EGG-NTAP-5993

August 1982

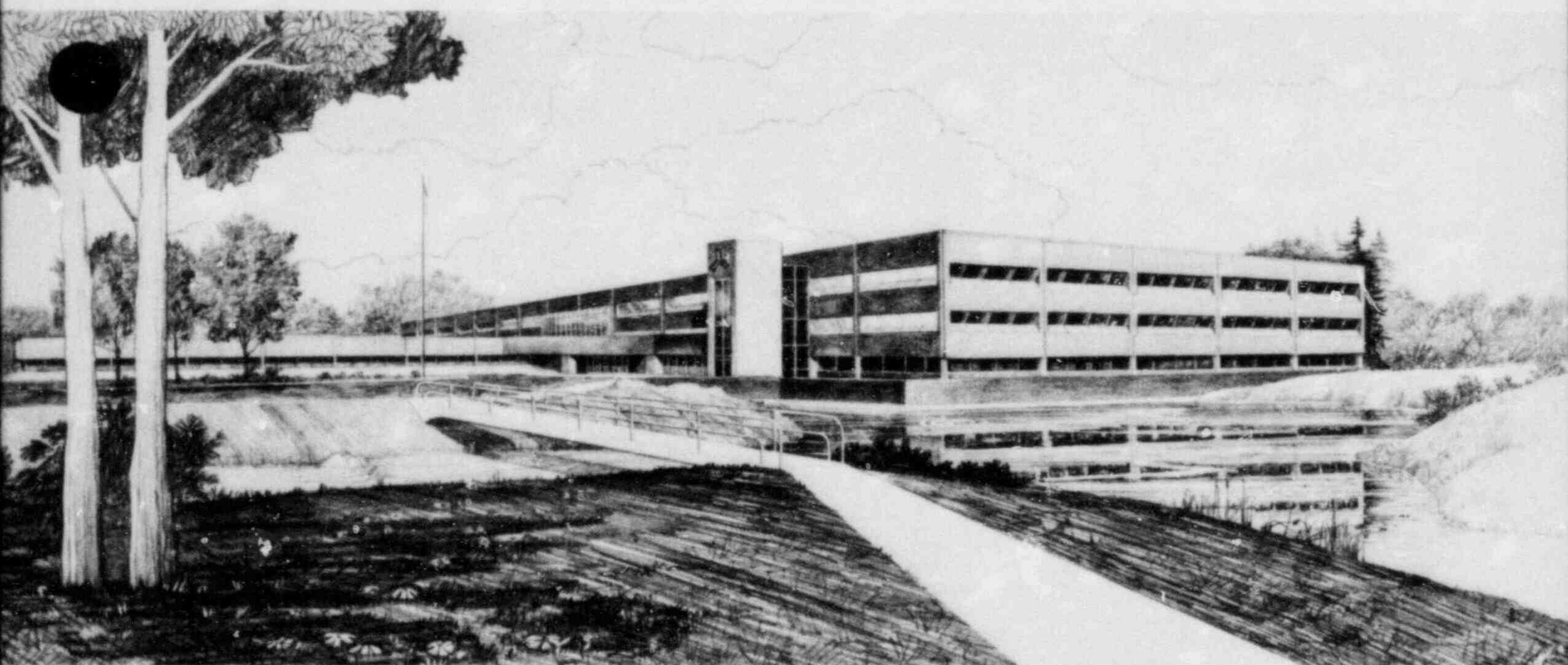
NRC Research and/or Technical Assistance Rept

PDR
LPDR
NSIC
NTIS
CF

A RELAP5 ANALYSIS OF A BREAK IN THE SCRAM DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

W. C. Jouse
R. R. Schultz

Idahc National Engineering Laboratory
Operated by the U.S. Department of Energy



This is an informal report intended for use as a preliminary or working document

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6354

EG&G Idaho

EG&G Idaho, Inc.

FORM EG&G-398
(Rev. 03-82)

INTERIM REPORT

Accession No. ______

Report No. EGG-NTAP-5993

Contract Program or Project Title: NRC Technical Assistance Program Division

Subject of this Document: A RELAP5 Analysis of a Break in the Scram Discharge Volume at the Browns Ferry Unit One Plant

Type of Document: Technical Report

Author(s): W. C. Jouse, R. R. Schultz

Date of Document: August 1982

Responsible NRC Individual and NRC Office or Division: R. Curtis, NRC

> This document was prepared primarily for preliminary or internal use. It has not received full review and approval. Since there may be substantive changes, this document should not be considered final.

EG&G Idaho, Inc.
Idaho Falls, Idaho **83415**

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C.
Under DOE Contract No. **DE-AC07-76ID01570**
NRC FIN No. A6354

INTERIM REPORT

EGG-NTAP-5993

A RELAP5 ANALYSIS OF A BREAK IN THE SCRAM DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

W. C. Jouse and R. R. Schultz

Published August 1982

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570

ABSTRACT

As part of the Severe Accident Sequence Analysis Program, a RELAP5/MOD1 simulation of a scram discharge volume piping break at Unit 1 of the Browns Ferry Nuclear Plant has been performed. The analysis shows eventual core uncovery can occur assuming that the operator takes no action to mitigate the consequences of this small break loss of coolant accident. Included also is a comparison of results to similar calculations performed at the Oak Ridge National Laboratory.

SUMMARY

As part of the Severe Accident Sequence Analysis (SASA) Program, a scram discharge volume small break loss of coolant accident (SBLOCA) of Unit 1 of the Browns Ferry Nuclear Plant is analyzed herein. The objective of this analysis is to determine the mechanism of core uncovery assuming no operator intervention. This Idaho National Engineering Laboratory (INEL) task parallels a similar effort of the Oak Ridge National Laboratory (ORNL).

The scram discharge volume (SDV) is located in the reactor building outside of primary containment. A non-isolable SDV break is assumed to occur shortly after a reactor scram. Vessel inventory is maintained by the high pressure coolant injection (HPCI) system. As the fission decay power decreases, the HPCI depressurizes the vessel below the shut-off head of the condensate/condensate booster pumps (CBPs). The CBPs flood the vessel and deactivate the HPCI turbine.

Subsequent to vessel flooding, reactor vessel pressure remains above the shut-off head of the low pressure coolant injection and core spray systems, precluding their operation. Without means of automatic inventory replenishment, depletion can only result in core uncovery if no operator action occurs. These results agree with the ONRL results.

CONTENTS

FIGURES

TABLES

A RELAP5 ANALYSIS OF A BREAK IN THE SCRAM DISCHARGE VOLUME AT THE BROWNS FERRY UNIT ONE PLANT

1. INTRODUCTION

The basis for the investigation discussed herein is rooted in the June 13, 1980 Browns Ferry Unit 3 Boiling Water Reactor partial failure to scram incident. Subsequent studies,[1,2] conducted by the Office for Analysis and Evaluation of Operational Data (AEOD) of the Nuclear Regulatory Commission (NRC), identified a break in scram discharge volume (SDV) as a potential problem since in some instances the break cannot be isolated.

