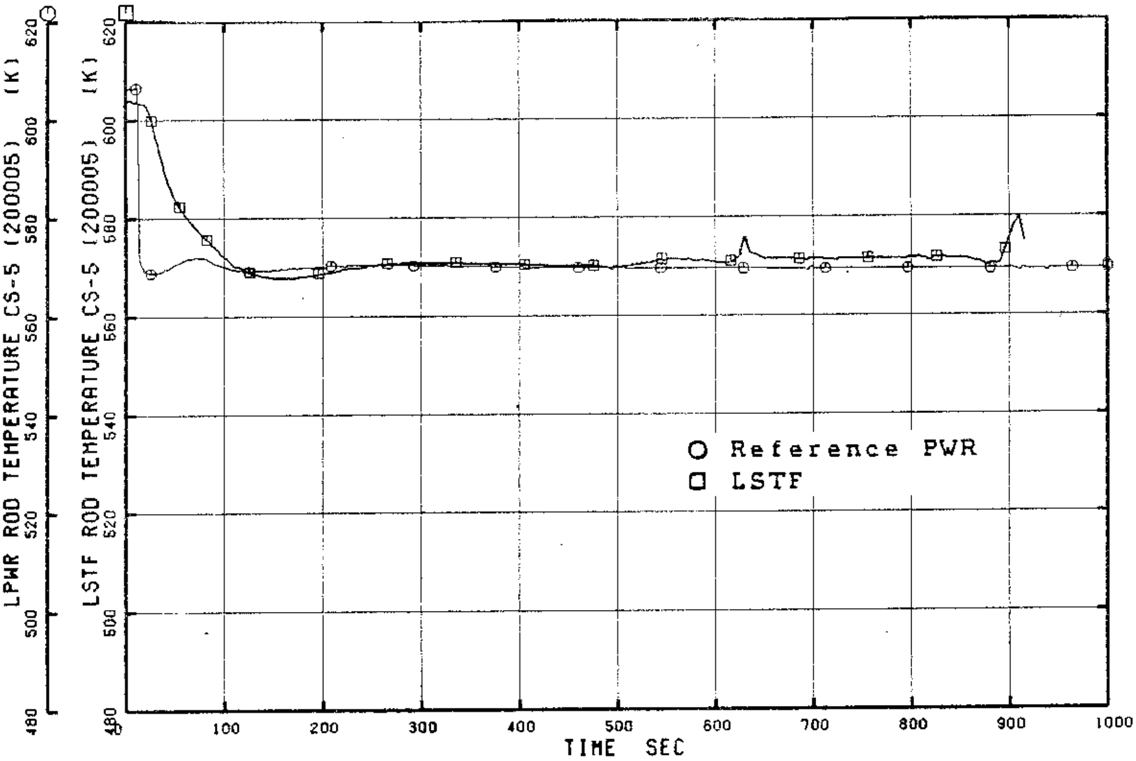


Figure 5.17 Comparison of the LSTF and reference PWR rod surface temperatures: calculation 3

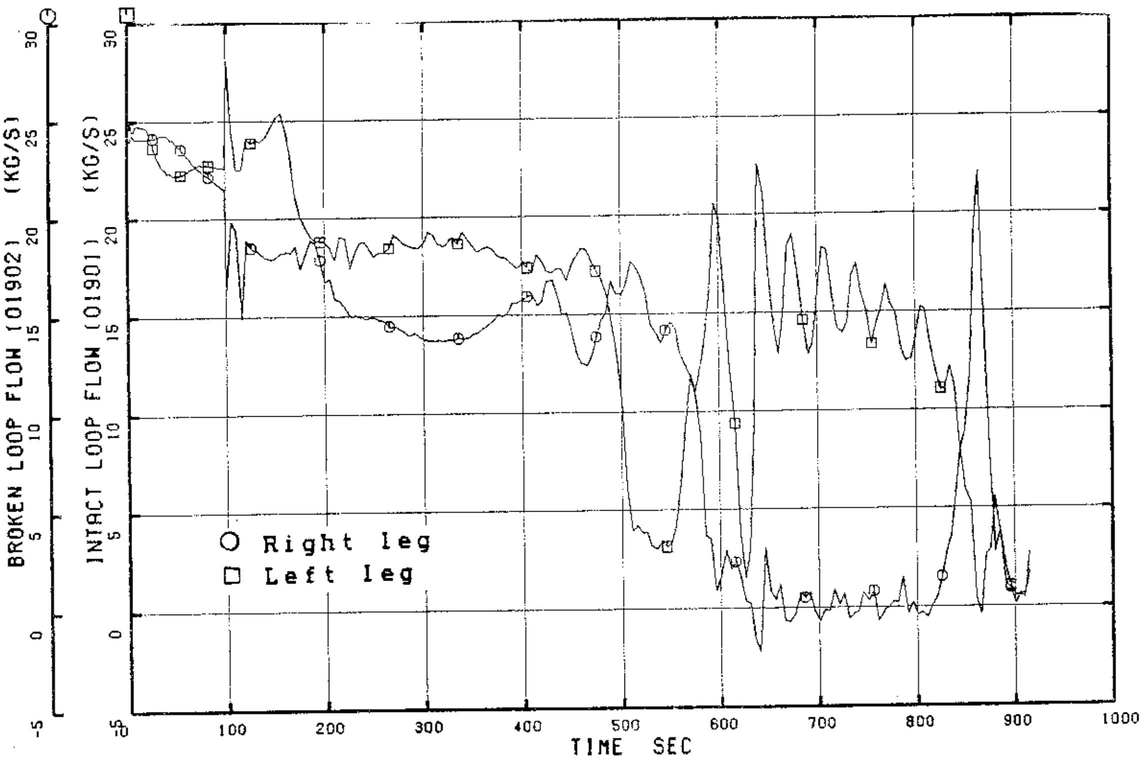


Figure 5.18 Comparison of the LSTF right and left primary loop mass flows: calculation 3

The difference in the PORV flow area between the baseline and calculation 4 is apparent in the transient depressurization rates (see Fig 5.19). Calculation 4 depressurized to 8.7 MPa in 150 s, whereas the baseline took 75 s. The slower depressurization rate, experienced in the calculation 4 transient delayed the RCP trip to 35 s and ECCS injection to 82 s. The effect of ECCS injection in the calculation 4 transient was more pronounced than in the baseline, since the mass loss through the PORV was less. Thus, the primary system mass inventory quickly began to increase at 82 s (see Fig 5.20) and continued to increase throughout the calculation 4 transient. The primary system pressure began to increase at 340 s as the system filled. Finally, the pressurizer became solid (see Fig 5.21) by 560 s.

5.3.2 Rated flow PORV transient with ECCS shutoff at 400 s:Calculation 5

The fifth calculation is analogous to calculation 2. Because calculation 5 had a rated flow PORV, the transient events occurred at roughly double the times of the calculation 2 transient. Thus, the reference PWR pressurizer collapsed liquid level reached the 92 % mark at 370 s (see Fig 5.22). Subsequently, the ECCS injection was switched off in both the reference PWR and LSTF calculations at 400 s.

5.3.2.1 Reference PWR

Following ECCS injection termination at 400 s, the primary mass inventory (see Fig 5.23) immediately began to decrease. Shortly thereafter, the turbine bypass valves opened (see Fig 5.24) in an effort to maintain the average primary loop temperature below 567.7 K such that the secondary was depressurized to 6.8 MPa by 450 s. The effect of the secondary depressurization was apparent in the primary system pressure (see Fig 5.25). Whereas the primary system was repressurizing shortly after 400 s, the action of the turbine bypass valve had depressurized the system to 8.2 MPa by 540 s. Such interactions between the primary and secondary continued throughout the calculated transient.

