NRC Research and/or Technical Assistance Rept

EGG-NTAP-6002

September 1982

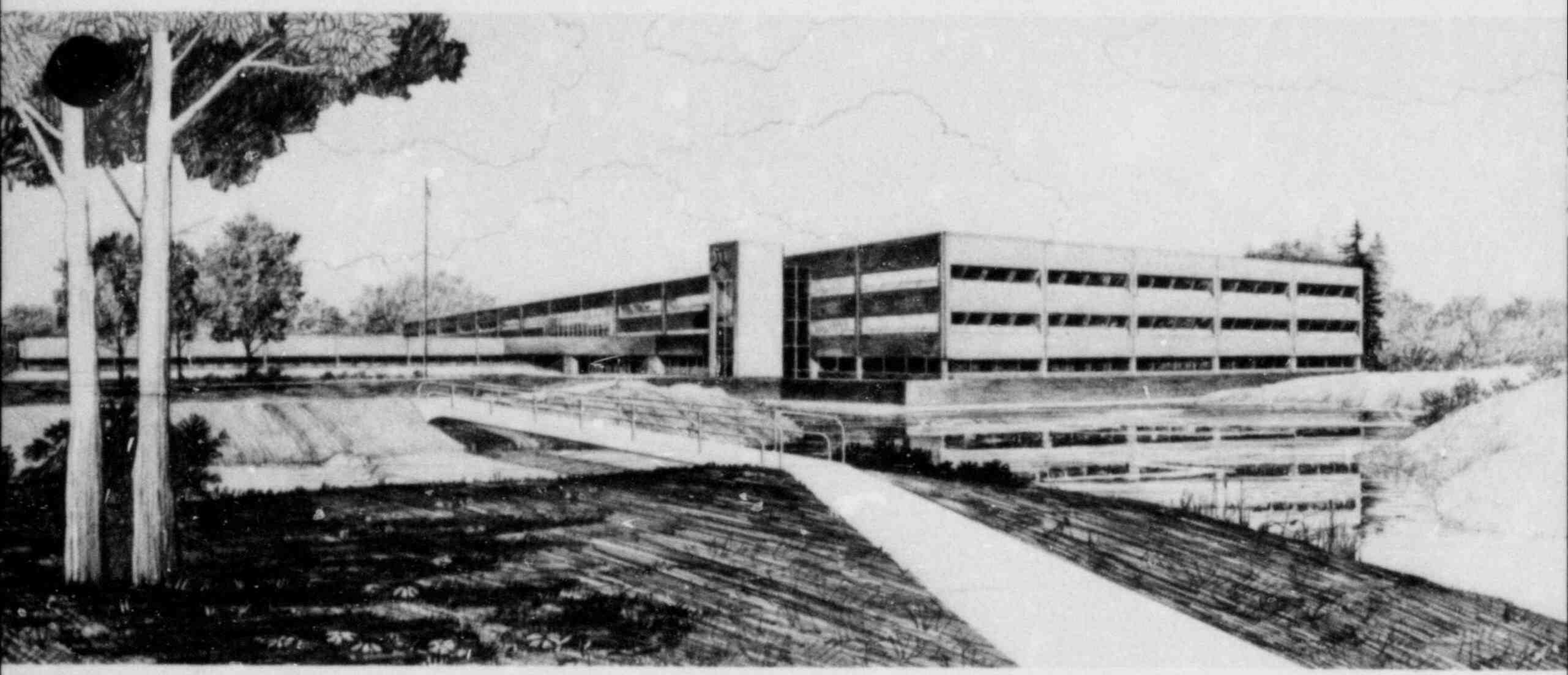
PDR
LPDR
NSIC
NTIS
CF

THE STATION BLACKOUT TRANSIENT AT THE BROWNS FERRY UNIT ONE PLANT

A SEVERE ACCIDENT SEQUENCE ANALYSIS

R. R. Schultz
S. R. Wagoner

Idaho 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
EG&G-398
(3-82)

INTERIM REPORT

Accession No. ____________

Report No. EGG-NTAP-6002

Contract Program or Project Title: NRC Technical Assistance Program Division

Subject of this Document: The Station Blackout Transient at the Browns Ferry Unit One Plant - A Severe Accident Sequence Analysis

Type of Document: Technical Report

Author(s): R. R. Schultz, S. R. Wagoner

Date of Document: September 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

ABSTRACT

As a part of the charter of the Severe Accident Sequence Analysis (SASA) Program, the station blackout transient has been analyzed using a RELAP5 model of the Browns Ferry Unit-1 Plant. The task was conducted as a partial fulfillment of the needs of the Nuclear Regulatory Commission in examining the Unresolved Safety Issue A-44: Station Blackout.

The station blackout transient was examined (a) to define the equipment needed to maintain a well cooled core, (b) to determine when core uncovery would occur given equipment failure, (c) to evaluate the quantities of mass and energy delivered to the containment pressure suppression pool and (d) to characterize the behavior of the vessel thermal-hydraulics during the station blackout transient (in part as the plant operator would see it). These items are discussed in the report.

Conclusions and observations specific to the station blackout are presented.

FIN No. A6354--Severe Accident Sequence Analysis

SUMMARY

Operating plant transients are of great interest for many reasons, not the least of which is the potential for a mild transient to degenerate to a severe transient yielding core damage. Using the Browns Ferry (BF) Unit-1 plant as a basis of study, the station blackout sequence was investigated by the Severe Accident Sequence Analysis (SASA) Program in support of the Nuclear Regulatory Commission's Unresolved Safety Issue A-44: Station Blackout. A station blackout transient occurs when the plant's AC power from a commercial power grid is lost and cannot be restored by the diesel generators. Under normal operating conditions, if a loss of offsite power (LOSP) occurs [i.e., a complete severance of the BF plants from the Tennessee Valley Authority (TVA) power grid], the eight diesel generators at the three BF units would quickly start and power the emergency AC buses. Of the eight diesel generators, only six are needed to safely shut down all three units.

Examination of BF specific data show that LOSP frequency is low at Unit 1. The station blackout frequency is even lower (5.7×10^{-4} events per year) and hinges on whether the diesel generators start. The frequency of diesel generator failure is dictated in large measure by the emergency equipment cooling water (EECW) system that cools the diesel generators.

Once a station blackout has occurred, the station operator is most concerned about starting the diesel generators and reconnecting the station to the TVA power grid. However, until AC power is restored, the operator will have (a) the plant station battery (available for 7 h), (b) both the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems [i.e., the vessel water inventory (VWI) equipment], and (c) 135,000 gallons of water reserved in the condensate storage tank (CST) [a Technical Specification (TS) Limit].

Major objectives of the SASA Program analysis of the BF Unit-1 station blackout sequence were to (a) characterize the transient as the plant operator would see it, (b) determine the time to core uncovery, and

(c) calculate the quantity of mass and energy transferred from the reactor pressure vessel to the pressure suppression pool (PSP). The analysis was performed using the RELAP5 Mod-1 Cycle 13 thermal-hydraulic code. However, several updates were used both to improve the cycle and to make the cycle appropriate to a boiling water reactor (BWR). Specifically, updates were added to enhance the behavior of the interphase drag models and the reactivity feedback models. In addition, the jet pump was treated as a special component in the BF model. Finally, the RELAP5 separator model was updated.

In the event a LOSP occurs, the power-load (time zero) unbalance experienced by the plant's main generator will be sufficient to initiate a reactor scram. The reactor vessel will be isolated during the scram phase of the transient. During the much longer period when only core decay heat is present, the VWI will boil and flash. The excess steam will bleed from the vessel through the safety relief valves (SRVs) to the PSP. Initially enough VWI will be lost so that the water level in the vessel downcomer will not be measurable on the level instrumentation available in the control room. The Emergency Operating Procedures (EOPs) dictate that the RCIC and the HPCI systems be manually activated as soon as possible. However, if the operator does not activate these systems, enough VWI will be lost to activate the vessel downcomer low-low trip and, thus, automatically turn on the RCIC and HPCI systems.

If neither the HPCI nor RCIC is available, the VWI will continually decrease as the SRVs open at regular intervals to bleed steam to the PSP. Core heatup would begin at 2300 s under such circumstances (but at 1680 s if a SRV became jammed open at the first cycle).

However, if either or both of the RCIC and HPCI systems are activated, water from the CST will be pumped into the vessel through the feedwater sparger. CST water will be delivered until either the operator manually shuts the systems off or the downcomer fills to the high level trip elevation, which will automatically deactivate the RCIC/HPCI systems.

Throughout the sequence, the SRVs will cycle either automatically as the vessel pressure exceeds plant setpoints, or manually as the operator follows EOPs. Left in the automatic mode, the same SRV will probably open and shut repeatedly to bleed the steam boiled by the core decay heat. Thus, a localized hot spot will be formed in the PSP at the SRV discharge port.

To change the location of steam delivery to the PSP and to reduce the number of SRV cycles, the operator is directed by the EOP to manually cycle alternate SRVs over a larger pressure range, e.g., about 200 psi. Such a procedure could be followed indefinitely if it were not for the heat transfer from the vessel to the drywell environment. Without the drywell coolers (lost during the LOSP), the drywell atmosphere will be heated to temperatures exceeding drywell seal specifications, which will lead to an increased potential for containment failure.

Consequently, the operator will probably depressurize the vessel after an initial waiting period (about an hour) at a rate that would not change the vessel temperature by more than 100°F/h (a TS limit). Thus, the heat load to the drywell would be reduced by the corresponding decrease in saturation temperature and the excess energy would be transferred to the PSP.

Several significant conclusions and observations resulted from the study:

1. A station blackout transient is improbable. Equipment or system unavailabilities yield a station blackout event frequency of 5.7×10^{-4} events/year.

2. The VWI equipment available to the operator during a station blackout is sufficient to maintain the VWI during the time when the station battery is available, even with a stuck-open relief valve.

3. Ultimate shutdown of the plant can only be accomplished if AC power is restored together with the residual heat removal (RHR) system.

4. RELAP5 can be used to model BWR long term sequences. The code has completed 7.8 and 9.7 h transients in 4.9 and 4.1 h calculated times, respectively.

ACKNOWLEDGEMENTS

The authors gratefully thank Messrs M. E. Stewart, S. E. Mays and W. H. Sullivan for their valuable input concerning the equipment unavailablities and the event frequencies discussed in Section 2. Also the authors are indebted to Mesdames E. E. Jenkins, N. L. Mecham, G. D. Gatenby and J. M. Mosher for their excellent typing, proofing, and graphics efforts.

CONTENTS

FIGURES

TABLES

1. INTRODUCTION

Operating plant transients are of great interest for many reasons, not the least of which is the potential for a mild transient to degenerate to a severe transient (see Reference 1). Such a consideration provided the motivation to create the Severe Accident Sequence Analysis (SASA) Program.

In early 1980 the Browns Ferry Unit 1 nuclear power plant was selected as a key facility for analysis by the SASA Program based on the following:

1. The plant is close to a major population area i.e., Huntsville, Alabama.

2. A nuclear simulator of Browns Ferry is available for training purposes.

3. The Tennessee Valley Authority has a history of cooperation with beneficial programs such as SASA.

4. Browns Ferry Unit 1 is a 251 inch diameter vessel BWR-4 with a Mark I containment and as such is representative of the largest group of operational or near term operating license (NTOL) Boiling Water Reactors (BWR) in the U.S. (see Appendix A).

5. Representative test data is available from Browns Ferry for operational transients of interest.

6. Browns Ferry is operational.

7. The Two Loop Test Apparatus (TLTA) at the General Electric Co. facility at San Jose, California was scaled based on the Browns Ferry class plants (see Reference 2).

Using Browns Ferry Unit 1 as a basis of study, the station blackout sequence was investigated by the SASA Program in support of the Nuclear Regulatory Commission (NRC) Unresolved Safety Issue A-44: Station Blackout. A station blackout transient occurs when the plant AC power buses are lost and cannot be powered by the diesel generators.

Station blackout sequences have been examined by SASA Program groups at both the Oak Ridge National Laboratory (ORNL) (see Reference 3) and the Idaho National Engineering Laboratory (INEL). The analytical division between ORNL and INEL has been to allot the detailed thermal-hydraulic calculations from the beginning of the transient to the time of core uncovery to INEL. ORNL has calculated lumped parameter thermal-hydraulic conditions from the beginning of the transient to the time of catastrophic failure using both the BWR-LACP and MARCH (See Reference 3) codes. Such calculations have provided the containment drywell and pressure suppression pool (PSP) behavior for the complete station blackout transients.