Further investigation[3] of the postulated SDV break conducted at the Oak Ridge National Laboratory (ORNL) determined that the worst SDV break scenario occurred if it is assumed that the operator did not intervene and the plant protection systems were allowed to trip automatically.

To provide an independent SDV break calculation, the worst case sequence was simulated by SASA personnel at the Idaho National Engineering Laboratory (INEL). The simplified one recirculation loop Browns Ferry model developed for the station blackout studies[4] was used to conduct the RELAP5[5] SDV break analysis assuming no-operator action.

The report provides a summary discussion of the SDV break in five sections which follow. Section 2 describes the problem, Section 3 discusses the code, the model and the initial boundary conditions. Section 4 summarizes the results. A comparison between the INEL/ORNL calculations is made in Section 5. Section 6 itemizes the conclusions and observations of the SDV break study.

2. PROBLEM DESCRIPTION

During a reactor scram the scram discharge volume (SDV) receives effluence from the control rod drives (CRDs). Normally unpressurized, the SDV becomes part of the reactor coolant system pressure boundary upon a scram and until the scram is reset. Any leakage from the SDV or its associated piping accumulates in the basement of the reactor building. Since the SDV is located outside of primary containment, as shown in Figure 1,[1] accumulation of effluence in the basement can eventually flood the vessel water inventory equipment (VWIE) located there, rendering it inoperative. In particular, the high pressure coolant injection (HPCI), reactor core isolation cooling (RCIC), residual heat removal (RHR), and core spray (CS) system pumps are located in the basement of the reactor building and will fail if flooded.

A key assumption in this accident sequence is that the operator takes no corrective or mitigating action (ORNL has demonstrated that normal operator intervention will make the transient recoverable, see Reference 3). Thus, the scram is not reset, the SDV is not isolated from the reactor, the RCIC is not reset after its injection terminates and the safety relief valves are not manually used by the operator.

Briefly, this accident proceeds as follows: The initiating event is a spurious high main steam line radiation signal which causes the main steam line isolation valves (MSIV) to close and the reactor to scram. Thirty seconds later, the SDV break occurs and effluence begins flowing into the basement. The vessel water inventory is automatically maintained by the HPCI-RCIC systems until the reactor vessel pressure falls below the shut-off head of the condensate and condensate booster pumps (CBPs). CBP injection floods the vessel and the HPCI steam turbines, thus leaving the vessel without means of inventory replenishment. The CBPs fail when their source of water i.e., the condensate hotwell is depleted.

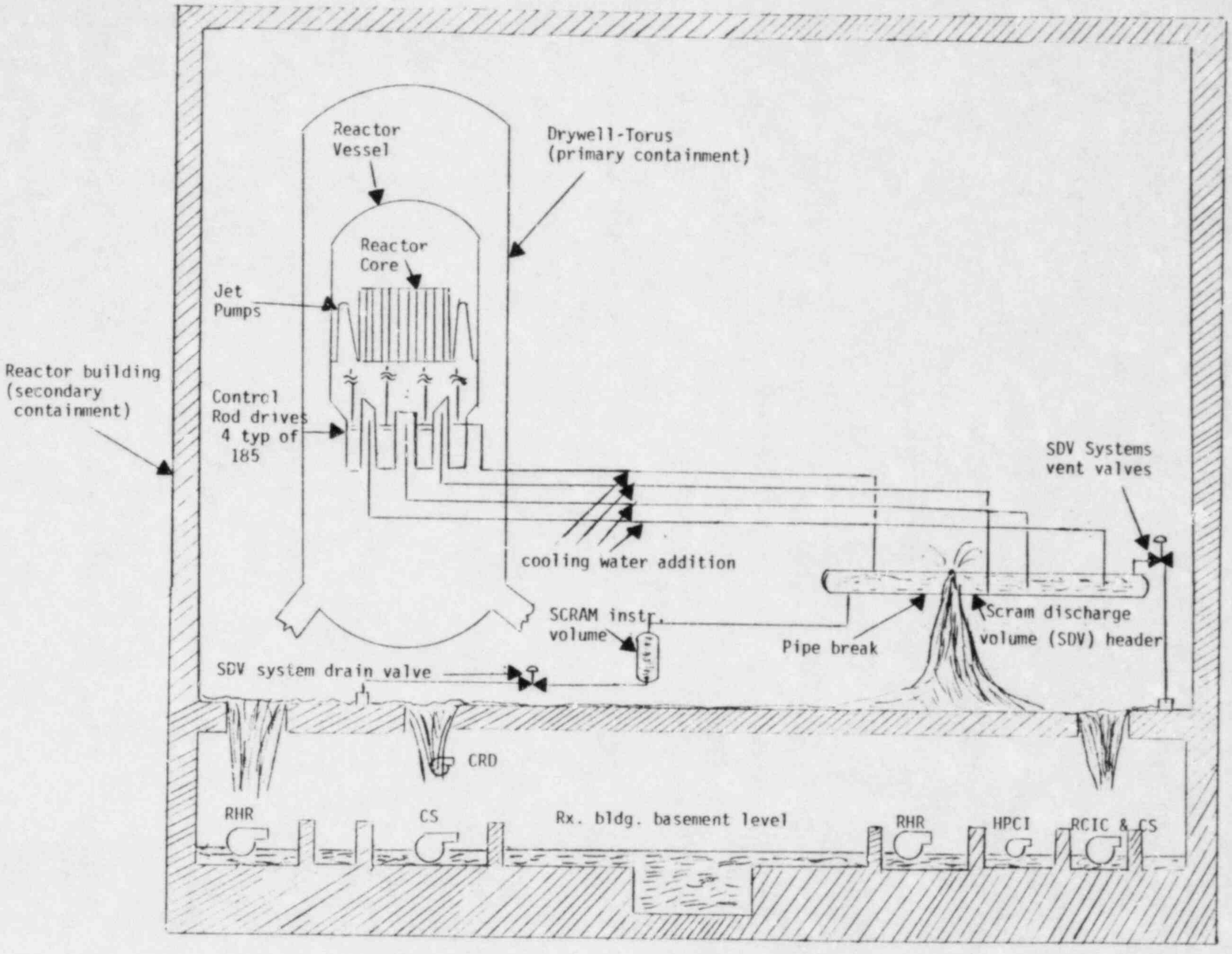


Figure 1. Reactor Building Elevation.

3. MODEL AND CODE

The overall response of the system depends upon coding, modeling, and application of boundary conditions. RELAP5[a] was used to perform the transient simulations. The model[4] used in the simulation was developed during the INEL station blackout studies. Figure 2 contains a nodalization diagram of the Browns Ferry Unit 1 model(Configuration Control No. F00930). Vessel conditions at time of reactor scram were 100 percent rated power (see Table 1).