The core mass flow, stabilized at 286 s as the RCPs coastdown was completed (see Fig 5.26), remained relatively constant for the remainder of the calculation. As such, the core was well cooled.

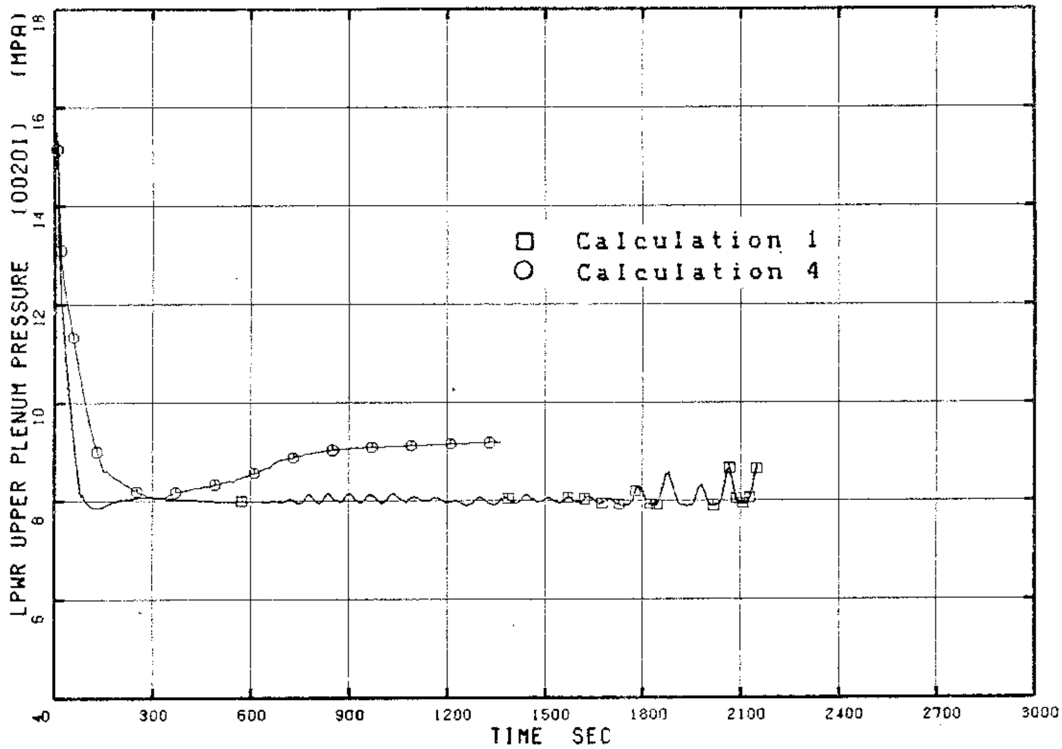


Figure 5.19 Comparison of the reference PWR upper plenum pressure behavior: calculations 1 and 4