The detailed vessel thermal-hydraulic station blackout calculations have been conducted at INEL using a RELAP5 model of the Browns Ferry Unit 1 vessel. The remainder of the report addresses the station blackout scenarios, the RELAP5 thermal-hydraulic model, how the model has been applied to analyze the selected station blackout sequences, the results of such analyses and conclusions which are derived from SASA Program studies to date.

The station blackout sequences are discussed in the following report format:

Section	Topic
2	The station blackout scenarios: the event tree, equipment unavailability, contributing events and operator guidelines are presented.
3	The RELAP5 model: The Browns Ferry vessel model is described together with the code modifications required to conduct the calculations.

Section	Topic
4	The RELAP5 station blackout analyses: the selected transients and detailed results are discussed.
5	Conclusions and recommendations.

The results discussed herein are a part of a much larger effort which will examine potentially dangerous transients at the Browns Ferry plant as identified by probabilistic risk assessment studies (see Reference 1). Furthermore, the effort relative to the Browns Ferry plant will be repeated to a degree by constructing detailed thermal-hydraulic models of other target plants (designated in Appendix A).[4]

2. STATION BLACKOUT

A station blackout transient occurs when the plant AC power buses are lost. Normally, the buses would be connected to the Tennessee Valley Authority (TVA) power grid. However, if a loss of off-site power (LOSP) occurred i.e., a complete severance of the Browns Ferry (BF) plants from the TVA grid the eight diesel generators available at the three units would quickly start and power the emergency AC buses. Of the eight diesel generators, only six are needed to safely shutdown all three of the BF units. Thus, it is highly unlikely that a complete loss of power would occur.

Examination of the data (see Reference 1) show that the LOSP frequency at the BF Unit 1 plant is low: 0.03 events/year. The station blackout frequency is even lower (see Figure 1). Given that a LOSP occurs (designated node T_L on Figure 1), a scram most likely would occur (node B) and enough of the thirteen safety-relief valves (SRVs) would open to limit the vessel pressure to a safe value (node J). Thereafter the possible event sequences branch (node K) to form two paths of interest i.e., dependent on whether all the SRVs close (0.028 events/year). A stuck open relief valve (SORV: 0.0017 events/year) is represented by the lower branch.

Whether a station blackout occurs is designated by node D/G i.e., do all the diesel generators fail? The frequency of node D/G is dictated in large measure by the emergency equipment cooling water (EECW) system.

The EECW system cools the diesel generators as well as other pieces of equipment. Table 1 lists the probability of a station blackout for various modes of failure of the EECW system. Although failure mode 1 has the highest probability of occurrence, it is not likely to have a disasterous impact since the operator could reduce the cooling load on EECW simply by removing nonessential equipment or supplying cooling water using the residual heat removal service water pumps (see Reference 5). Even so, the probability of failure mode 1 was used to calculate the frequencies of

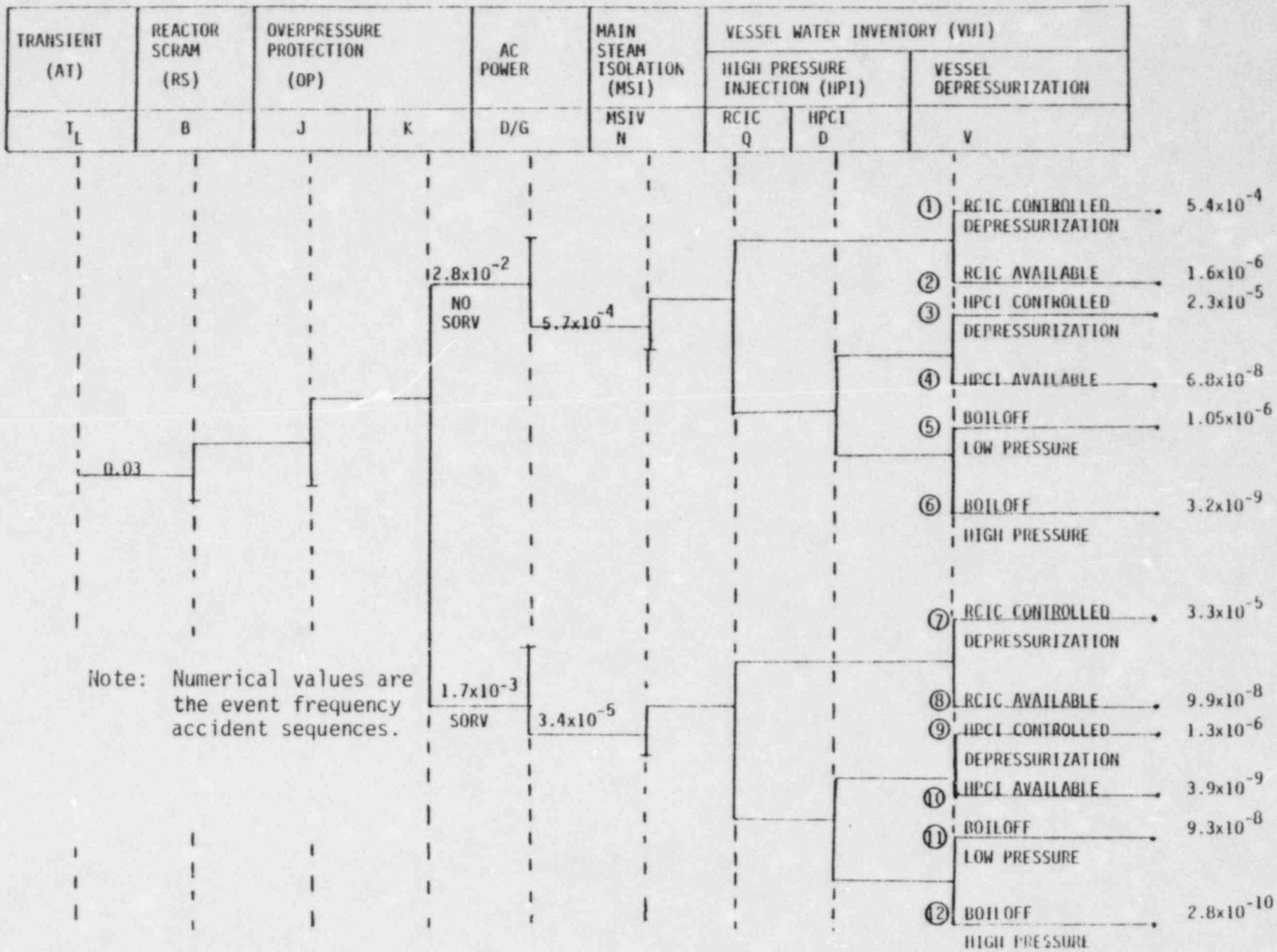


Figure 1. The loss of offsite power initiated station blackout accident sequences.

TABLE 1. FAILURE CAUSE AND PROBABILITY OF ALL DIESEL GENERATORS (D/Gs) AT THE BROWNS FERRY PLANTS

Failure Mode	Cause	Failure Probability No Operator Action	Impact on Diesel Generators
1	Failure of 2 of 4 EECW pumps	2.0×10^{-2}	Prolonged operation of diesel generators. Failure of D/Gs probably will not occur, but further analysis required.
2	Failure of 3 of 4 EECW pumps	2.3×10^{-3}	Failure time unknown
3	Failure of all EECW pumps	$\sim 10^{-4}$	Failure shortly after startup
4	Independent failure of all diesel generators	1.1×10^{-5}	No AC power available

possible station blackout sequences (see Figure 1). Failure modes 2 and 3 have greater potential for damaging the diesel generators since the operator would have less time to react. In summary, failure of the EECW system is the leading contributor to the station blackout scenario.

Given that a station blackout occurs, the main steam line isolation valves (MSIV) will most likely close (node N). Thus, subsequent to MSIV closure, twelve possibilities or station blackout paths are feasible depending on whether the operator chooses to use or has as an alternative: the reactor core isolation cooling (RCIC-node Q) or the high pressure coolant injection (HPCI-node D) systems. In addition, the operator could decide to depressurize the system (node V). These options form the basis for the calculations conducted with the Browns Ferry model.

Each of the possible options are numbered sequentially as V1 through V12 (see Figure 1). The sequences labeled as "boiloff" are unique since no means of replenishing the vessel water inventory is available and the water simply boils away.

Once a station blackout has occurred, the station operator is most concerned about starting the diesel generators and reconnecting the station to the TVA power grid. However, the operator also has several other questions to be answered in the interim to assure the plant's safe operation until AC power restoration:

1. How long will the station batteries be operable?

2. How much water is available in the condensate storage tank (CST) for use by the vessel water inventory (VWI) systems?

3. Are the high pressure VWI systems i.e., the HPCI and RCIC available?

These questions are all equipment/facility/operator procedure dependent. As such, given that the station blackout initiating event does not fail the above systems, the plant procedures will ensure that certain minimum standards exist for plant safety.

The station battery is projected to last for seven hours (Reference 3, Appendix G) based on a battery capacity analysis done by TVA engineers. Thus, the power source for the HPCI and RCIC would be available for a substantial period of time.

The CST capacity is 375,000 gallons of which 135,000 gallons (see Reference 5) is a guaranteed reserve for the emergency core cooling systems (ECCS) and the RCIC. So the guarenteed reserve alone is sufficient to provide vessel water inventory for a substantial time period. The length of time is one of the items discussed in the calculational results.

Both the HPCI and RCIC are discussed in detail in Reference 3. Thus, only the features of these systems which bear on the station blackout analyses will be discussed hereafter. Under normal circumstances both systems should be available for either automatic or operator governed actions.

3. THE BROWNS FERRY RELAP5 MODEL, CODE UPDATES AND INITIAL BOUNDARY CONDITIONS

The structure of this report is to confine update and model details to the appendices. Thus, the discussions given in the following subsections should be used in conjunction with Appendices B and C if the reader desires a detailed model treatment. Otherwise the main text will only give the broad picture describing the code updates, the models and the code input.

The code updates required to model a boiling water reactor (BWR) are briefly discussed in Subsection 3.1, and discussed in detail in Appendix B. The code version is also recorded in Subsection 3.1.

The RELAP5 Browns Ferry model used to characterize the plant behavior during postulated station blackout sequences is described in detail in Appendix B and in summary form in Subsection 3.2. The boundary and initial conditions used in the station blackout analyses are discussed in Subsection 3.3.

3.1 The RELAP5 Cycle and Updates

The calculations were conducted using Cycle 13 of the RELAP5/MOD1 code (Configuration Control Number FU0341). However, several updates were used with Cycle 13 both to improve the cycle and to make the code appropriate to a BWR.

Because RELAP5 MOD1 does not simulate the presence of momentum mixing, a jet pump cannot be modeled without modifying the code. The jet pumps were modeled using an update developed by Intermountain Technologies, Inc. (Reference 6--see Appendix B.1.2) that in effect simulates a pressure rise across the downcomer and the jet pump component to model the fluid momentum transfer from the drive jet to the suction flow.

The BWR separator proved to be another component that was difficult to model since the unaltered Cycle 13 hydrodynamic equations were unstable when the separator component was used. Thus, a specific update applicable to Junction 70 of Volume 875 (see Figure 2) was used throughout the station blackout studies. Further updates were also required to enhance the behavior of the interphase drag models.

During the course of conducting the station blackout analyses, system mass error became a problem at times, particularly for analyses which had continuous flows from the vessel i.e., the stuck open relief valve analysis. The mass error was found to be related principally to phase boundary crossings i.e., when the model volume in question changed from all steam to two phase or all liquid to two phase. A short discussion of the mass error problem is presented in Appendix C.

Finally, during the course of the station blackout calculations, the decay heat model was found to be low by approximately 8%. The corrective updates are available from the Code Development Branch at the INEL.

3.2 The RELAP5 Browns Ferry Models

The RELAP5 Browns Ferry model (see Appendix B) is shown in Figure 2. The illustrated model is the most detailed and represents both recirculation loops. A simplified version of the model has only one recirculation loop and a coarse nodalization scheme in the recirculation loop and the steam lines (see Appendix B).

3.2.1 Summary Description of the Browns Ferry Models

The model includes the feedwater lines from the feedwater heaters to the vessel. Volumes 690 and 685 represent the lines upstream of the containment inboard check valves. The steam lines were modeled to the turbine control valve and include the presence of all the safety-relief valves (SRVs) and the turbine bypass valves.

The recirculation loops were modeled to include the recirculation pumps, the jet pumps, the reactor water cleanup, and the pump isolation valves.