Only minor modifications necessary to apply boundary conditions or alleviate computational difficulties were made on the Reference 4 model. The modifications implemented are listed below:

1. The break area (see junction 437 Figure 2) was sized to pass 550 gpm assuming upstream conditions of: 1050 psia, 525°F. The break occurred 30 s after the transient began (see Table 2). The break area was assumed constant for 90 min. Thereafter, the area was slowly increased over a 6.5 hr time span until 1800 gpm would leak from the vessel. Such behavior simulates the action of a line break (see Figure 1) with the break flow being limited by the CRD Graphitar seals. The slow increase assumed to begin at 90 min is physically caused by erosion of the seals. The maximum area obtained at 8 hr in the transient represents the complete erosion of the seals.

2. The level trip logic was improved from the Reference 4 model. The HPCI/RCIC systems were modeled to trip on when the downcomer water elevation was 476 in. and off at the 582 in. level. However, to prevent spurious elevation signals from either tripping the systems off or on incorrectly, the logic was modified to include a three point moving average computer;

---

a. RELAP5 MOD1 Cycle 13, Code Configuration Number F00341 with updates was used (see Reference 5)

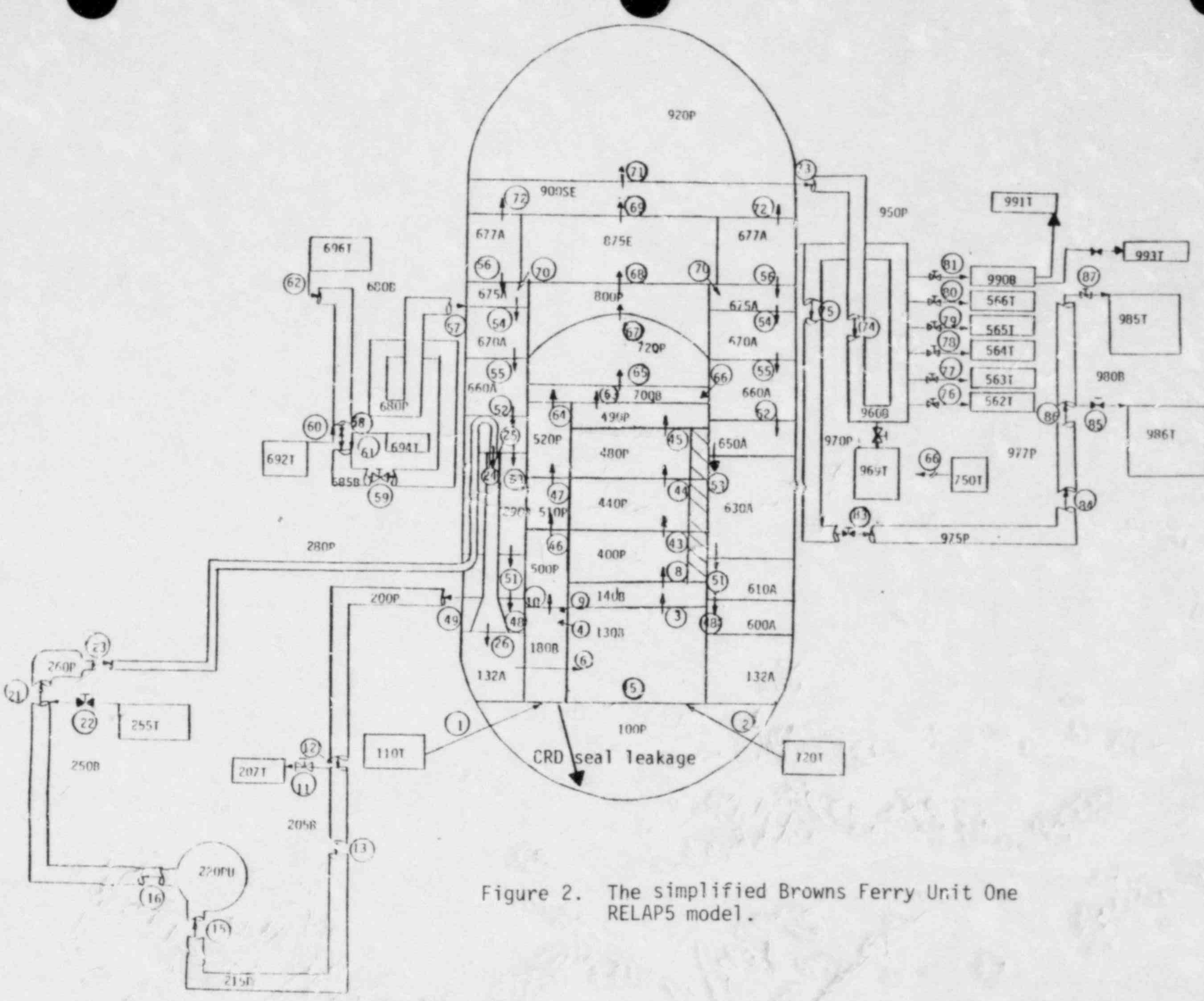


Figure 2. The simplified Browns Ferry Unit One RELAP5 model.

TABLE 1. REACTOR CONDITIONS AT TIME OF SCRAM

Condition	Value
Total Reactor Power	3,357.9 MW (t)
Main Steam Flow	13.63 x 10^6 lbm/hr
Feedwater Flow	13.46 x 10^6 lbm/hr
Downcomer Level	544.5 in
Steam Dome Pressure	1016.1 psia
Lower Plenum Pressure	1053.4 psia
Lower Plenum Temperature	526.5°F
System Total Inventory	7.295 x 10^5 lbm
Core Mass Flow Rate	94.3 x 10^6 lbm/hr

TABLE 2. CHRONOLOGY OF MAJOR EVENTS

Time (sec)	Event
0	Reactor trip on high main steam line radiation.
3	Main steam line isolation valves closed.
30	SDV piping break initiates.
450	HPCI and RCIC initiate on low vessel water level.
780	HPCI and RCIC trip off on high level. The RCIC is not reset.
450-12900	Level maintained between 476 in. and 582 in. by HPCI system.
12900-18660	CBPs initiate and rapidly fill the vessel, flooding the HPCI turbine. Level maintained until CBPs fail on loss of suction.
19300	Reactor vessel pressure begins to increase after reaching a minimum of 316 psia.
19300 +	Vessel pressure continues to rise and eventually actuates the safety relief valves. A boil-off ensues, reducing vessel inventory until the core uncovered.

$$EL = \frac{1}{3}\left[EL\ (n - 2) + EL\ (n - 1) + EL(n)\right]$$

where

EL = downcomer water level

n = current time step

with the trip occurring after a 10 s delay (see Reference 4 Appendix B for a description of EL)

3. The recirculation pumps were modeled to runback to 20 percent speed upon reactor scram and trip off when the downcomer water level reaches 476 in. (low-low level).