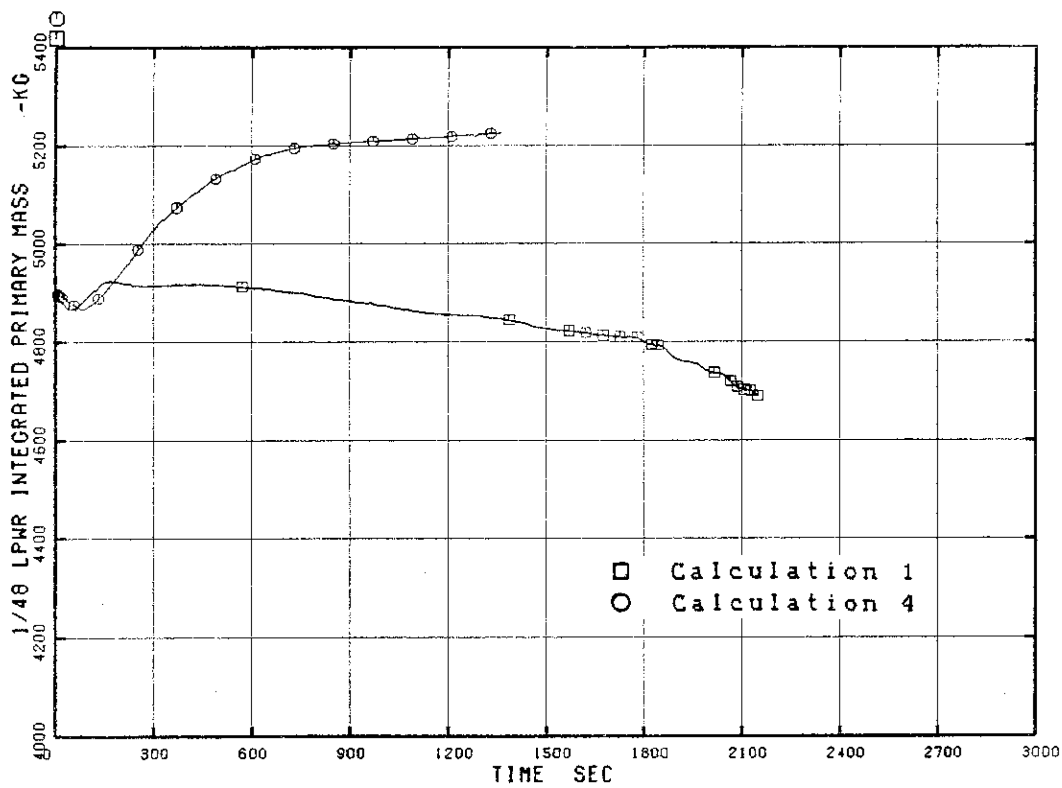


Figure 5.20 Comparison of the reference PWR primary mass: calculations 1 and 4

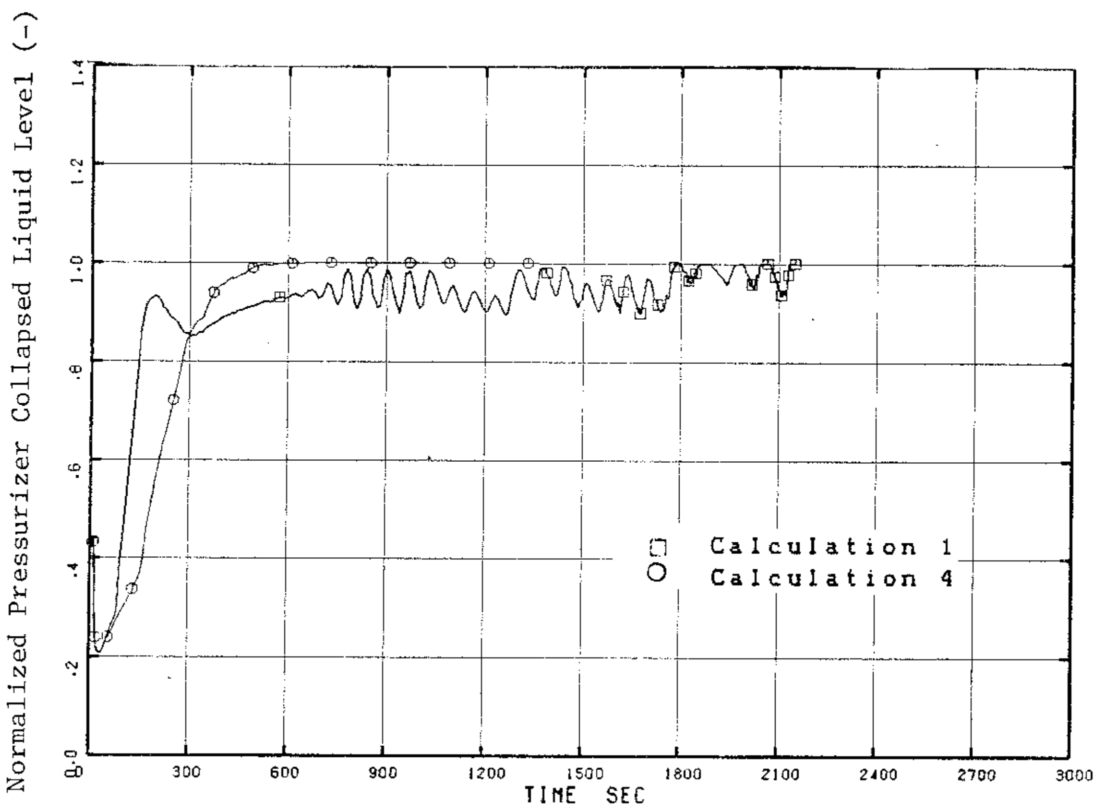


Figure 5.21 Comparison of the reference PWR pressurizer collapsed liquid level: calculations 1 and 4

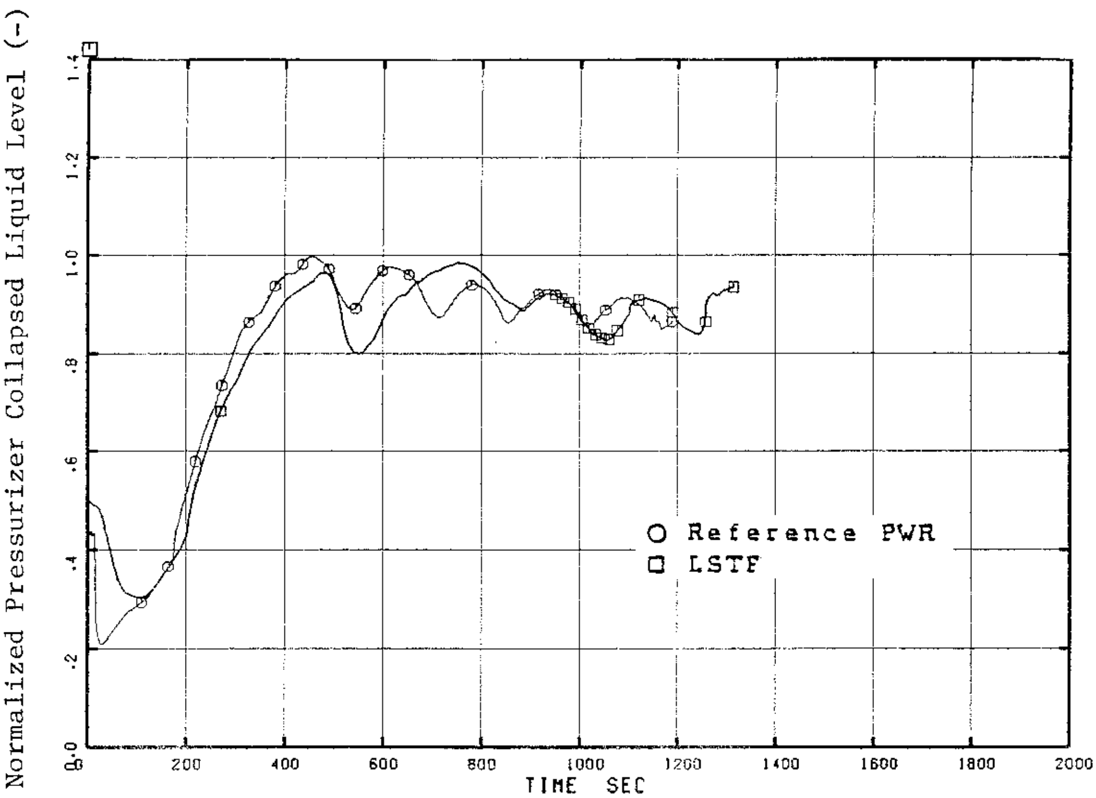


Figure 5.22 Comparison of the LSTF and reference PWR pressurizer collapsed liquid level: calculation 5