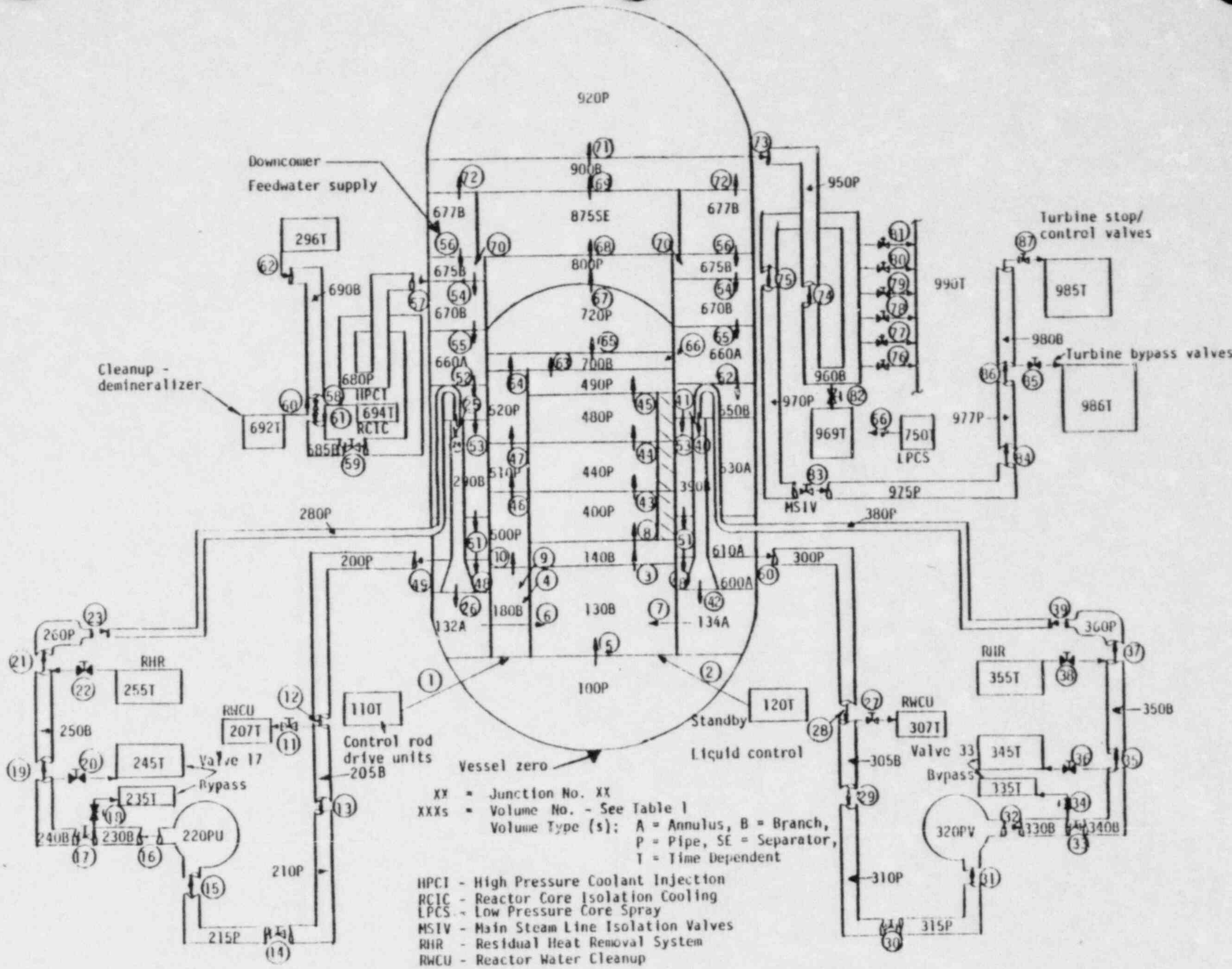


Figure 2. The Browns Ferry RELAP5 model.

The vessel was modeled to include all internal components. Specifically, the core was modeled as an average bundle with three vertical heat slabs connected to Volumes 400, 440, and 480, respectively. The core bypass i.e., Volumes 500, 510, and 520 were modeled to receive flow from the lower plenum, the guide tubes and the bundle core inlets. The steam separators were represented by Volume 875 and the steam dryer region by Volume 900. Thus, separator components were used for both these volumes for most analyses.

The emergency core cooling systems (ECCS) were modeled or provisions were included to allow input of the desired boundary conditions. The low pressure core spray (LPCS) was included using Volume 750. The low pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system was included using Volumes 255 and 355. Finally the high pressure coolant injection (HPCI) system was modeled to inject water into the feedwater line at Volume 685 from Volume 694.

The reactor core isolation cooling (RCIC) system, which is not an ECCS, was modeled in the same manner as the HPCI. The RCIC steam (and the HPCI steam turbine) turbine was modeled to remove steam from the steam line Volume 960 to Volume 969.

3.2.2 Applicability of the Browns Ferry Models to the Station Blackout Sequences

Although additional information are needed--see Appendix B.5, the current model capability is sufficient to meet the needs of the station blackout studies. Specifically, the modeling detail present in the models is sufficient to obtain (taken from Reference 7):

a. The time to core uncovery.

b. The time of actuation and the systems which must be actuated to prevent the core from uncovering.

c. The time history of the vessel pressure and level, the reactor coolant makeup rate and the SRV discharge rate.

since only global vessel thermal-hydraulics are required. For example detailed core geometry and fuel peaking factors are not crucial information when the core remains covered, the core mass flows are low and only the core decay heat is being transferred to the core fluid. In addition, the vessel pressure and level history (used to trip vessel water inventory equipment on and off) given by the model should be adequate since the vessel water and metal sensible heat release rates are accurately distributed and included in the model together with the core decay heat.

3.3 The Browns Ferry Boundary and Initial Conditions

The initial pressure and flow distributions were taken directly from information provided by the Tennessee Valley Authority (see Reference 8) for the Browns Ferry plant at 100% power and rated flow. Initial conditions are listed in Table 2.

Flow rates for the safety-relief valves (SRVs) were taken from the FSAR (Reference 5)--see Appendix B.1.4. The turbine bypass valves were sized to pass 3.99×10^6 lbm/hr at an upstream pressure of 950 psig (see Reference 9).

The reactor core power was distributed from the lower to the upper slab in a 39, 38, and 23% split, respectively (as listed in Reference 8). Each core slab was equal in size and 4.056 ft long. Point kinetics together with the RELAP5 fission product decay (including actinides) were used in all calculations.

Flow rates for the HPCI and RCIC systems pumps and drive turbines were taken directly from the FSAR. The fluid pumped by the HPCI/RCIC systems from the condensate storage tank (CST) to the vessel was assumed to be at 1150 psia, 140 F. Such a high temperature was assumed to crudely account for the hot environment surrounding the feedwater lines during a prolonged station blackout transient without the drywell coolers operable.

TABLE 2. BROWNS FERRY MODEL INITIAL CONDITIONS

Core power = MWt	3318.0
Mass flow rates = lbm/s	
Bypass	2801.3
Control rod drive	13.9
Core	25530.0
Feedwater	3740.0
Jet pump drive (one loop)	4686.1
Jet pump suction (one loop)	9473.2
Reactor water cleanup (one loop)	18.5
Steam line	3743.7
Pressure = lbf/in^2 absolute	
Lower plenum	1050.4
Steam dome	1014.0
Downcomer mixture level[a] = ft	45.2
Vessel enthalpies = BTU/lbm	
Feedwater	351.08
Lower plenum	519.75

a. Collapsed water level, elevation relative to the vessel zero.

4. STATION BLACKOUT SEQUENCE RESULTS

The station blackout results obtained using the Browns Ferry RELAP5 plant model are discussed herein. The results are discussed in a general format in Subsection 4.1. Thereafter six calculated transients are presented on an individual basis in Subsections 4.2 through 4.7. The results are summarized in Subsection 4.8.

4.1 Overview

In the event a LOSP occurred, the power load (time zero) unbalance experienced by the plant main generator would be sufficient to initiate a reactor scram (see Table 3). The turbine bypass valves would quickly open (fully open in 0.1 s), but would only pass about a third of the rated steam flow. Also, the turbine control valve would rapidly shut (fully closed in 0.2 s). Consequently, the vessel pressure would rapidly increase to the safety-relief valve (SRV) setpoints. Four SRVs would open to limit the vessel pressure peak. The scram would be complete by 3.9 s. If no SRVs failed open, three would be closed by 13 s and the fourth by 22 s as the reactor total power decreased to the decay level.

During the scram phase of the transient and the much longer period when only core decay heat is present, the vessel liquid inventory would boil and flash. The excess steam would bleed from the vessel through the SRVs to the pressure suppression pool (PSP). Initially enough vessel liquid inventory would be lost so the level could not be measured on the downcomer level instrumentation available in the control room. Under these conditions the Emergency Operating Procedures (EOP) dictate that the RCIC and the HPCI systems be manually activated as soon as possible. However, even if the operator did not act, in time enough vessel inventory would be lost to activate the vessel downcomer low-low trip[a] and thus, automatically turn on the RCIC and HPCI systems.

---

a. Elevation = 39.67 ft above vessel zero.

TABLE 3. CHRONOLOGY OF THE PRIMARY THERMAL-HYDRAULIC EVENTS

Time (s)	Event
0	Loss of off-site power occurs. Power-load unbalance occurs. The turbine control valves receive a signal to close. Scram logic is initiated. The recirculation pumps trip off and the feedwater system begins coastdown.
0.1	The turbine bypass valves are fully open.
0.2	The turbine control valves are fully shut.
3.0	Four safety-relief valves open.
3.9	The scram is complete.
4.0	The reactor water cleanup systems receive a signal to isolate. The control rod discharge flows becomes zero.
5.0	The feedwater pump coastdown is complete. The main stream line isolation valves (MSIVs) receive a signal to close.
8.0	The MSIVs shut.
9.0	The reactor water cleanup system flows are zero.
13.0	Three safety relief valves shut.

Once either or both the RCIC and HPCI systems were activated, water from the condensate storage tank (CST) would be pumped into the vessel through the feedwater sparger. CST water would continue to be delivered until either the operator manually shut the systems off or the downcomer filled to the high level trip elevation[a] which would automatically deactivate the RCIC/HPCI systems.

Throughout the transients, the SRVs would cycle automatically as the vessel pressure exceeded plant setpoints or manually as the operator followed EOPs. If the SRVs were left in the automatic mode, the SRVs would open at vessel pressures of 1131 psia (FSAR value plus one percent) and close after a 50 psi decrease. Left in the automatic mode, the same SRV would probably open and shut repeatedly to bleed the steam boiled by the core decay heat. Thus, a localized hot spot would be formed in the PSP at the SRV discharge port.

To change the location of steam delivery to the PSP and to reduce the number of SRV cycles, the operator is directed by the EOP to manually cycle alternate SRVs over a larger pressure range e.g., open at 1100 psig and close at 900 psig. Such a procedure could be followed indefinitely if it were not for the heat transfer from the vessel to the drywell environment. Without the drywell coolers, (lost during the LOSP) the drywell atmosphere would be heated to temperatures exceeding drywell seal specifications (see Reference 3) with an increased containment failure potential.

Consequently, the operator would probably depressurize the vessel after an initial waiting period (about an hour) at a rate which would not change the vessel temperature by more than 100°F/h (a Technical Specification Limit). Thus, the heat load to the drywell would be reduced by the corresponding decrease in saturation temperature and the excess energy would be transferred to the PSP.

---

a. Elevation = 48.5 ft above vessel zero.

Six analyses have been conducted assuming a LOSP as an initiating event i.e., Sequences V6, V12, V2, V4, V8, and V1--see Figure 1. Each of the analyses is briefly described, a minimum time to core uncovery is given (if appropriate) and the plant systems required to maintain core coverage for an indefinite period of time are listed. Instrument ranges available in the Browns Ferry Plant are superimposed on the downcomer water level plot. Other instrumentation available in the control room allows the reactor vessel pressure to be monitored using a 0 to 1200 psig pressure gauge (Reference 3). Also, instrumentation is available to monitor the full flow range provided to the vessel by the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems.

Each of the six sequences follows the same path initially. The six cases are identical until 22 s when the last safety-relief valve (SRV) would close unless a SRV failed open. The second and fifth calculations assume a safety-relief valve remains stuck open (SORV). The calculations are discussed on an individual basis in the following subsections.