4. The total liquid delivered to the reactor building was tracked during the transient. The liquid delivered was calculated as the sum of the break flow and a constant 170 gpm control rod drive cooling flow which bypasses the reactor vessel. The VWIE located in the basement is assumed inoperative when 470548 gal of water have accumulated there.

4. RESULTS

The initiating event of this severe accident sequence is a high main steam line radiation signal, which results in the closure of the main steam line isolation valves (MSIV) and a corresponding reactor scram. Concurrently, the feedwater pumps begin a rapid run-out as shown in Figure 3, but fail upon loss of steam. The recirculation pumps run back to 20% speed in 50 sec. The motor-driven condensate and CBPs remain operational, but are unable to inject fluid into the vessel because of their relatively low shut-off head (415 psia).

MSIV closure causes the reactor vessel pressure to rise rapidly, and the safety/relief valve (SRV) setpoints are reached in 4 s.[a] Safety/relief valve cycling initiates and continues while both the net steaming rate (from the core decay heat transfer) and level remain high. At 450 sec, the first VWIE (HPCI-RCIC) injection initiates when the vessel level falls to the low-low trip (476 inches above vessel zero). Vessel depressurization follows as shown in Figure 4. Injection terminates at 780 s when the level rises to the high water level (582 inches above vessel zero) The vessel pressure begins to increase. The RCIC is automatically shut off after the first injection cycle and is not reset.

Shortly after scram, the core and core bypass flows (Volumes 500, 510, and 520) decrease from rated value (see Figure 5). The bypass flow reverses as the liquid in the upper plenum begins to flow downward through the bypass and then into the core inlet piece and upwards through the core. Such a natural recirculation pattern is generated by core decay heat transfer.

In the natural recirculation mode of core cooling, core and core bypass flows are driven by a gravity head difference. This head is balanced by frictional losses. Introduction of subcooled emergency core cooling (ECC) injection fluid upsets this balance, and causes the bypass

---

a. The first SRV setpoint is 1120 psia. The remainder are staged upward. All reseat 50 psi lower than their opening setpoints.