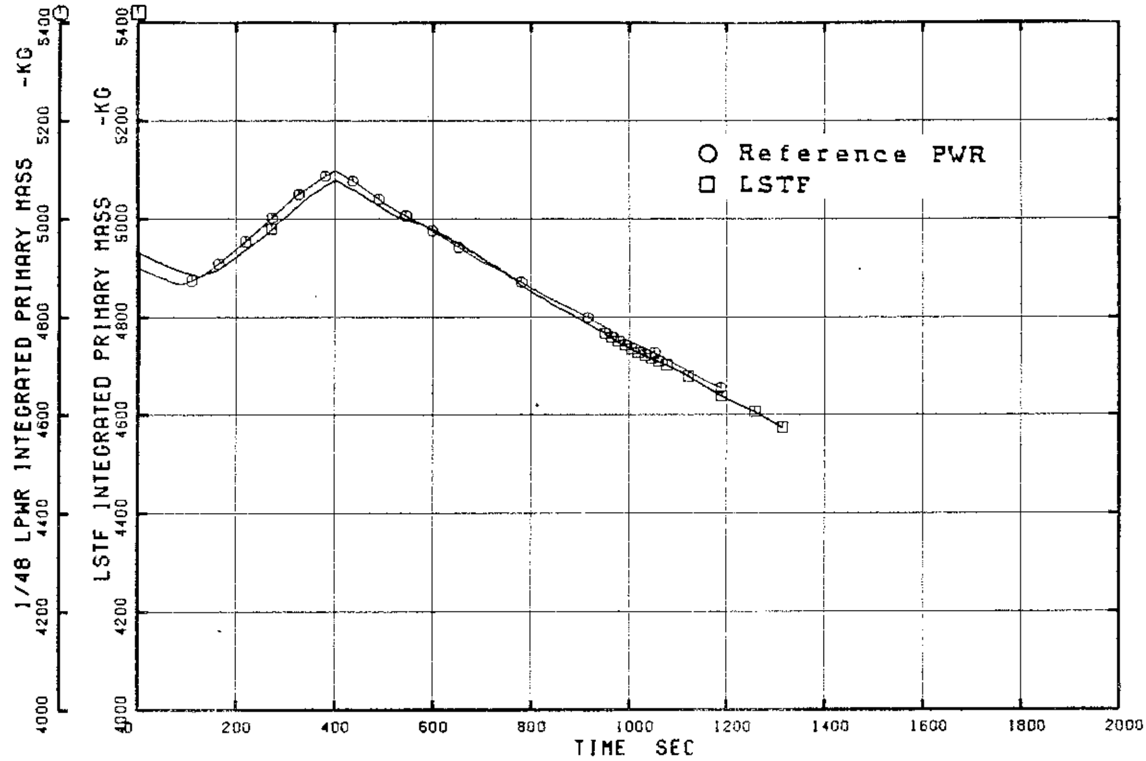


Figure 5.23 Comparison of the LSTF and reference PWR primary mass: calculation 5

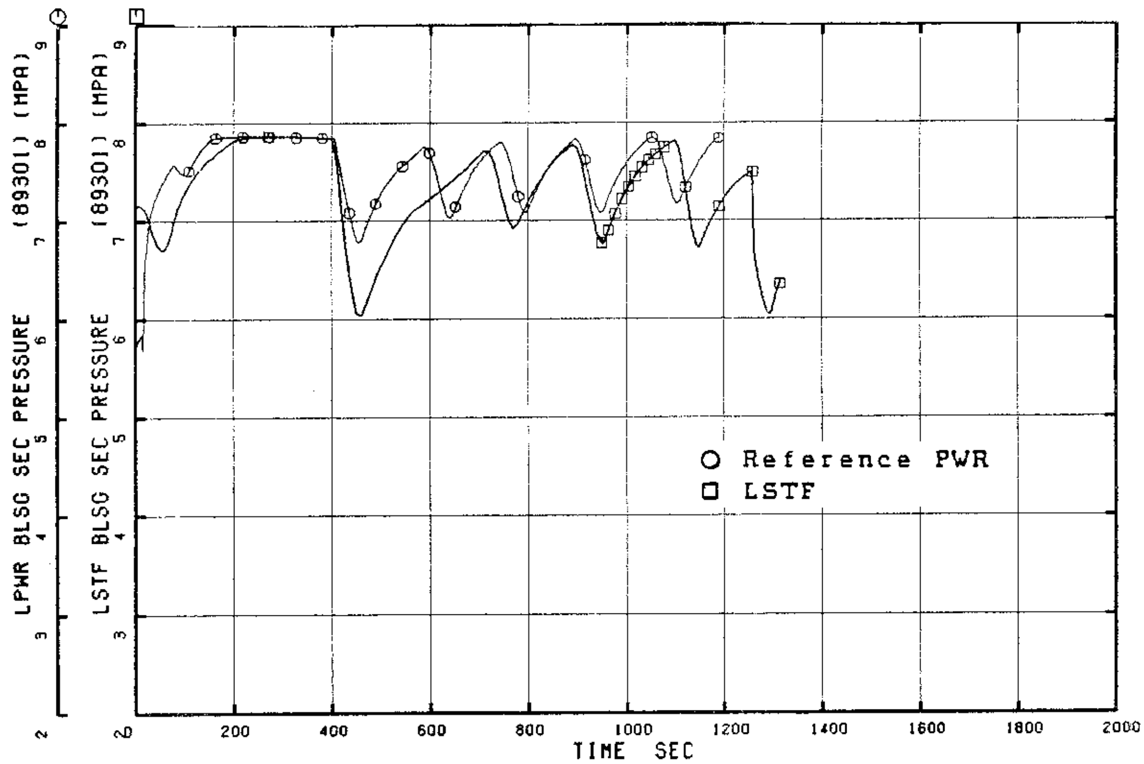


Figure 5.24 Comparison of the LSTF and reference PWR secondary pressure: calculation 5

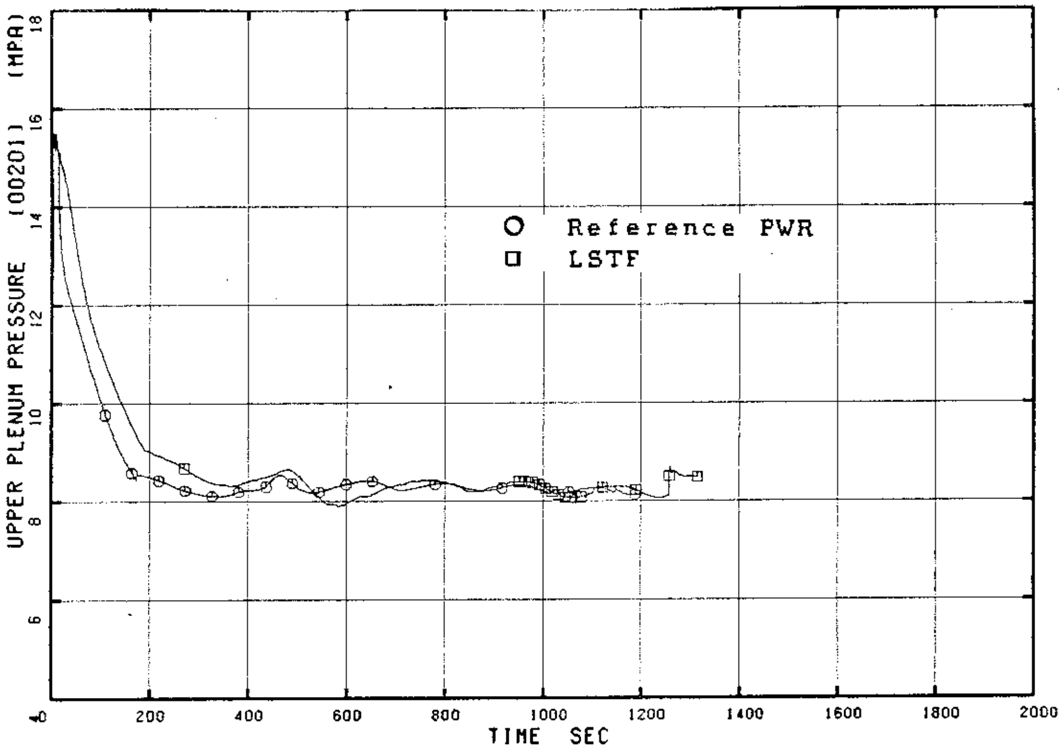


Figure 5.25 Comparison of the LSTF and reference PWR upper plenum pressure: calculation 5

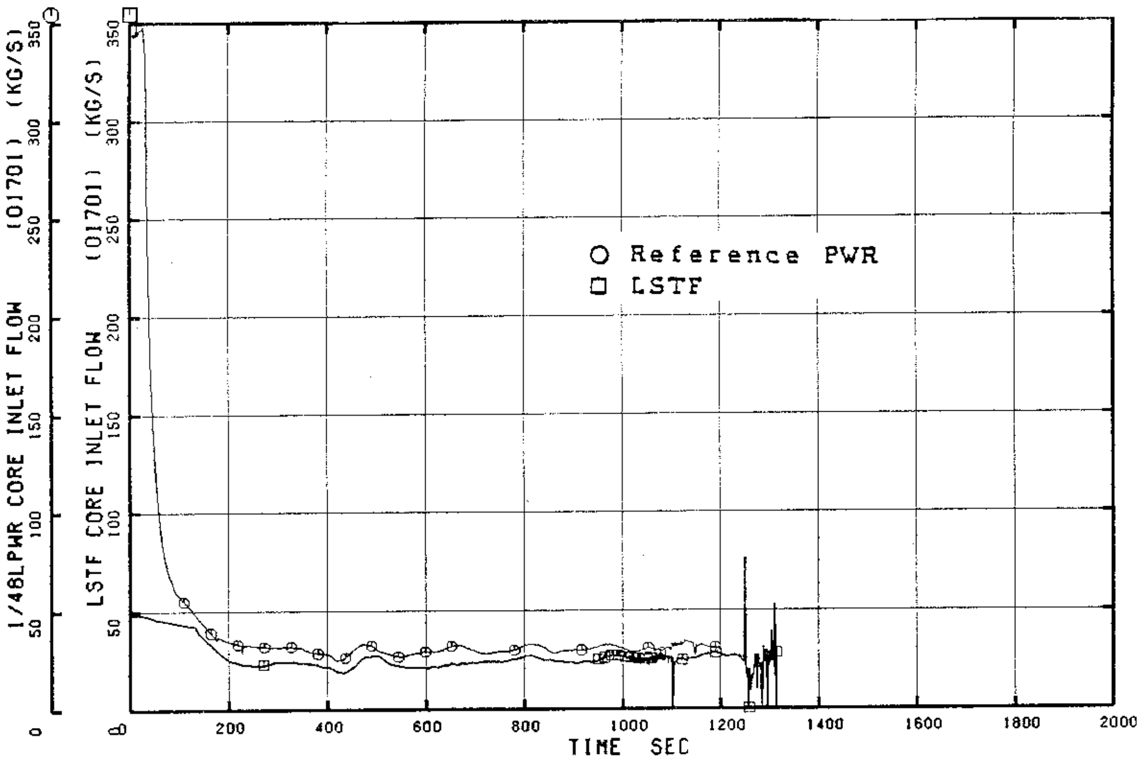


Figure 5.26 Comparison of the LSTF and reference PWR core inlet mass flow: calculation 5

5.3.2.2 LSTF

The LSTF showed the same trends as the reference PWR calculation. The LSTF pressurizer level was at the 86 % mark at 370 s (see Fig 5.22). As the ECCSs were switched off at 400 s, the LSTF primary mass inventory also began to immediately decrease (see Fig 5.23). Also, the LSTF turbine bypass valves depressurized the secondary (see Fig 5.24) to compensate for the tendency of the primary system to repressurize (see Fig 5.25).