Each calculation is portrayed with a set of seven figures. The figures are discussed in summary fashion in Subsection 4.2 for Sequence V6. Thereafter, only some of the figures are specifically discussed in Subsections 4.3 through 4.7. However, the remaining figures are given in Appendices D, E, F, G, and H for Sequences V12, V2, V4, V8, and V1, respectively to provide boundary conditions for further analysis.

4.2 The Boiloff Transient - Sequence V6

Assumptions inherent in the boiloff transient are that following the LOSP, all equipment is unavailable that would allow the operator to replenish the vessel water inventory. Thus, the operator might allow the plant to behave as designed while he tried to find a means of replenishing the vessel water inventory. With the plant left in the automatic mode, the SRV would immediately open (at 3s--see Table 3) and relieve the vessel pressure to 1081 psia. Thereafter a SRV would open whenever the vessel pressure reached 1131 psia (see Figure 3) and would again close at 1081 pisa. The length of time between SRV cycles is a direct function of

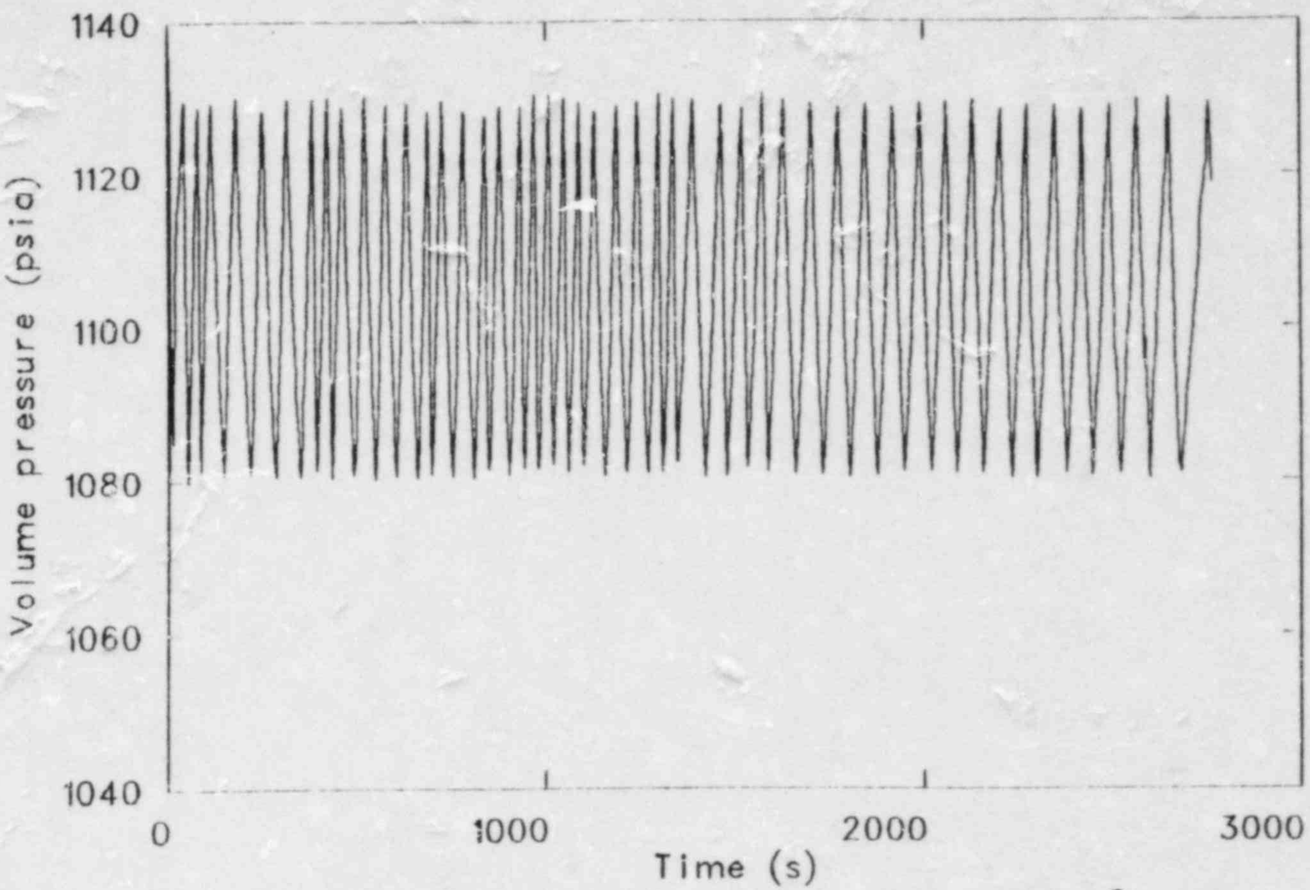


Figure 3. Vessel steam dome pressure - sequence v6

the core decay heat (as the vessel pressure increases); and a direct function of the core decay heat together with the vessel inventory flashing rate (as the vessel pressure decreases to the SRV shutoff pressure).

The behavior of the core temperature (see Figure 4) mirrors that of the vessel pressure when the fuel rods are well cooled. Thus, the core temperature increases and decreases with the saturation temperature adjacent to the fuel rod.

The vessel water inventory (see Figures 5 and 6) continually decreases in the V6 sequence. The water level measured in the vessel downcomer (see Figure 2) immediately decreases as the turbine control valve shuts and the vessel pressure rises. Such behavior results from extensive void collapse, the loss of the recirculation pumps and the loss of the feedwater pumps. Thus, as the mass flow (see Figure 7) through the core decreases (resulting from the recirculation pumps trip and the scram) the water level in the core shroud (Figure 6) and the downcomer tend to be equal.

The water level in the vessel then decreases with time as the system inventory is discharged through the lowest pressure setpoint SRV to the pressure suppression pool (PSP). The downcomer water level drops below the instrument range observable in the control room in the first few seconds of the transient (Figure 5). However, the plant operators can meter the water level by observing the instrumentation readings directly outside the control room (Range 2--Figure 5).

The collapsed water level inside the core shroud is not metered by the operator since no instrumentation is available. However, the collapsed water level falls below the top of the active fuel (TAF) at 1630 s (see Figure 6). But, core heatup does not occur since the water froth and core mass flow is sufficient to maintain adequate core cooling.

By 1700 s (see Figure 5) the downcomer water level reaches the top of the jet pump suction elevation with complete uncovery established by 2000 s. The core begins a prolonged heatup at 2300s (see Figure 4) as the upper core volume void fraction becomes nominally one.