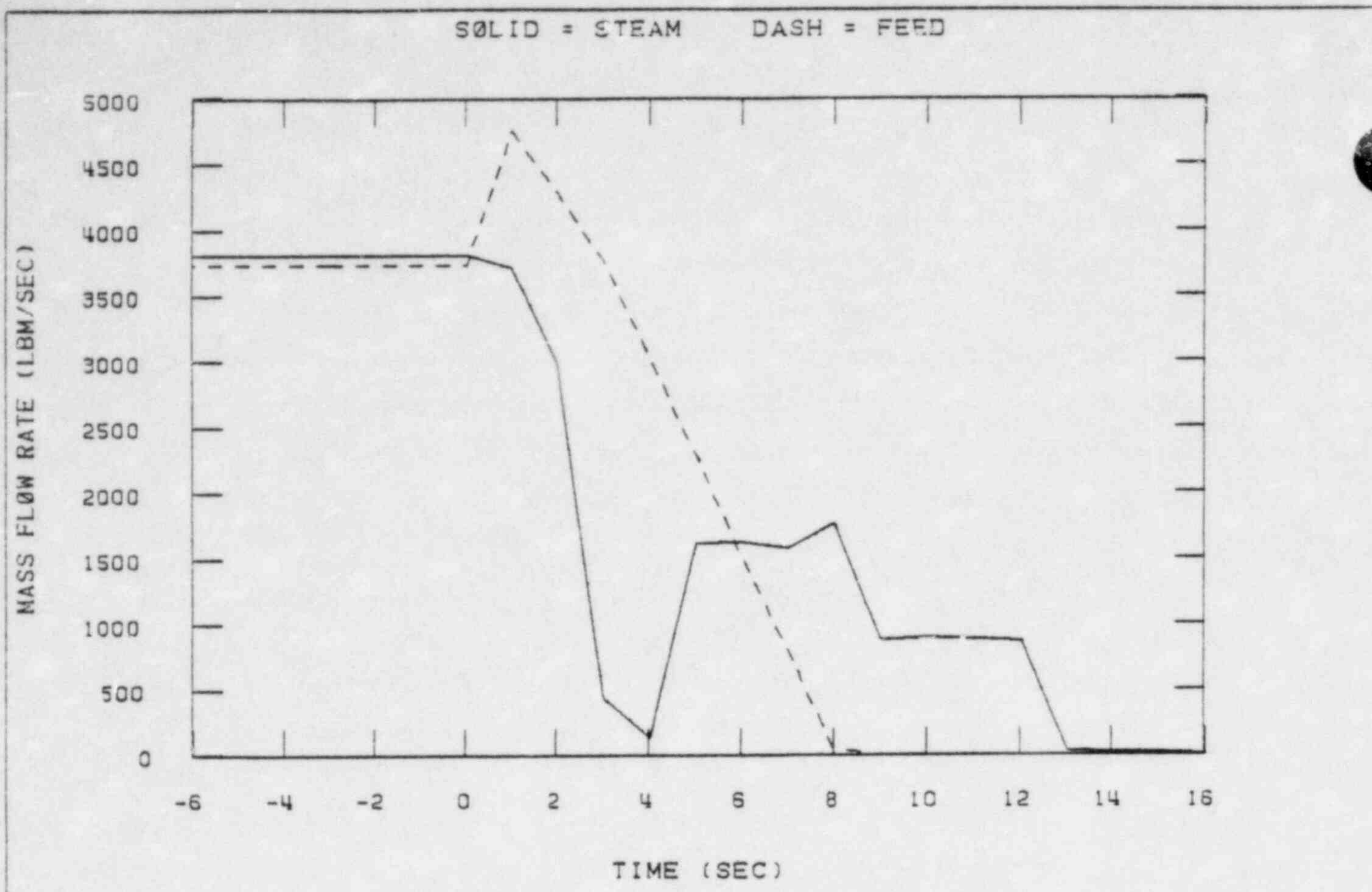


FIGURE 3: NEAR-SCRAM STEAM AND FEED

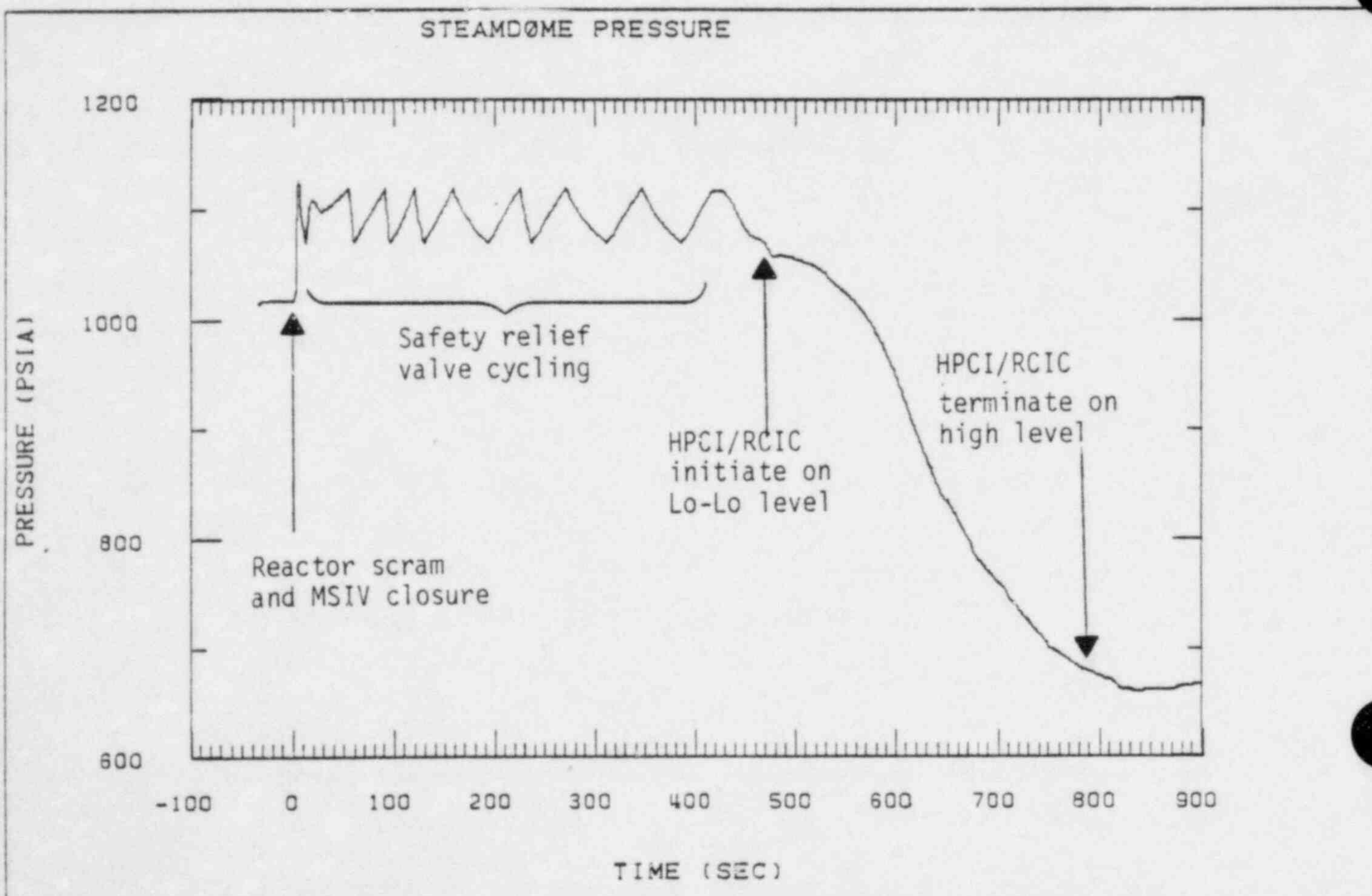


FIGURE 4: SHØRT TERM STEAMDØME PRESSURE

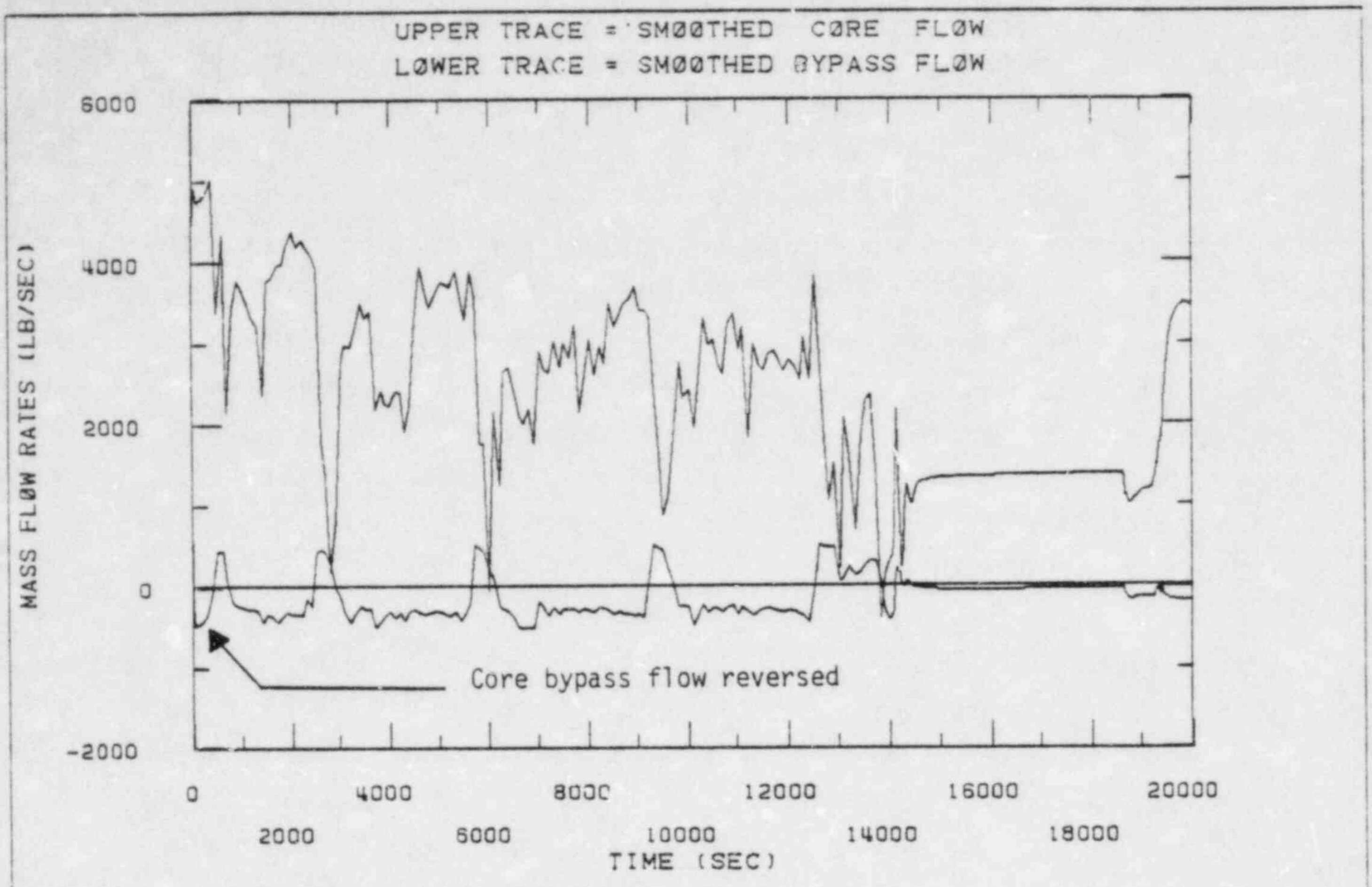


FIGURE 5; CØRE AND BYPASS FLØWS

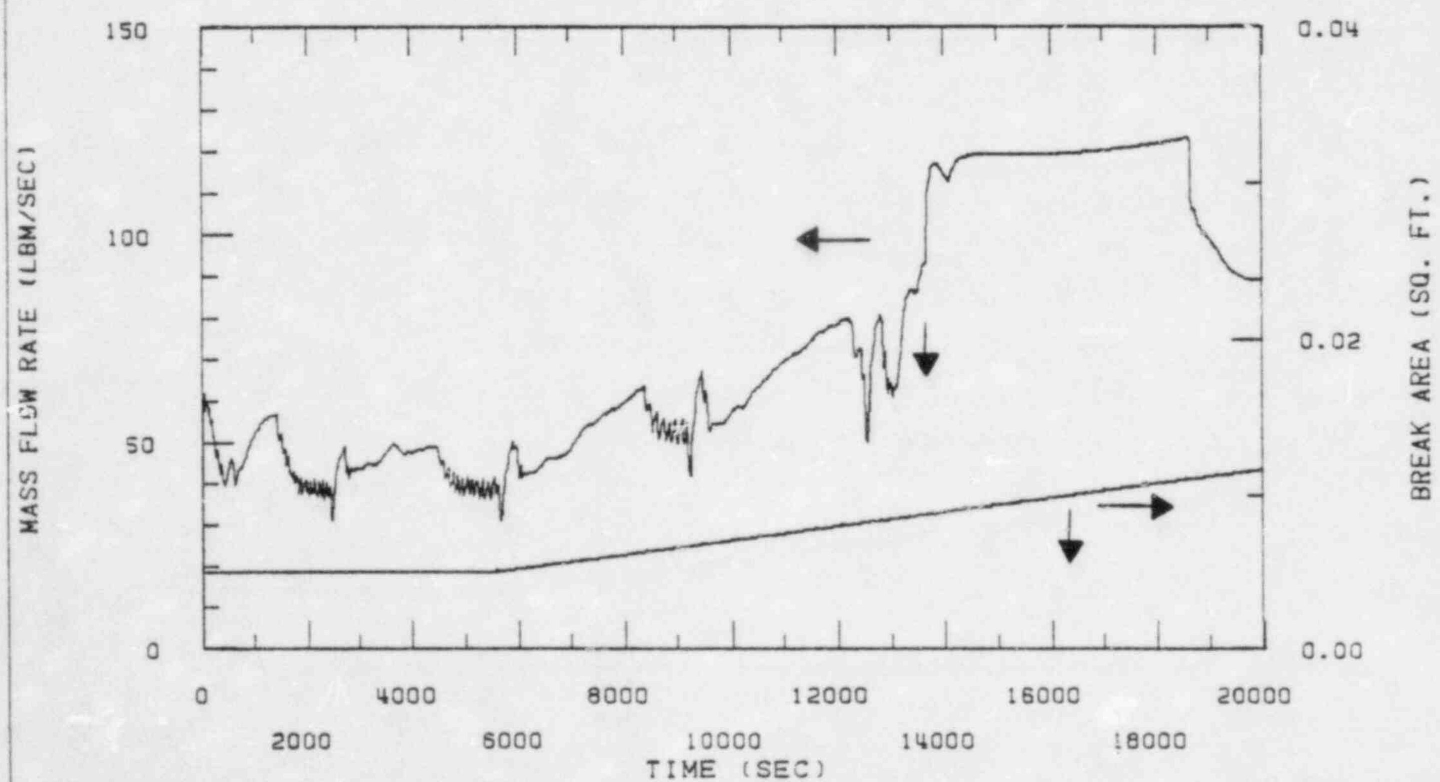


FIGURE 6; BREAK SEAL FLØW AND SIZE

flow to briefly revert to its normal direction. The magnitude of the core flow is reduced by the subcooling as the steaming rate falls, but picks up at the termination of an injection cycle.

The vessel pressure, the direction of the bypass flow, the size of the break area and the static quality of the fluid in the guide tube (Volume 180) all influence the break flow. Thus, the break flow (see Figure 6) decreases as the vessel pressure decreases, and decreases when the bypass flow is reversed since saturated fluid enters the break rather than subcooled fluid. These factors combine to give the break flow behavior shown in Figure 6. An overall increase in flow occurs after 5430 s as the break size increases.

The cycling VWIE systems (see Figure 7) take suction from the atmospheric temperature condensate storage tank. The flow of cool water into the downcomer during injection rapidly condenses steam in the dome and upper downcomer, subcooling and depressurizing the vessel (see Figure 8). In addition, core steaming is subsequently reduced and finally stopped as the fluid in the core shroud becomes subcooled. For the most part, vessel thermodynamic conditions correspond closely to saturation. The effect of localized downcomer subcooling on vessel pressure is such that core voiding tends to increase during the first few seconds of injection. Vessel pressure and temperature rise rapidly upon the termination of an injection cycle.

In total, the VWIE cycles five times (see Figure 7). The fifth and last HPCI cycle depressurizes the vessel to such an extent that the CBPs begin to flow into the vessel at t = 13000 s. A brief, high-flow period follows in which the vessel is nearly filled, as shown in Figure 7. The HPCI steam turbine is flooded and hence inoperative. The CBPs keep the vessel filled and the vessel pressure at their shut-off head until the condensate hotwell is depleted of liquid at 18570 s.