The LSTF core mass flow remained at a lower value than the reference PWR throughout the transient for the reasons presented in previous discussions. The core remained well cooled for the duration of the transient.

5.3.3 Rated flow PORV transient with one SI pump: Calculation 6

The final calculations (No 6) are analogous to calculation 3. One SI pump was assumed available for the transient calculation.

5.3.3.1 Reference PWR

The primary system pressure (see Fig 5.27) decreased during the early portion of the transient. The decrease in primary system mass inventory (see Fig 5.28) was arrested by ECCS initiation at 82 s. The primary pressure, stabilized at 8.4 MPa (at 170 s) by adequate primary to secondary heat transfer slowly began to increase as the ECCS injection rate slightly exceeded the PORV break flow. As such, the pressurizer slowly began to fill (see Fig 5.29) similar to the calculation 4 transient. Positive core mass flow was maintained throughout the calculation (see Fig 5.30). The core remained well cooled.

5.3.3.2 LSTF

The trends exhibited by the LSTF calculation 6 transient match those of the reference PWR. The primary pressure (see Fig 5.27) decreased to the SI pump shutoff head (10.7 MPa) at 115 s. Thus, the primary system mass inventory (see Fig 5.28) began to increase.

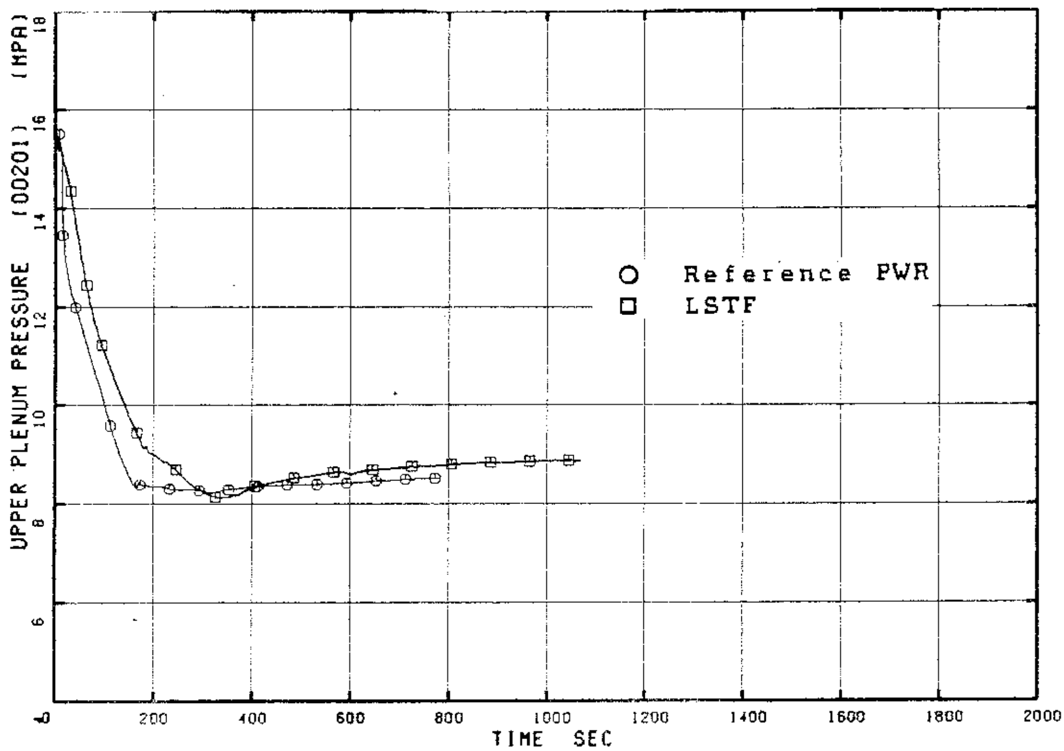
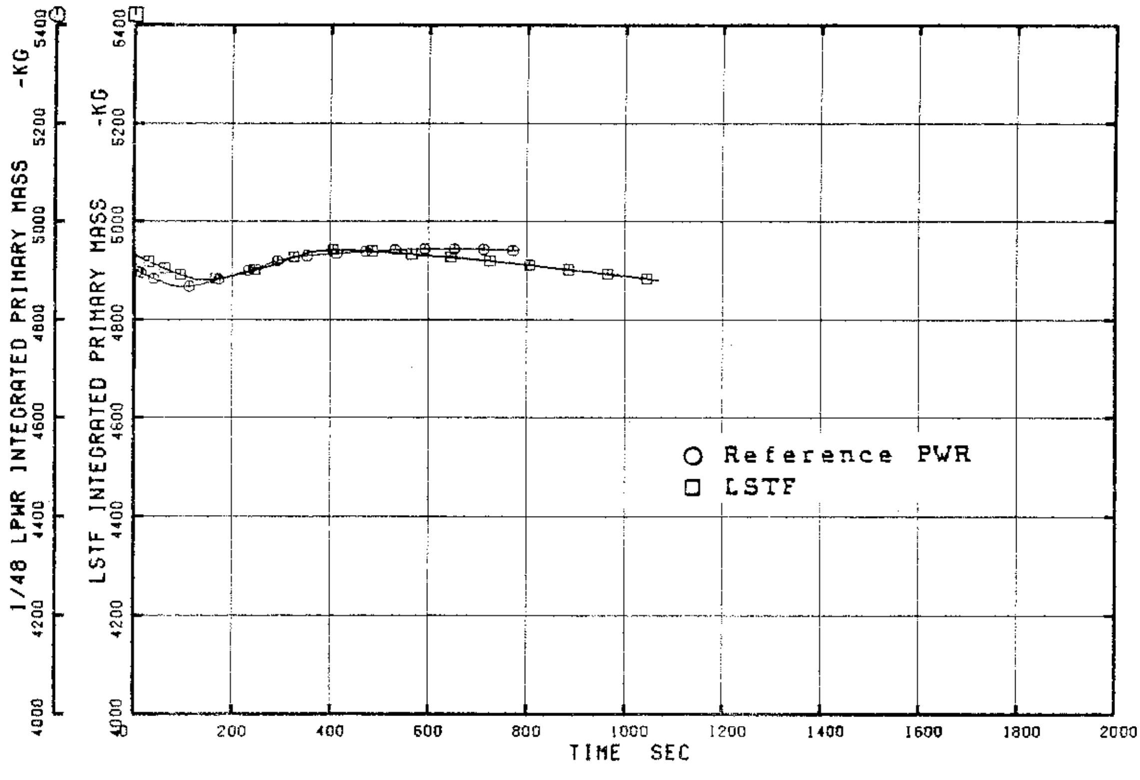


Figure 5.27 Comparison of the LSTF and reference PWR upper plenum pressure: calculation 6



5.28 Comparison of the LSTF and reference PWR primary mass: calculation 6

Although the total primary system mass decreased slowly between 400 and 1070s, the net decrease was only 0.8%. During this time period, steam collected in the upper head such that volumes 1, 23, 24, and 26 were nearly filled with steam by 1000s. However, the quantity of liquid mass in the remainder of the system slowly increased. For example, the quantity of liquid mass increased over 3.6% in the core from 400 to 1000s. Thus, the single SI pump mass injection was sufficient to maintain the primary system mass inventory.