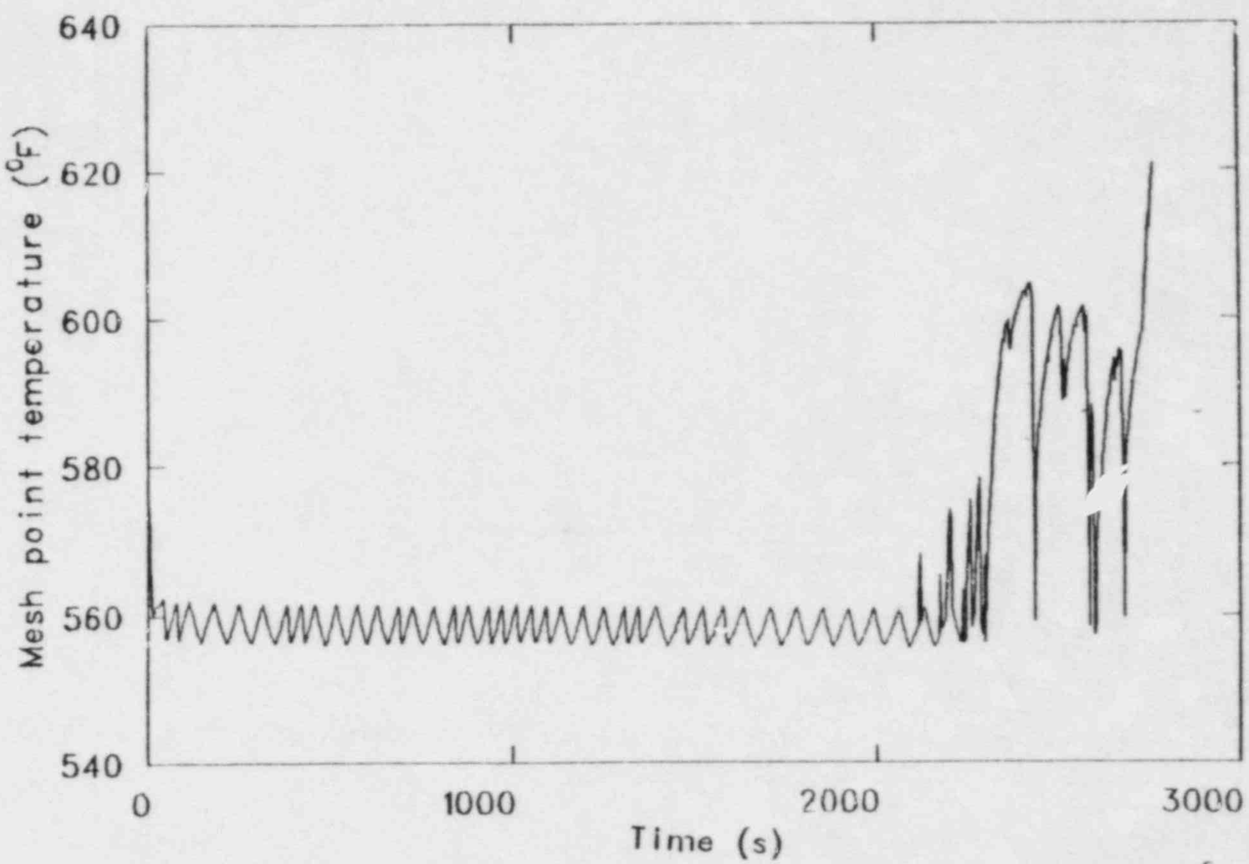


Figure 4. Upper elevation core temperature – sequence V6

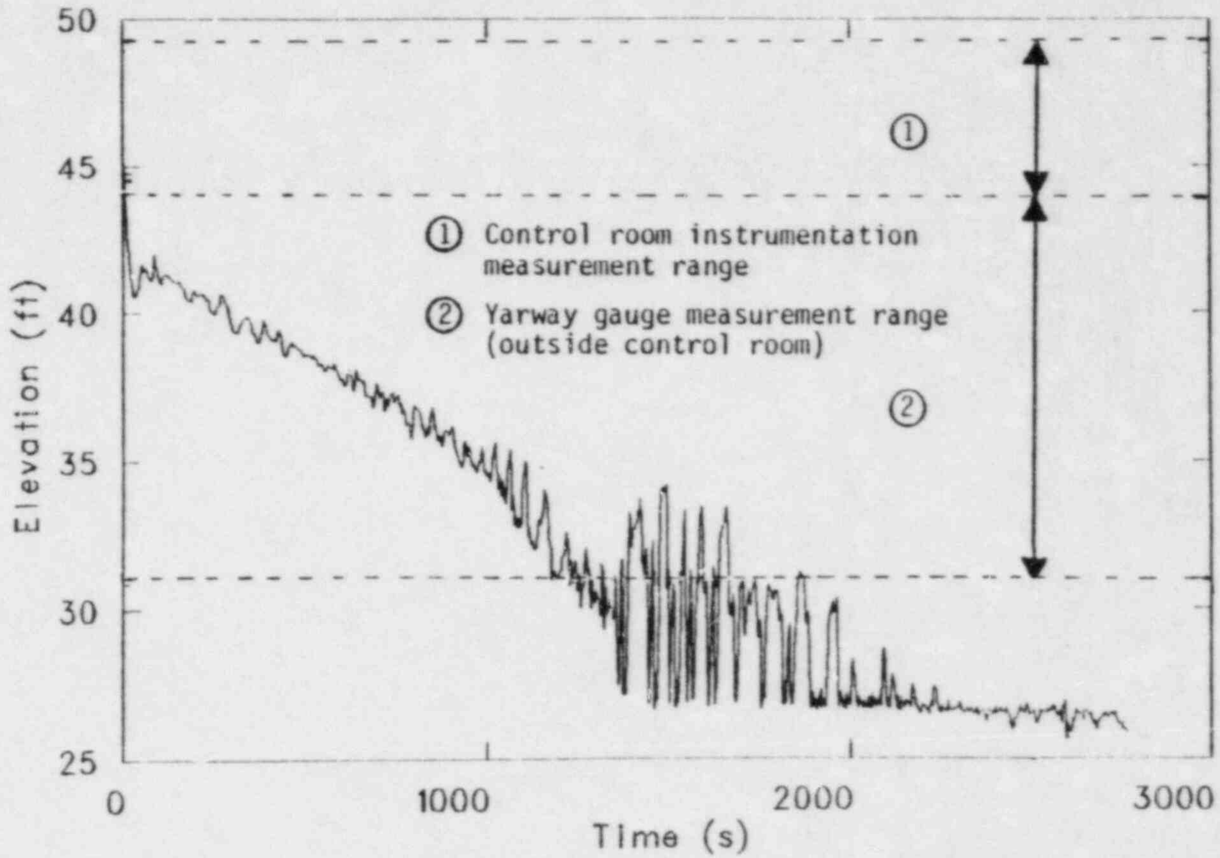


Figure 5. Downcomer water level. – sequence v6

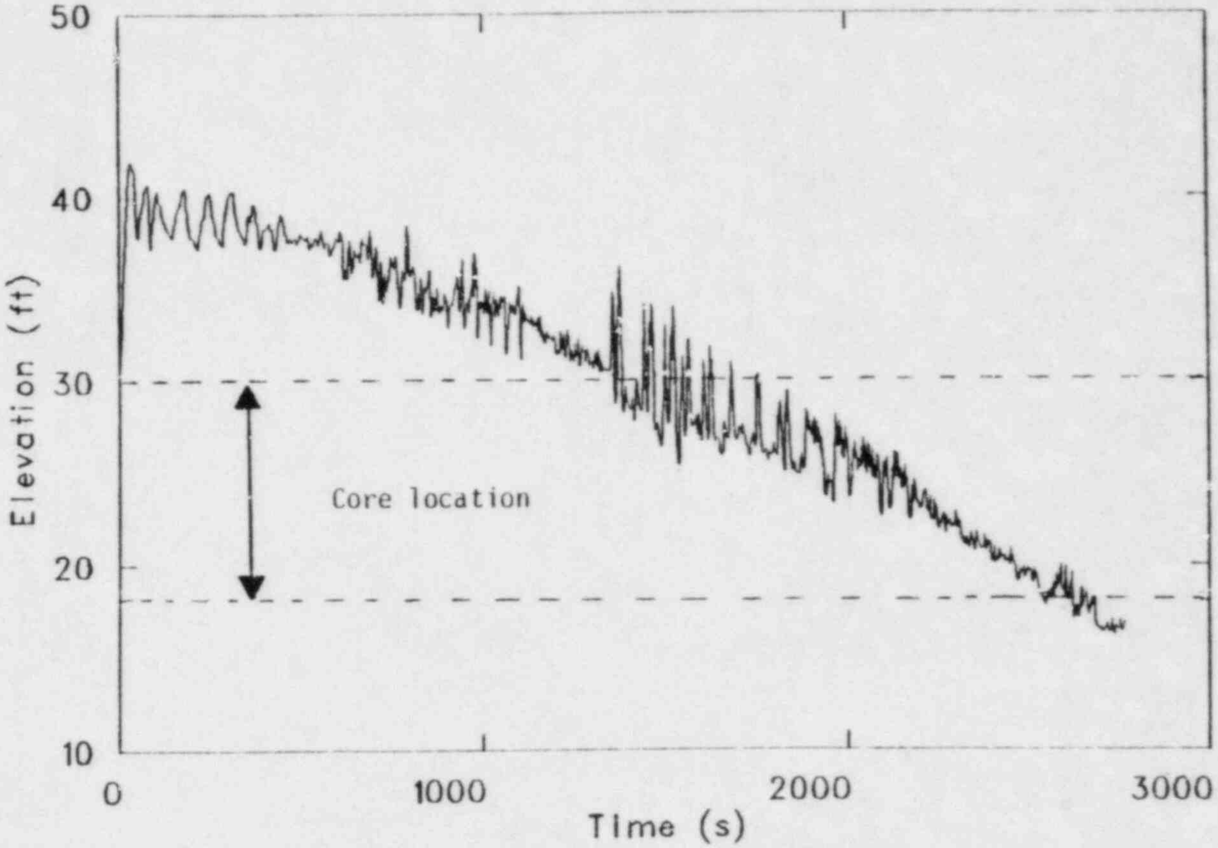


Figure 6. Core collapsed water level – sequence v6

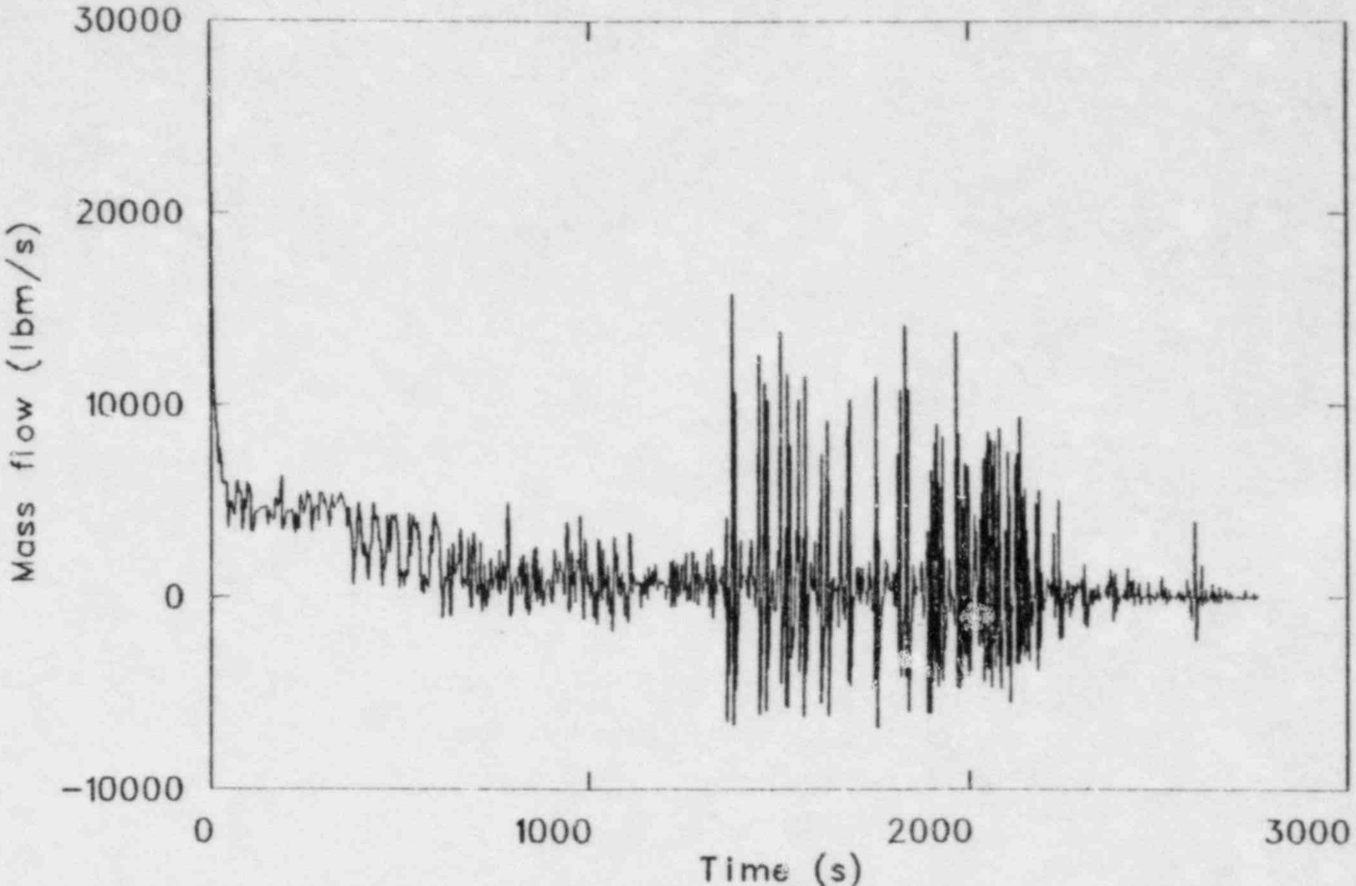


Figure 7. Core mass flow – sequence v6

The fact that core heatup is not shown by the model when the core collapsed water level reaches the top of the core is in full agreement with available data (BWR-FLECHT: Reference 10). Core average bundle power levels at approximately 2300 s after scram are between 60 to 80 kW. BWR-FLECHT data indicate that the tested core bundle geometry will remain well cooled with only a 6 ft elevation head. Inasmuch as the BWR-FLECHT 7 x 7 core geometry is similar to the BWR plant 8 x 8 geometry, an extension of the data to the BF plant core can be made. Thus an indication of core heatup at core collapsed water levels below 6 ft (times greater than 2200s) is expected. The analysis shows core heatup beginning at a core collapsed water level of 6 ft (t = 2300 s) in qualitative agreement with the data.

Throughout the transient, vessel mass and energy were delivered to PSP as depicted in Figures 8 and 9. The total mass delivered to the PSP by 2300 s was 245,000 lbm and the total energy delivered was 2.90 x 10^8 BTUs. The vessel receives no mass input (see Figure 10) beyond the mass delivered to the vessel as the feedwater pumps coasted to zero over the first 5 s of the transient.

Although extensive core uncovery would not be indicated if the HPCI or the RCIC systems were activated prior to 2300 s, the water inventory systems should be activated as early as possible since void collapse in the core shroud will occur as subcooled water is introduced into the vessel. Given that water inventory systems become available, the operator should strive to initiate the systems prior to 1600 s. If either or both the HPCI and the RCIC systems are available, the core will be adequately cooled for the 7 h period the station batteries are available. Thereafter, only initiation of systems independent of the plant DC buses will insure core coverage on the long term i.e., the residual heat removal system.