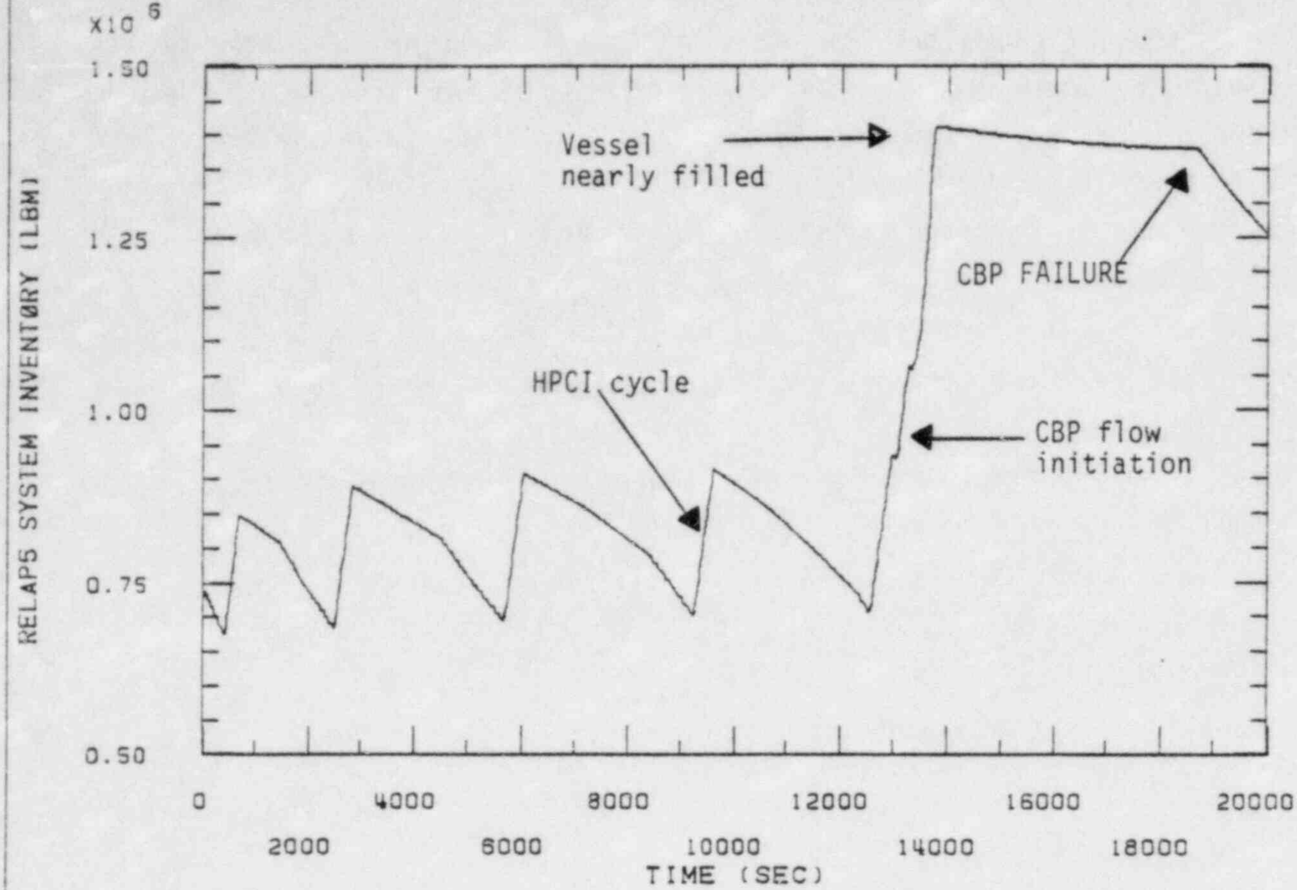


FIGURE 7; SYSTEM INVENTØRY

SØLID = RELAP5 DASH = NUREG CR/2672 (FIGURE 3.1)

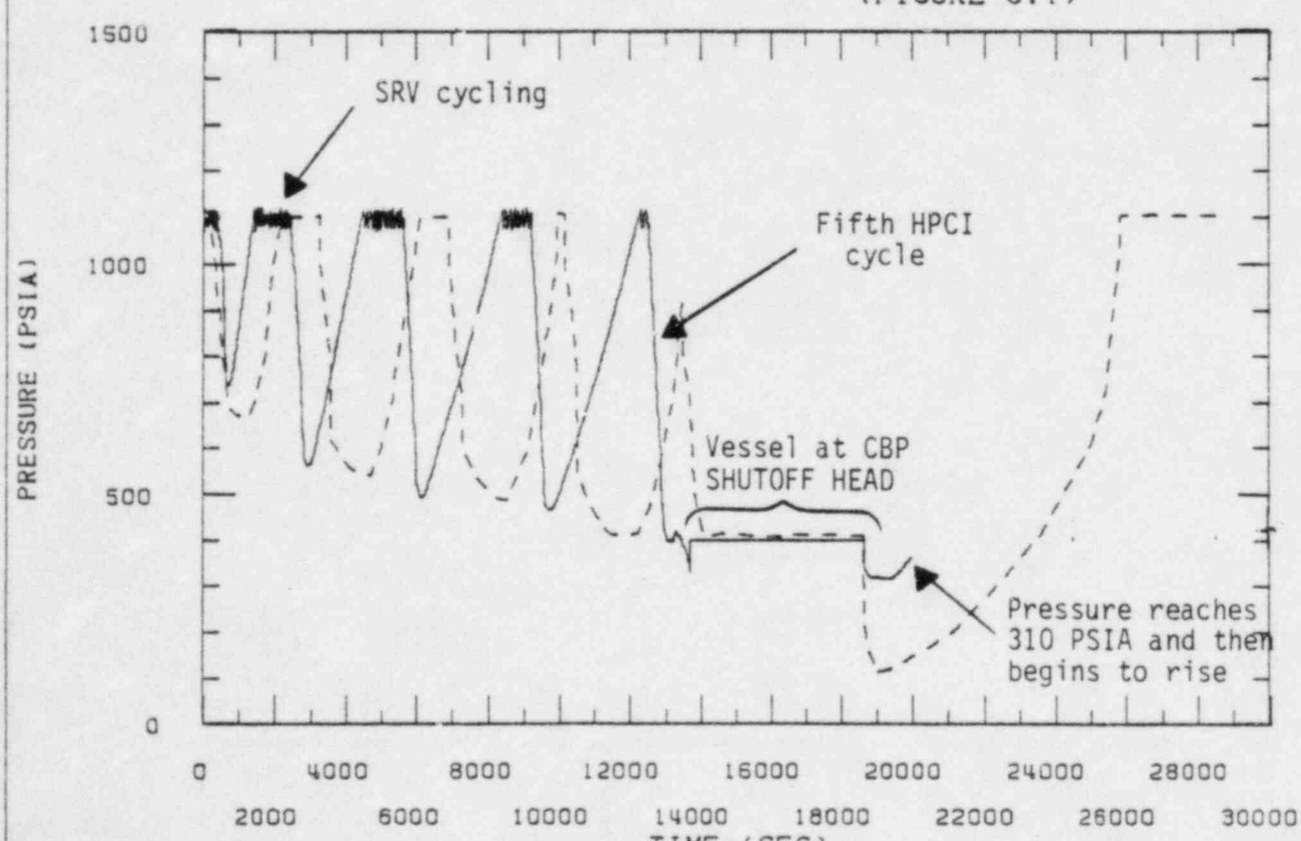


FIGURE 8: STEAM DØME PRESSURE CØMPARISØN

Thus, by 18570 s only the low pressure ECCs remain to provide water to the vessel. Following loss of the condensate/condensate booster pumps, the vessel pressure decreases to 316 psia, but begins to increase as the core decay heat thermally swells the vessel fluid. The low pressure systems are never able to begin injection as the vessel pressure always remains above their shut-off heads. The lack of vessel injection capability will eventually lead to core uncovery due to the continuing break inventory depletion rate. Table 2 contains a concise chronology of these results.