Also, the primary pressure began to slowly increase as the system slowly filled (see Fig5.29). The LSTF core mass flow (see Fig 5.30), always less than the reference PWR, slowly decreased, but provided adequate core cooling for the duration of the calculation.

5.4 Synopsis of the Parametric Calculations

The six calculation groups were discussed in detail in the previous subsections. In summary, those transients consisting of a rated area PORV, or no ECCS flow or both (calculations 1,2,3 and 5) were calculated to remain at a primary system pressure slightly greater than the secondary for both the reference PWR (see Fig 5.31) and the LSTF (see Fig 5.32). The transients calculated assuming a rated flow PORV together with at least one SI pump (calculations 4 and 6) tended to repressurize since the ECCS capacity exceeded the PORV mass loss rate.

The differences between the calculations are also clear by comparison of the respective primary mass inventories. The primary mass inventories for the reference PWR calculations are shown in Fig 5.33. Calculations 1,2 and 3 predicted the primary system mass to decrease as the transient proceeded, since the rated ECCS flow of even two SI pumps was not sufficient to replace the PORV mass flow. The same trends are shown for the LSTF in Fig 5.34. Calculations 4,5 and 6 demonstrated that even one SI pump was sufficient to maintain the primary system mass inventory with a rated flow PORV in either the reference PWR or the LSTF.

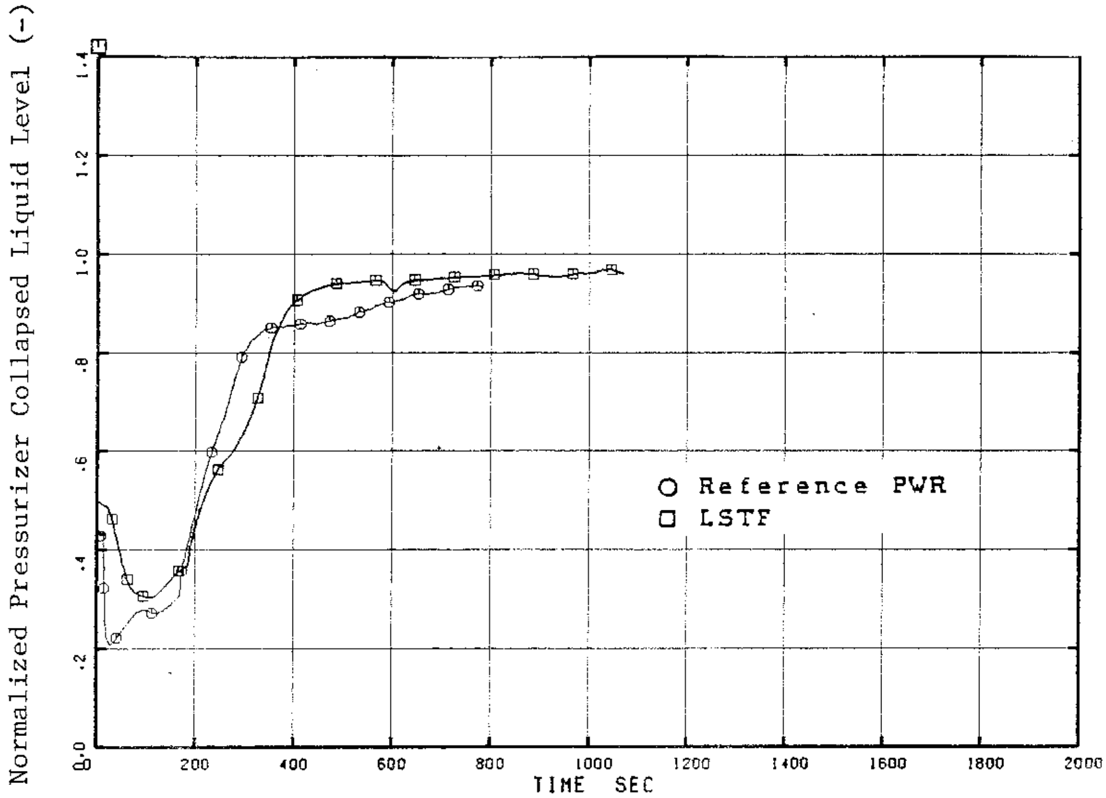


Figure 5.29 Comparison of the LSTF and reference PWR pressurizer collapsed liquid level: calculation 6

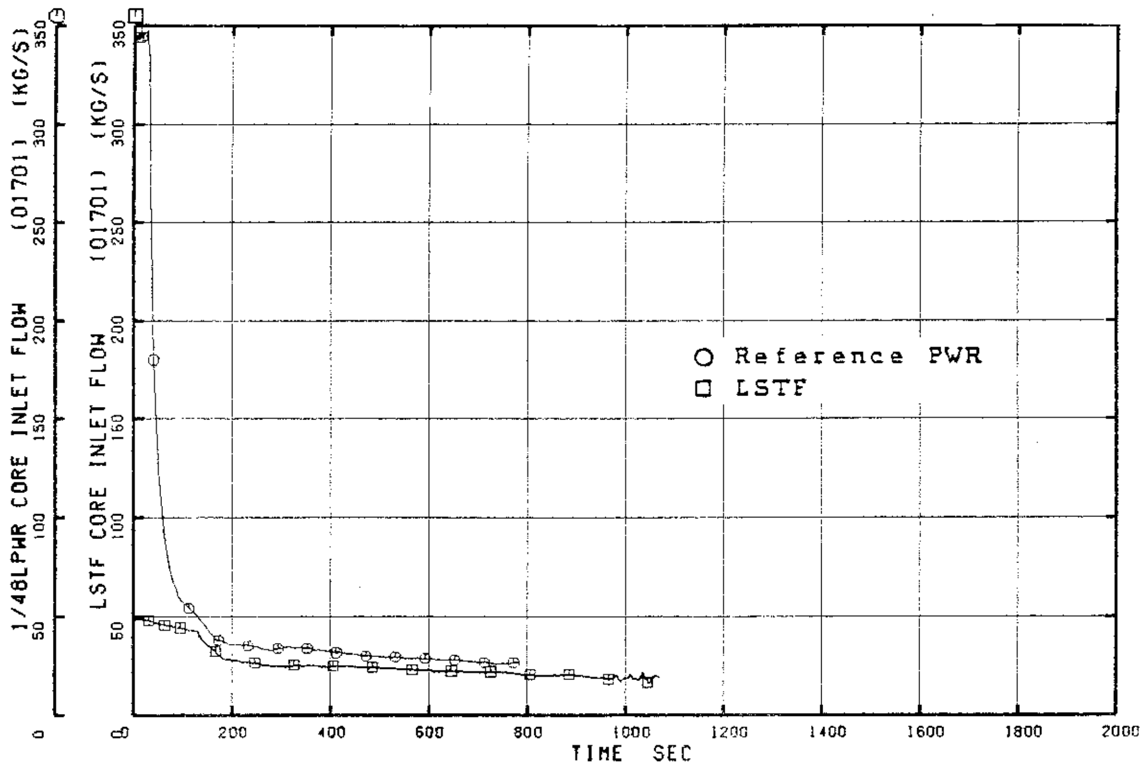


Figure 5.30 Comparison of the LSTF and reference PWR core inlet mass flow: calculation 6