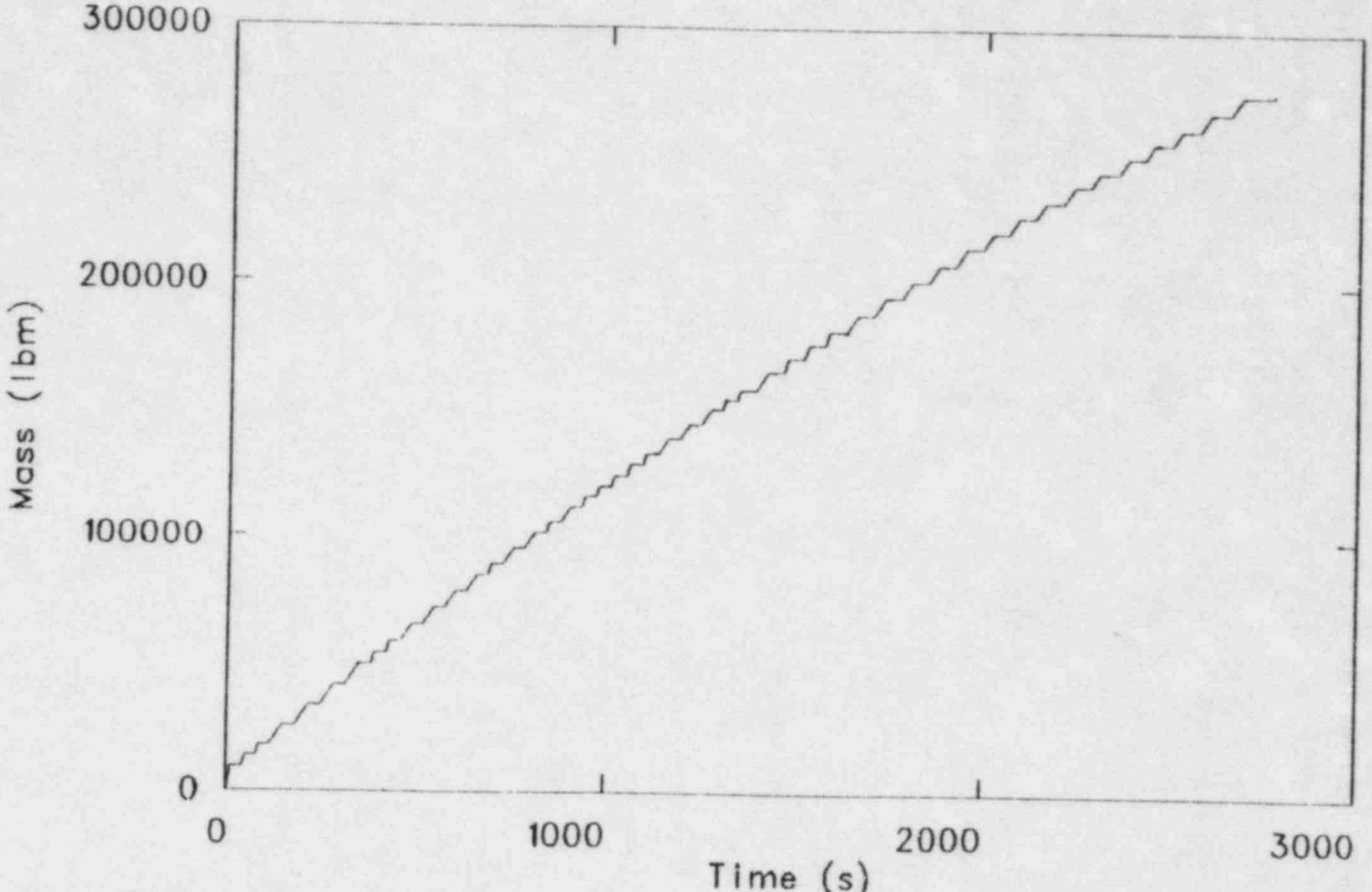


Figure 8. Total mass transfer to pressure suppression pool – sequence v6

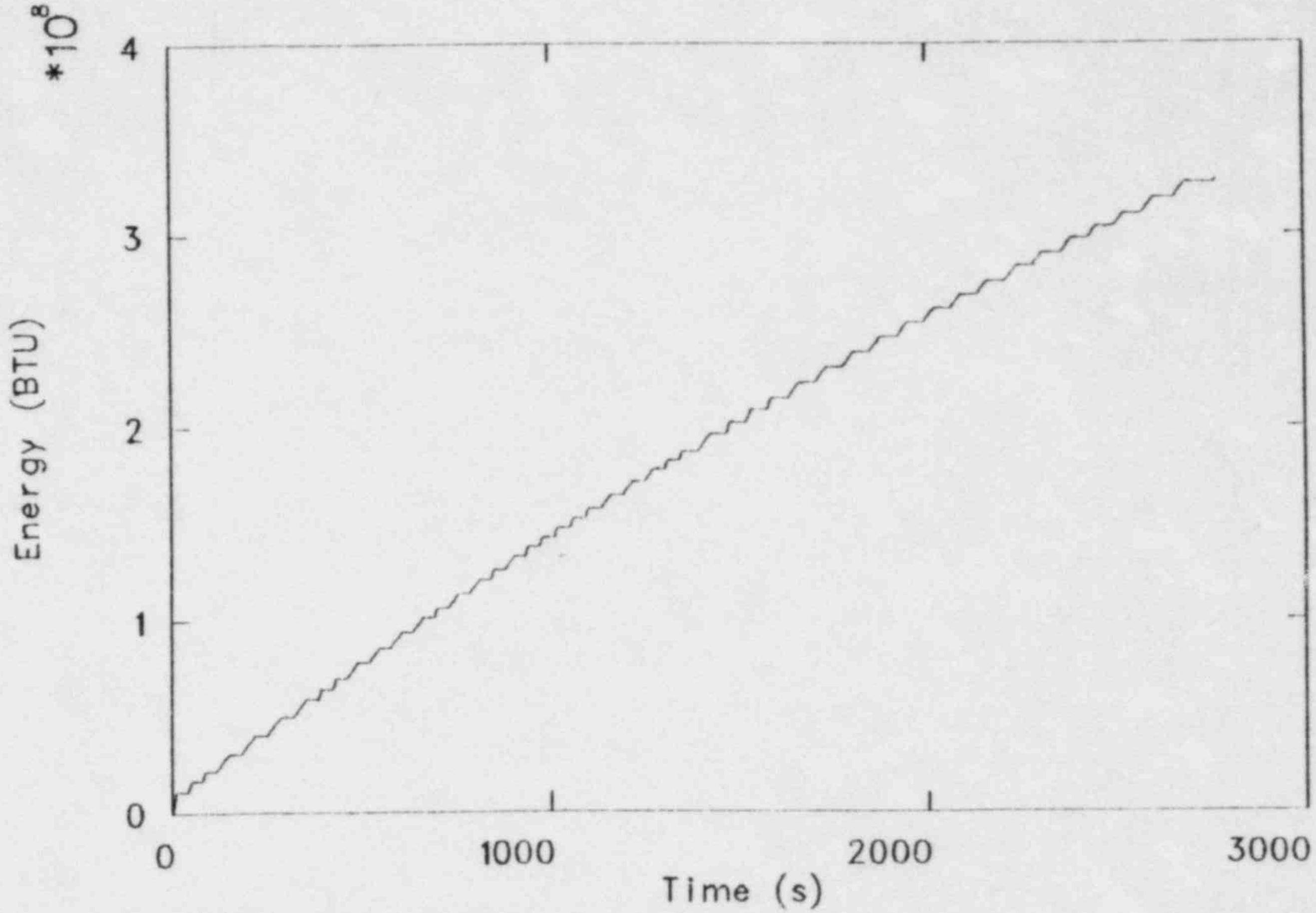


Figure 9. Total energy transfer to pressure suppression pool – sequence v6

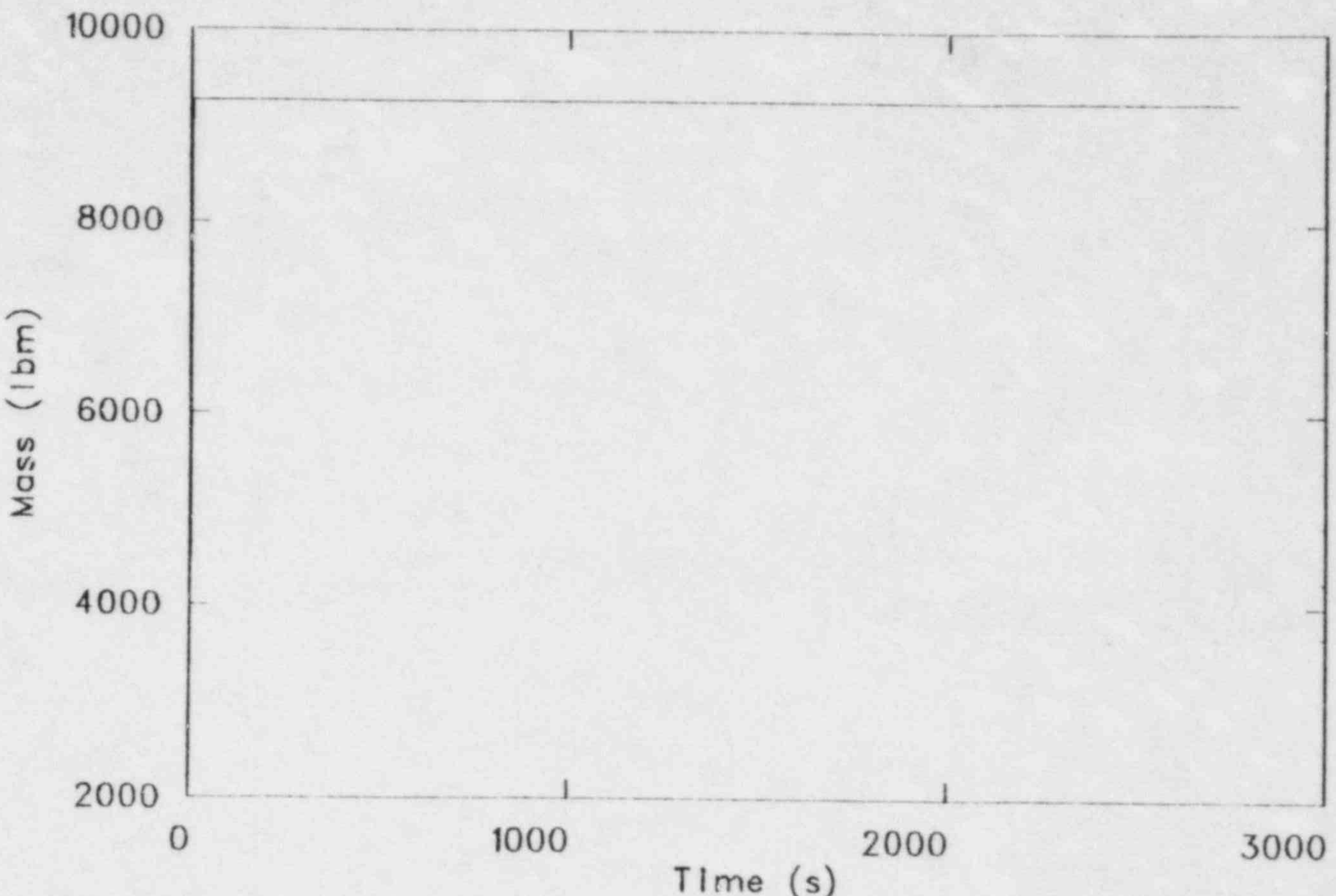


Figure 10. Total mass transfer to vessel - sequence v6

4.3 The Boiloff Transient with Stuck Open Relief Valve (SORV) - Sequence V12

The second calculation was based on the same set of assumptions as the first calculation except a SRV was assumed to fail open after it lifted the first time. Thus, without any systems available to replenish the vessel inventory, the vessel depressurizes (see Figure 11) until core uncovery occurs intermittently as early as 1580 s (see Figure 12). Core uncovery is sufficient by 1680 s to allow a sustained heatup of over 20 s. Additional figures are included in Appendix D.

Core uncovery would have been temporarily procluded if either the HPCI and/or RCIC system had been initiated prior to 610 s. Thereafter, only initiation of the residual heat removed system will insure core coverage on the long term.

4.4 The Station Blackout with the RCIC System Available - Sequence V2

The calculated results represent the sequence that would occur if Unit 1 experienced a station blackout when the HPCI system was unavailable. The operator would respond as soon as possible to the large drop in vessel water level (a 90 s response time is assumed in the calculation) by turning on the RCIC system (short term: Figure 13, long term: Figure 14). The SRVs would be opening and shutting at steam line pressures defined by their setpoints (see Figure 15). However, to distribute the steam discharge from the valves to the pressure suppression pool (PSP) uniformly, the operator was assumed to manually operate the valves 120 s after the station blackout occurred. The operator would open a valve at 1100 psig and discharge steam to the PSP until the vessel pressure was relieved to 900 psig. At the same time, the operator would use the RCIC to maintain the downcomer water level in the control room instrumentation measurement range i.e., between 44.8 and 48.2 ft (measured with respect to the vessel zero).