These RELAP5/MOD1 results are stored under Configuration Control No. F00944.

5. COMPARISON OF THE INEL AND ORNL CALCULATIONS

The RELAP5 model (INEL) used for the SDV break analysis contains a more detailed nodalization scheme than the BWR-LACP model[3]. Thus the thermo-hydraulic state of each volume (see Figure 2) is rigorously calculated in the RELAP5 model. Thermodynamic state distributions are available within a region e.g., volumes 600 through 677 represent the downcomer. As an example, the initial injection of subcooled ECC into the saturated two phase downcomer rapidly depressurizes the vessel without directly affecting the core steaming rate. As injection proceeds, core boiling is terminated as the core inventory becomes subcooled. However, core boiling begins shortly after ECC injection is terminated. Boiling occurs first at the top of the core and proceeds rapidly downward. As a result the vessel repressurizes. The BWR-LACP code treats the vessel regions on an average basis, thus the system response is dynamically slower. Such behavior is apparent if the steam dome pressure calculated by the two codes is compared (see Figure 8). The RELAP5 results show a faster time response in pressure e.g., the RELAP5 analysis predicts a faster repressurization following termination of VWIE injection. The overall vessel energy/mass balances agree quite well, however.

Figure 9 shows the seal leakage mass flow rate plotted against both BWR-LACP results and time. Although generally in good agreement, the RELAP5 results show a flow degradation due to safety/relief valve cycling induced voiding where the break takes suction, thus reducing the flow rate.

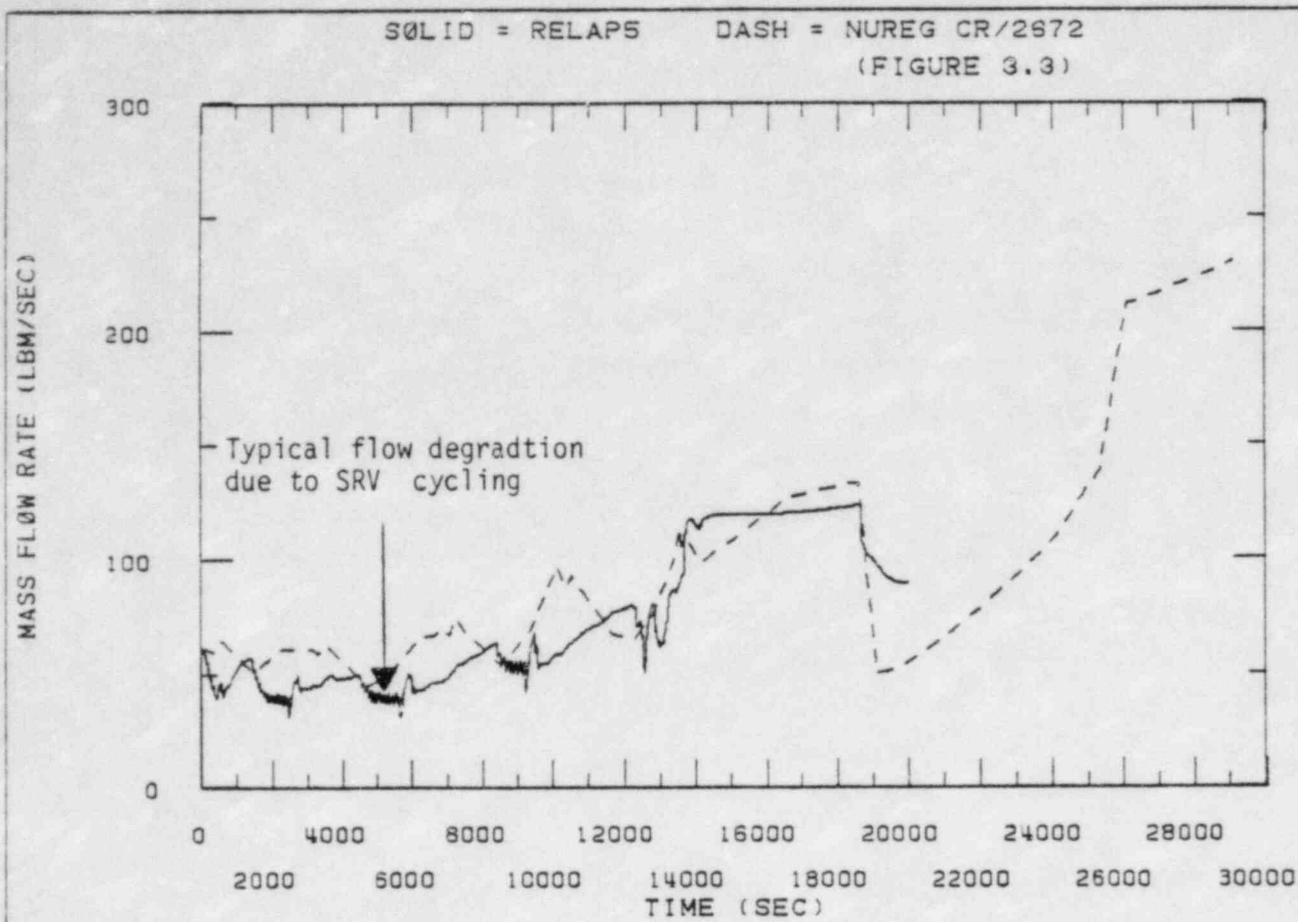


FIGURE 9: SEAL MASS FLØW RATE CØMPARISØN

6. CONCLUSIONS/OBSERVATIONS

Several conclusions and observations emerged in the SDV calculation conducted using the RELAP5 Browns Ferry model:

1. Core uncovery will occur during this transient if the operator does not act. The automatic systems are not sufficient to prevent the core from uncovery.

2. Once the CBPs have depleted the condensate hotwell, only operator action reducing the vessel pressure by opening one or more SRVs such that the low pressure ECCS could inject water into the vessel can prevent core uncovery.

3. The SDV break flow is influenced directly by the core bypass thermodynamic conditions.

4. Generally, RELAP5 and BWR-LACP code calculations show good agreement.

7. REFERENCES

1. Safety Concerns Associated with Pipe Breaks in the BWR Scram System, NUREG-0785, draft dated March 1981.

2. General Safety Evaluation Report Regarding Integrity of BWR Scram System Piping, NUREG-0803, August 1981.

3. Condon et al., SBLOCA Outside Containment at Browns's Ferry Unit One - Accident Sequence Analysis, NUREG/CR-2672, ORNL/TM-8119/VI, to be published.

4. R. R. Schultz and S. R. Wagnor, The Station Blackout Transient at the Brown's Ferry Unit One Plant - A Severe Accident Sequence Analysis, to be published.

5. Ransom et al., RELAP5/MOD1 Code Manual, NUREG/CR-1826, November 1980, Vol. 1 and 2.