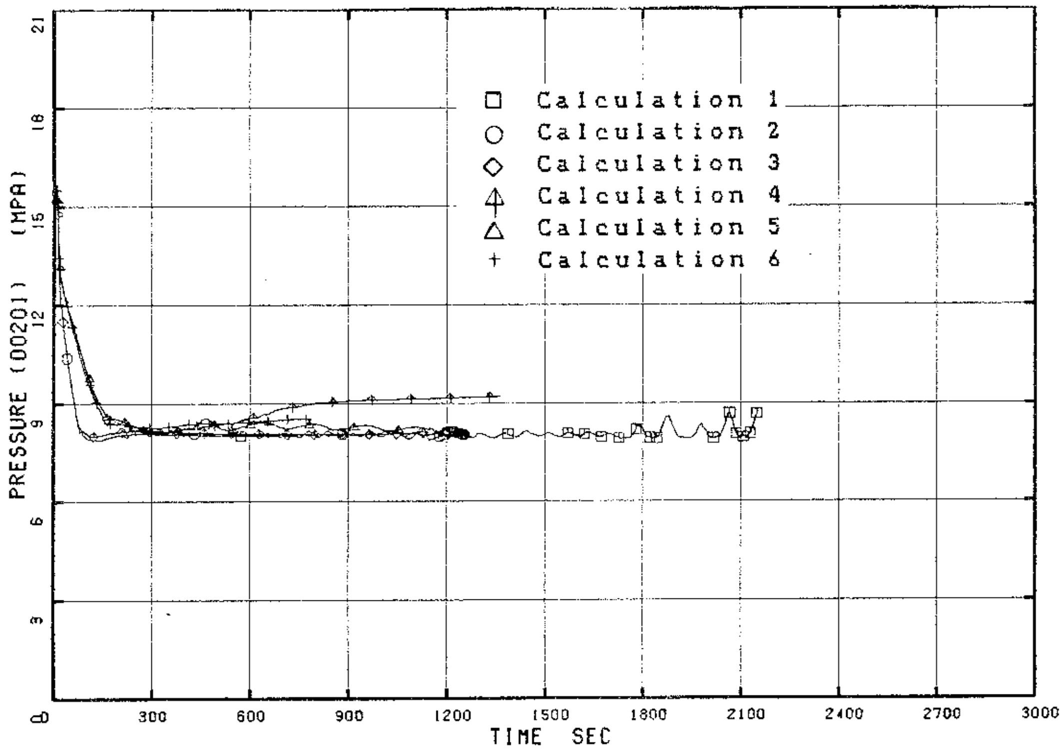


Figure 5.31 Comparison of the reference PWR upper plenum pressure: calculations 1 through 6

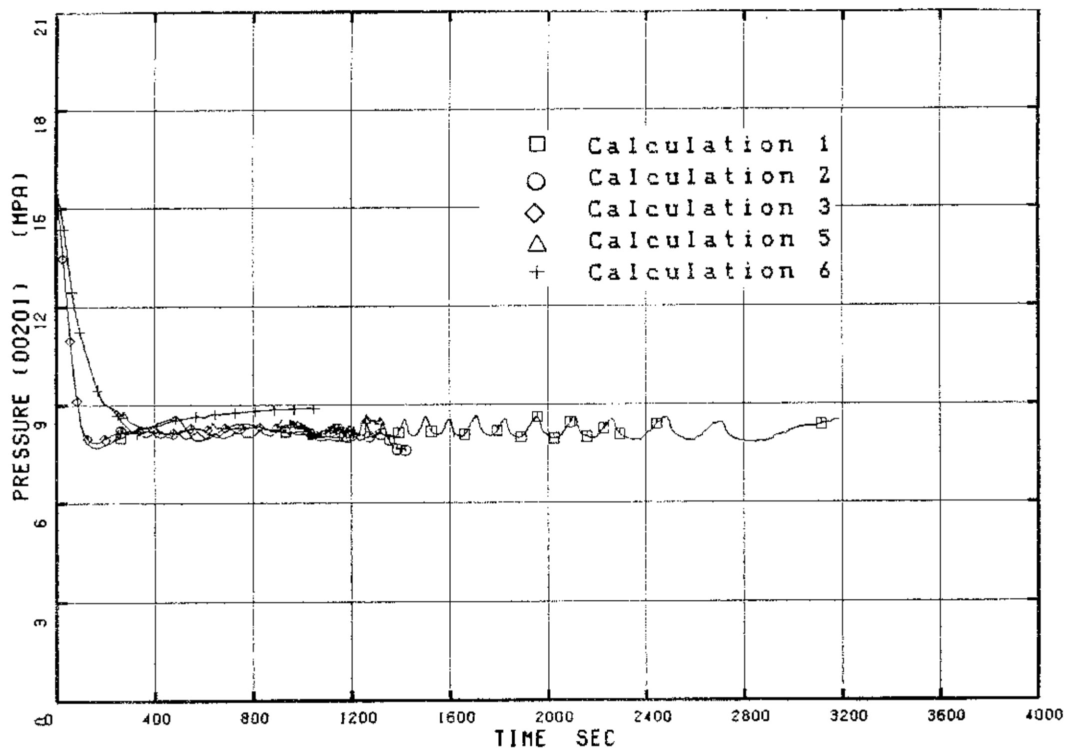


Figure 5.32 Comparison of the LSTF upper plenum pressure: calculations 1, 2, 3, 5 and 6

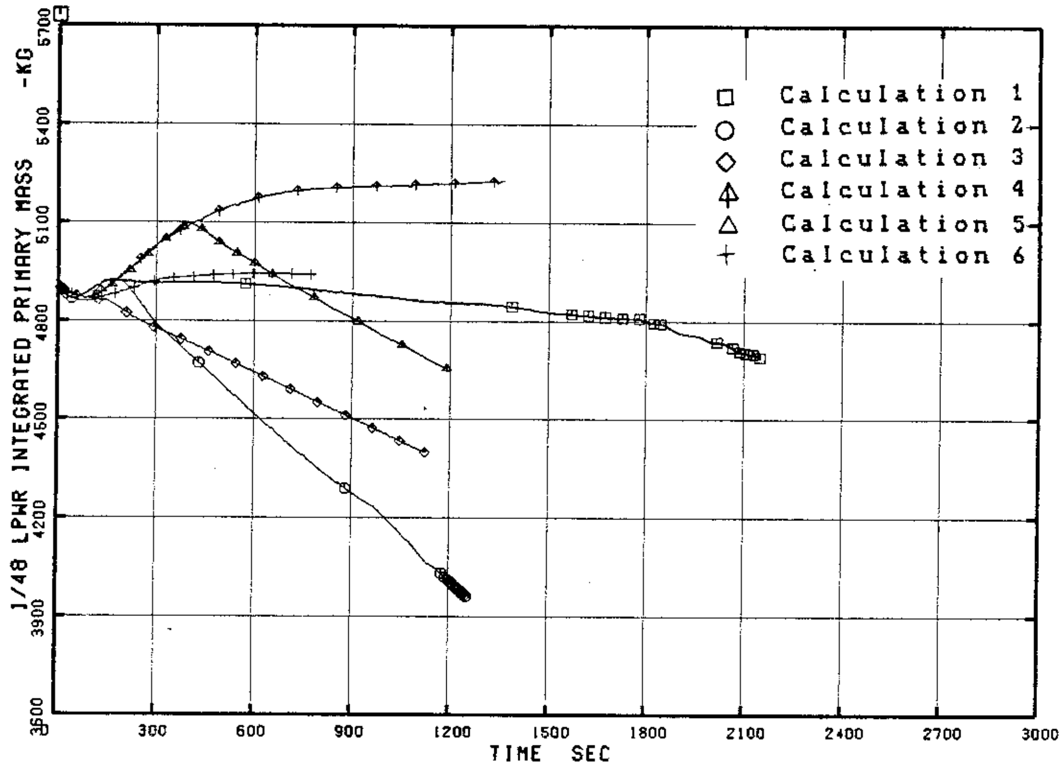


Figure 5.33 Comparison of the reference PWR primary mass: calculations 1 through 6