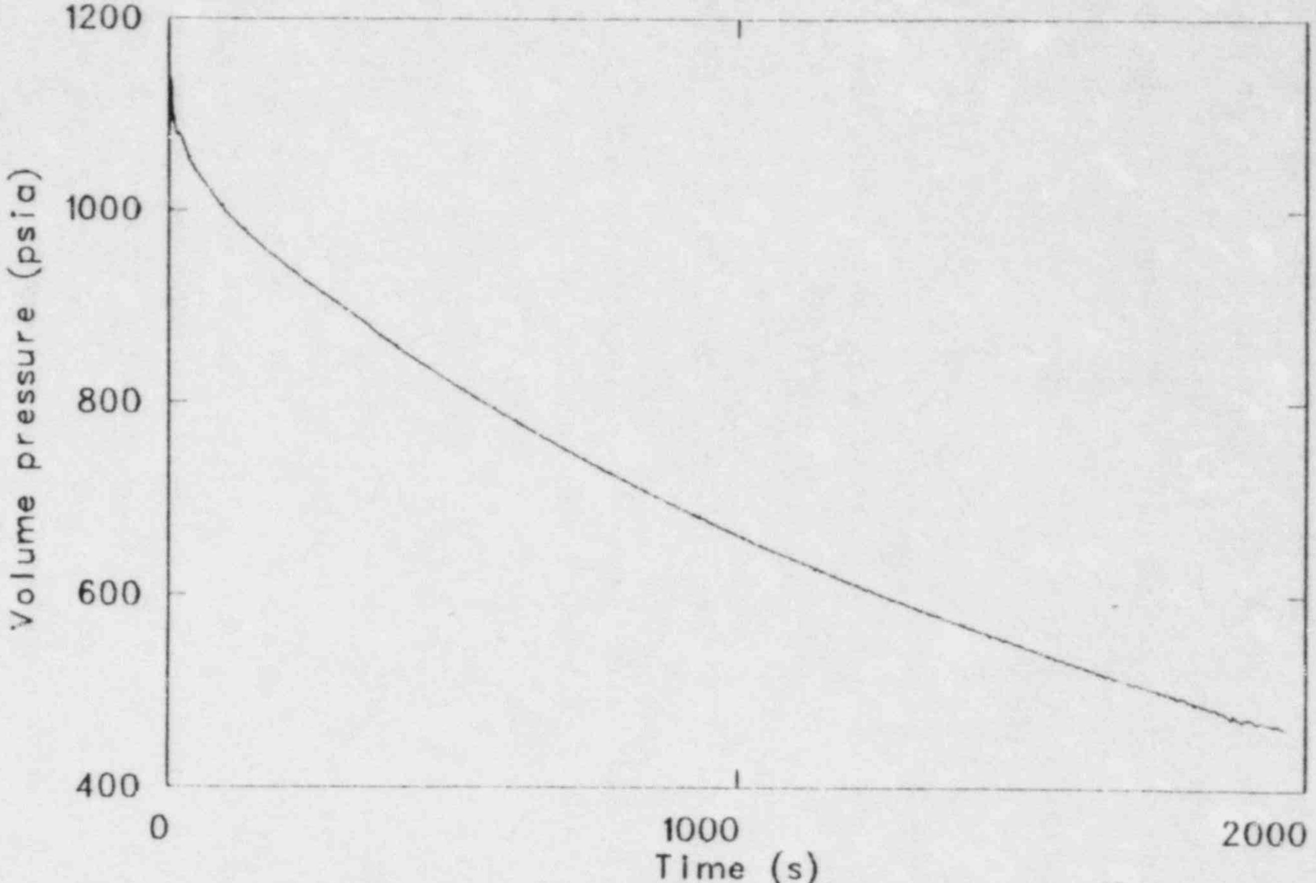


Figure 11. Vessel steam dome pressure - sequence v12

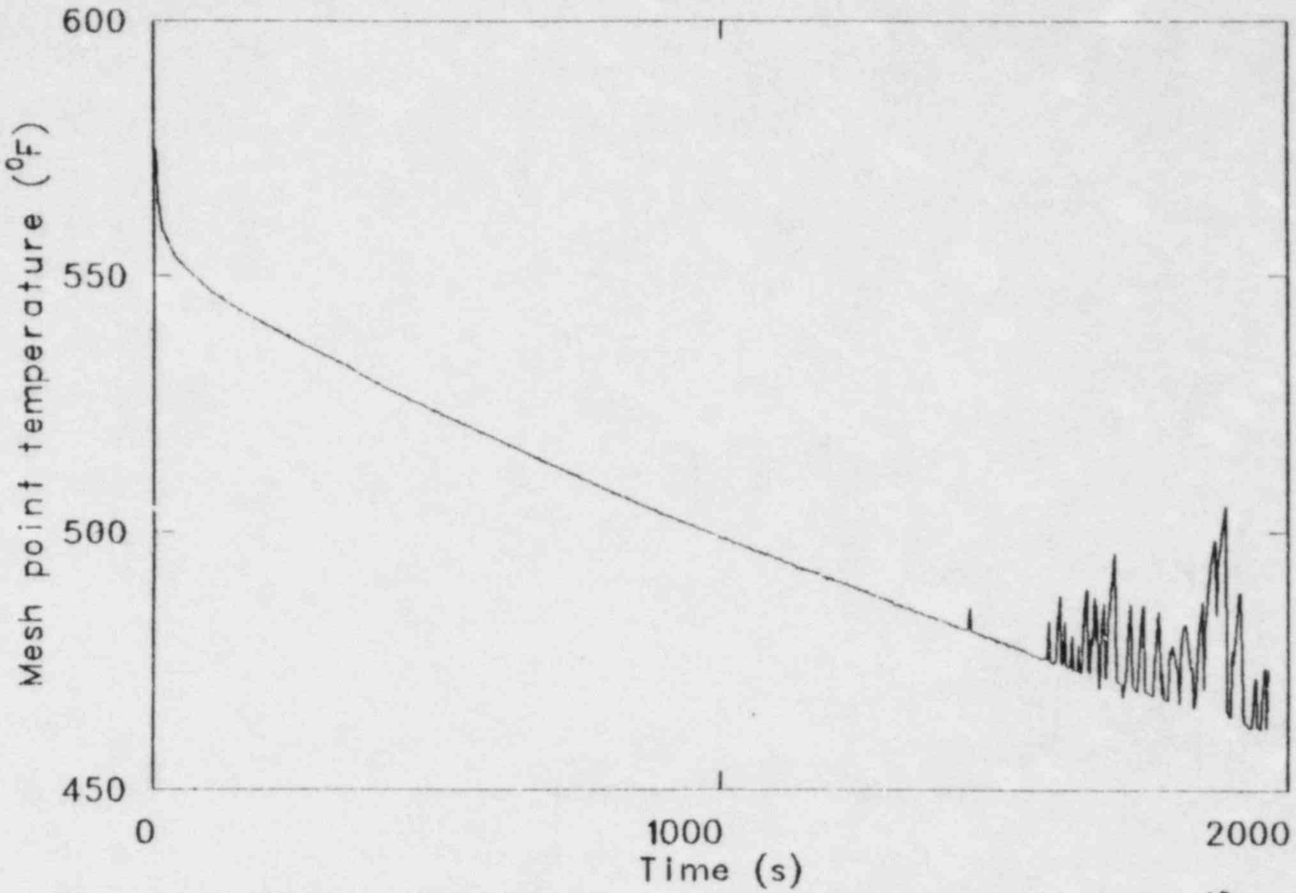


Figure 12. Upper elevation core temperature - sequence v12

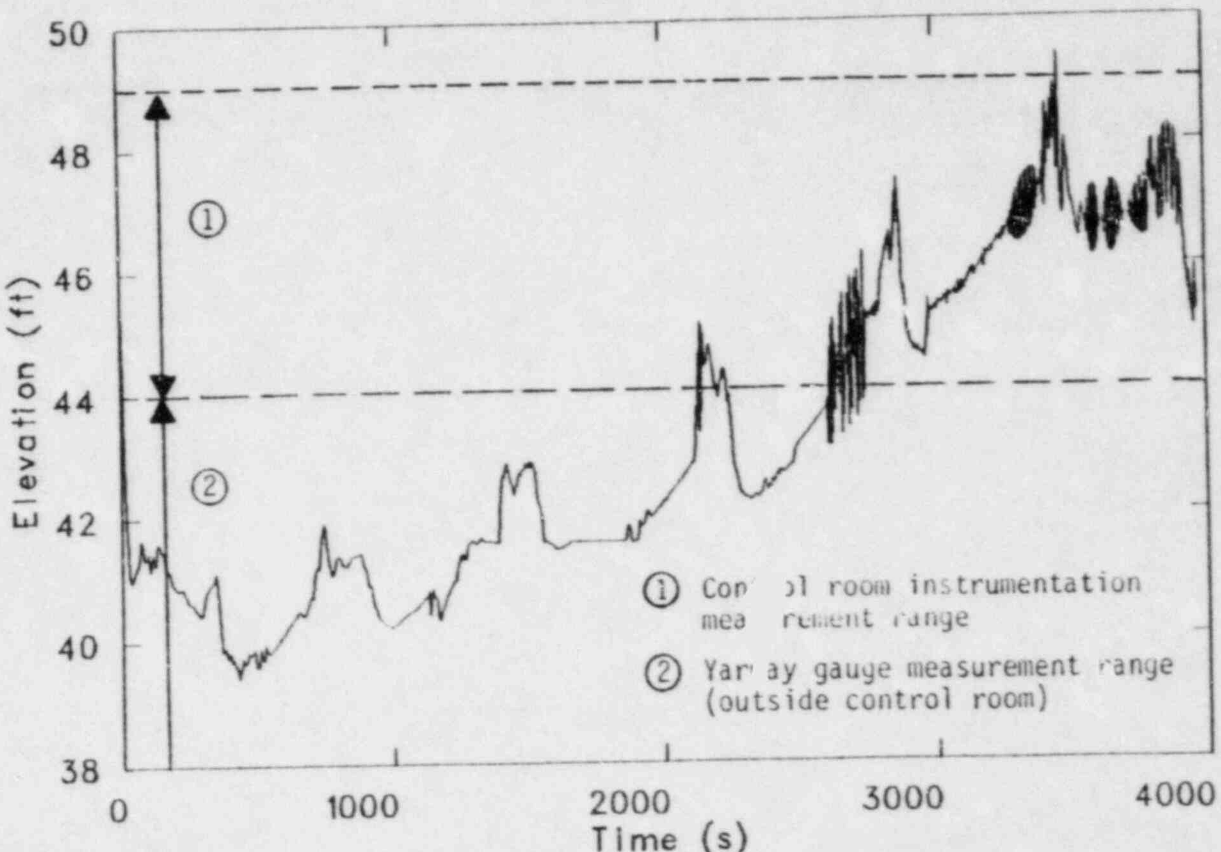


Figure 13. Downcomer water level (short term) - sequence v2.

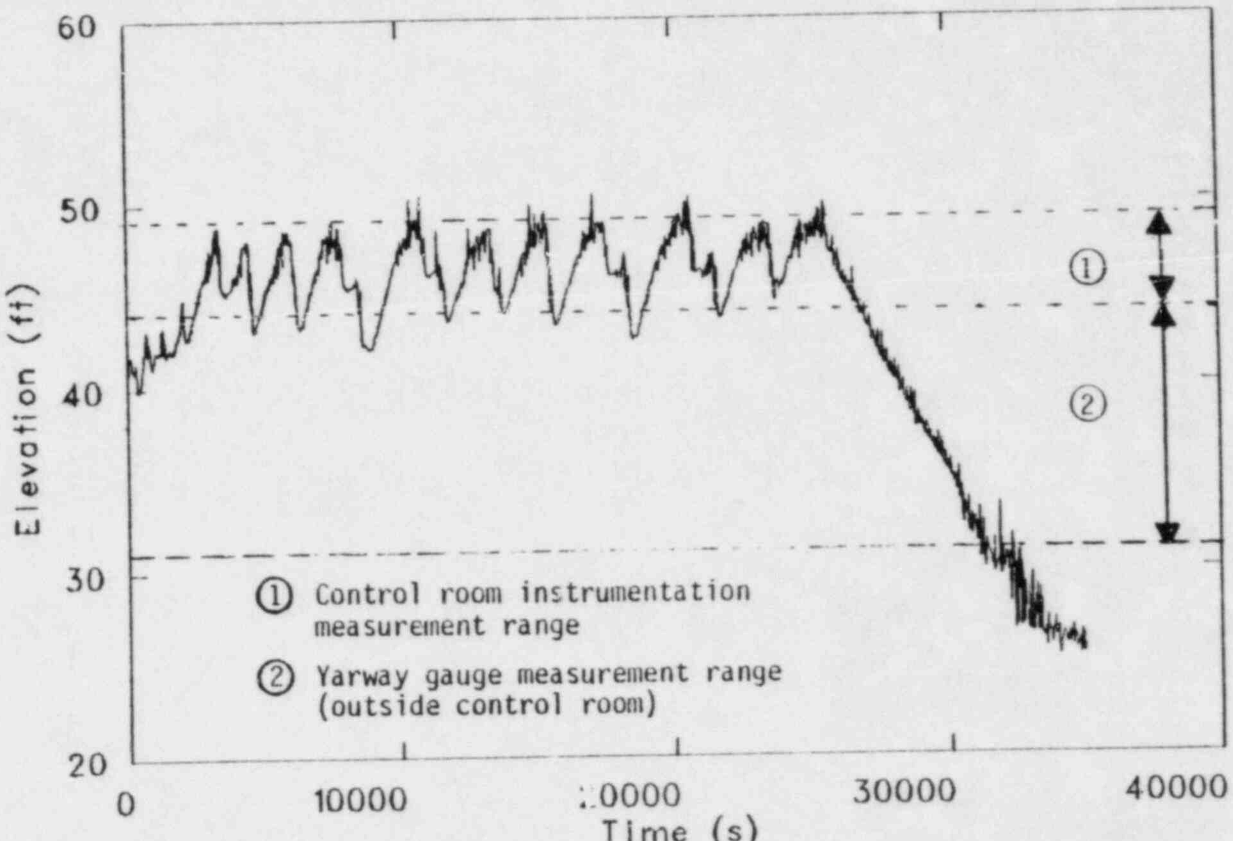


Figure 14. Downcomer water level (long term) – sequence v2.