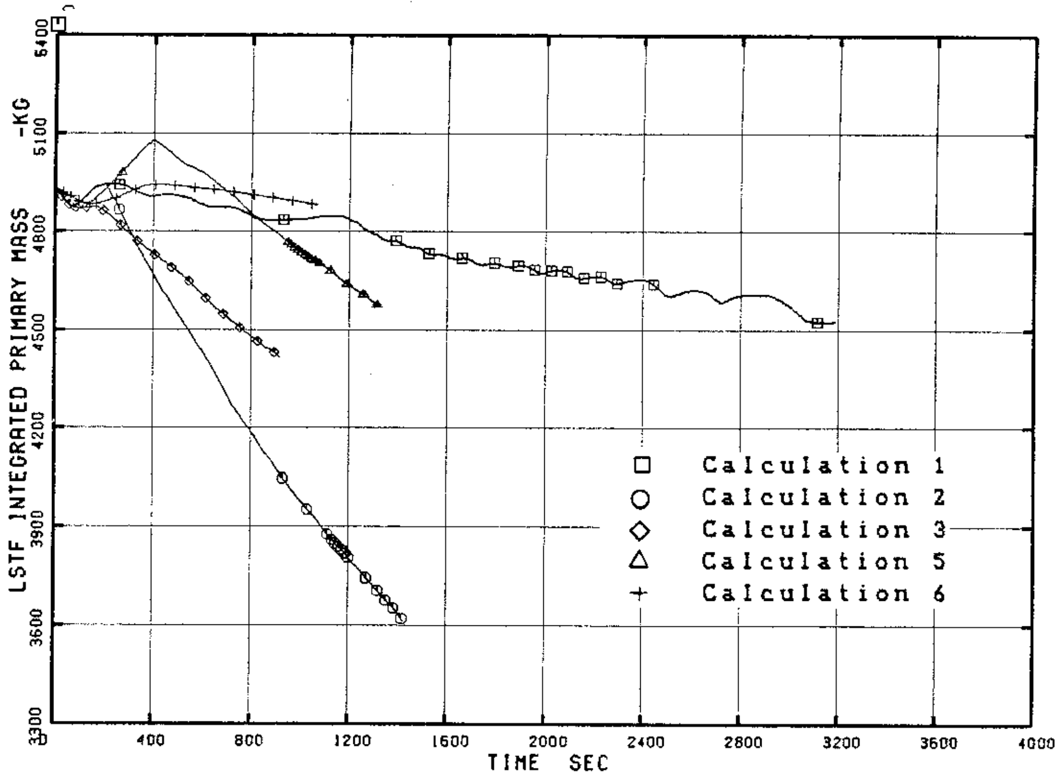


Figure 5.34 Comparison of the LSTF primary mass: calculations 1, 2, 3, 5 and 6

6. CONCLUSIONS AND OBSERVATIONS

Conclusions and observations can be listed relating to three distinct areas: (1) the TMI-2 scenario sequence matrix, (2) the LSTF hardware and test procedures and (3) the future direction of the RELAP5 LSTF and reference PWR modeling effort. Each of these areas will be discussed separately in the following paragraphs.

6.1 The TMI-2 Scenario Sequence Matrix

Discussion presented in Section 2 centered on the dissimilar nature of a B & W TMI-2 plant and a W type four loop plant. In particular, the large differences in the SG geometry between the two plants alone will cause different fundamental responses and event times. Thus, several observations and conclusions are offerred:

1. The general characteristics of the scenario experienced at the TMI-2 plant can be summarized as listed in Table 1.1.
2. An attempt to simulate the exact timing and behavior of the TMI-2 plant using the LSTF will produce an unrepresentative test sequence for a W type four loop plant (see Table 2.1).
3. Considering the transients investigated, the basic characteristics of a TMI-2 scenario in a W type four loop plant can best be simulated by using sequence 5 (see Table 3.3) i.e., a loss of feedwater in conjunction with an open PORV (rated flow)*. Test boundary conditions would include the presence of only the SI systems until the pressurizer mixture level reached an agreed elevation. Thereafter, the SI system would also be shut off until core temperatures reached a specified level.
4. Variations of sequence 5 , e.g., ECCS injection timing should be included in the test matrix, but should be included only after further analysis is complete.
5. The calculations shown in Section 4.1 have demon-

* Sequence 5 is thought to be the best candidate since four of the five TMI-2 scenario characteristics were simulated, i.e. items 1, 2, 3 and 5 - Table 1.1, and the scaled PORV flow rate is representative of the reference PWR.

strated that the pressurizer PORVs will not be challenged unless special reactor scram trip failures occur i.e., the low and low-low SG secondary water level trips. Thus, a primary ingredient of a W four loop plant TMI-2 scenario i.e., a stuck open PORV is best simulated by assuming the valve receives a spurious signal to open.

6. Comparison of calculations 1,2 and 3 with 4,5 and 6 demonstrate that the LSTF (and reference PWR) SI systems have sufficient capacity to replace the primary inventory lost through a rated flow PORV.
7. The TMI-2 sequence calculations discussed herein have demonstrated that the LSTF has the same qualitative thermal-hydraulic response as a W type four loop plant for such transients.

6.2 The LSTF Hardware and Test Procedures

The RELAP5 calculations conducted to date combined with an analysis of the LSTF loop resistance call attention to two items of concern:

1. As discussed in Section 3.1.3, primary differences between the LSTF and the reference PWR RELAP5 models were identified in the loop resistance contributions given by the grid spacer plate and the pumps. The calculated resistance values have not been confirmed by data to date. However, the calculations serve to identify the grid spacer plate (together with the lower plenum pressure loss) and the LSTF pumps as potential sources of difference. Thus, the grid spacer, lower plenum and pump resistances should be examined carefully as the facility design progresses.
2. The loop to loop oscillations observed in the RELAP5 analyses of the LSTF (see Fig 4.13) are thought to be a real phenomenon characteristic of the symmetrical two loop construction of the plant simulation. Although the calculations presented thus far may not give the correct primary loop flow frequency and magnitude, early test data must be carefully monitored to detect the extent of such behavior in the LSTF.

6.3 Future Direction of the RELAP5 Modeling Effort

Review of the LSTF and reference PWR models has resulted in several observations concerning future model changes:

1. The elevation differences between the various locations in the LSTF model should be examined carefully and revised if necessary.
2. The LSTF core inlet area should be revised to conform to the as-built geometry.
3. The LSTF RCP as built performance characteristics should be included in the model.

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