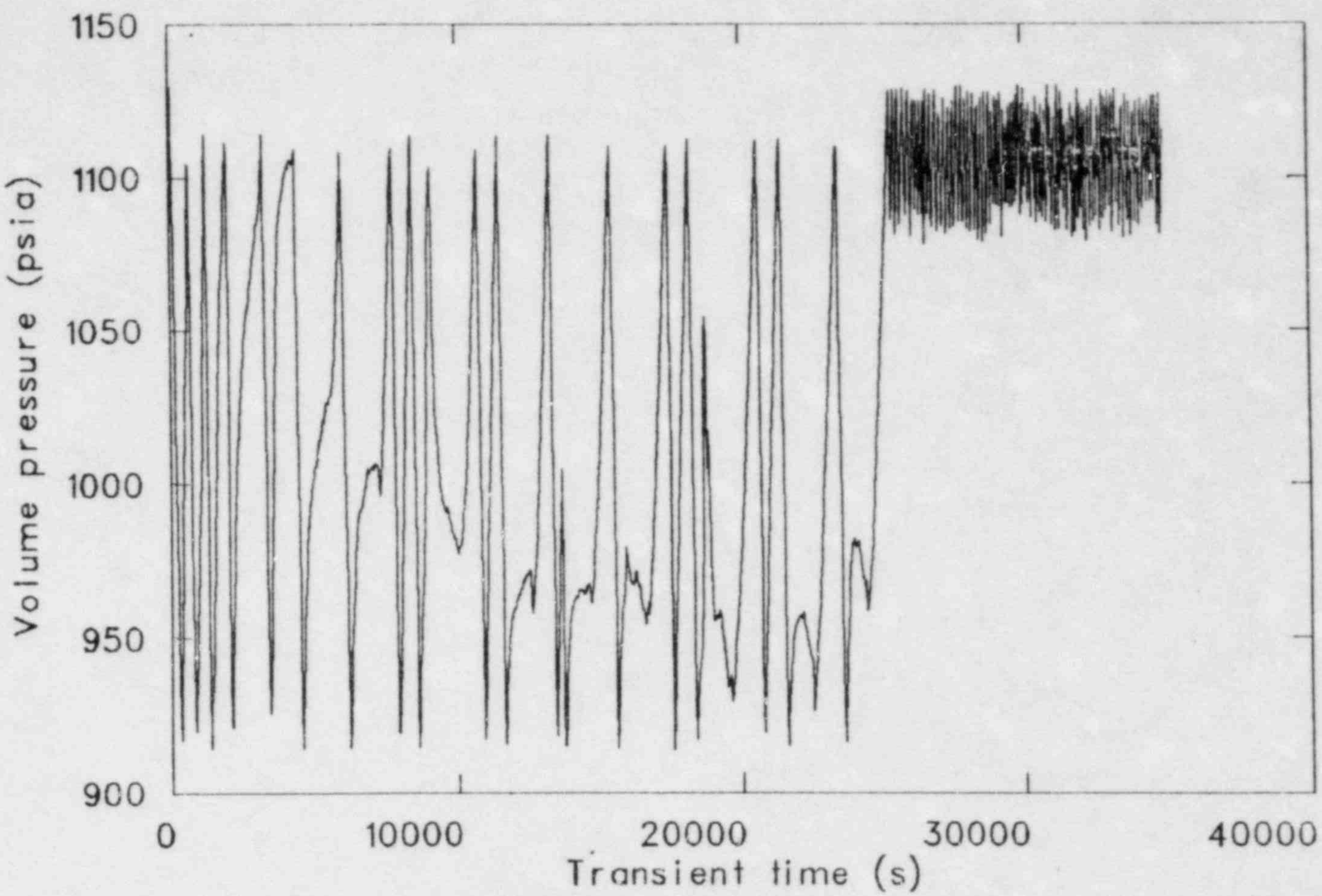


Figure 15. Vessel steam dome pressure - sequence v2.

The above actions (defined in Reference 3) were assumed to continue until complete loss of the station batteries occurred at seven hours. Thereafter, the SRVs began cycling at their automatic setpoints. The RCIC system was assumed to no longer be functional. The vessel water inventory boiled off until core heatup began at 34,000 s (see Appendix E). Core heatup could only be precluded by providing an alternate power source for the RCIC. However, such an action would only be a temporary solution. The long term solution would be to activate the residual heat removal (RHR) system.

4.5 The Station Blackout Transient with the HPCI System Available - Sequence V4

The calculation was based on the same set of assumptions as the first except that the HPCI system was assumed available. The calculation was conducted principally to establish a system behavior when the HPCI is available in the automatic mode.[a] The RCIC system is assumed unavailable.

System behavior was the same as in the first calculation until 295 s (see Figure 16) when the HPCI was initiated by a low-low level trip (see Figure 17). The system provided flow until 525 s when the system was tripped off. The HPCI was initiated a second time at 2385 s and pumped fluid to the vessel until 2645 s. Core uncovery did not occur (see Appendix F) during the seven hour period that the station batteries were available. However, following station battery failure, the core should uncover in the same time frame as Sequence V2.

4.6 The SORV Transient with RCIC Available - Sequence V8

Assumptions inherent in the Sequence V8 stuck open relief valve (SORV) transient are that following LOSP, only the RCIC system is available to replenish the vessel water inventory. Following the scram and the vessel

a. Automatic trip on: 39.67 ft elevation. Automatic trip off: 48.5 ft elevation. All elevations measured above the vessel zero.

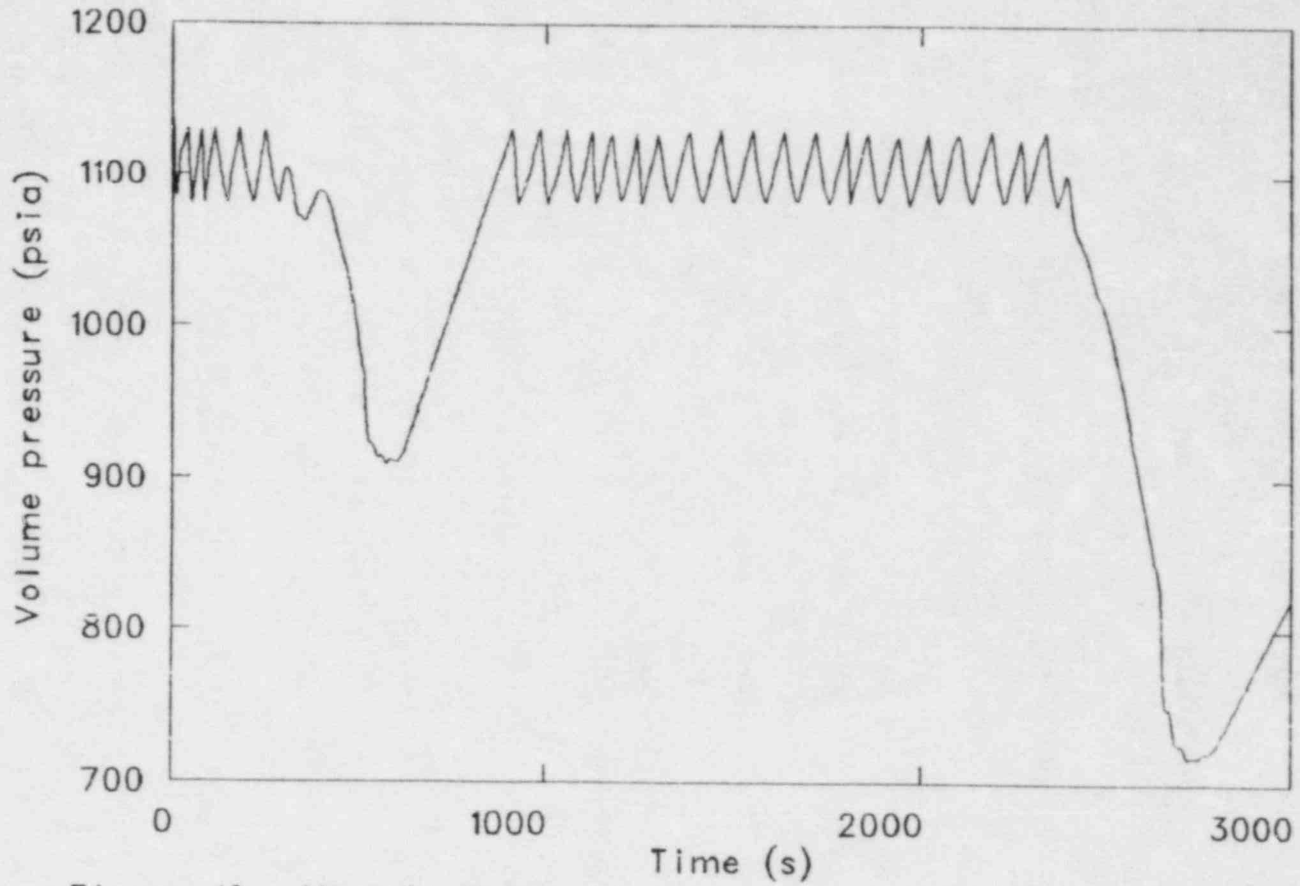


Figure 16. Vessel steam dome pressure – sequence v4

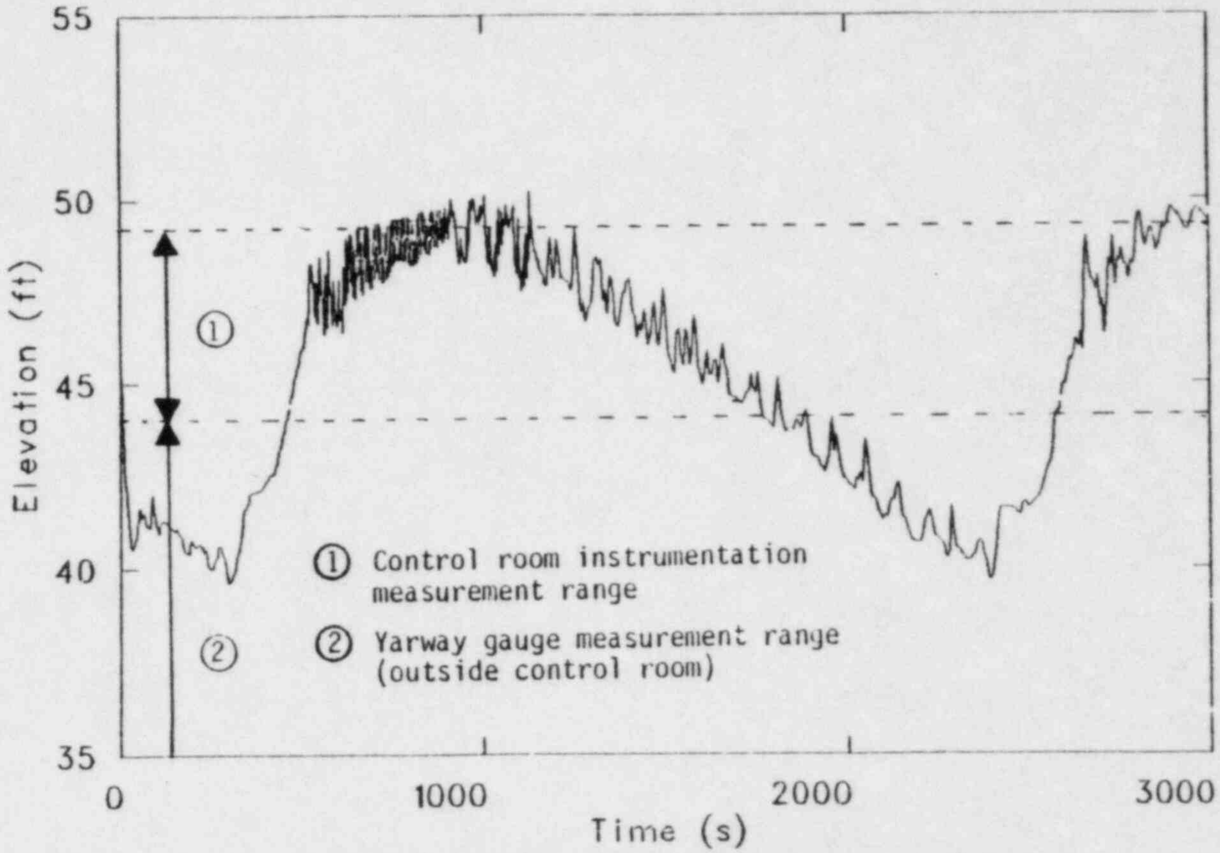


Figure 17. Downcomer water level — sequence v4

pressurization (see Figure 18), which accompanies the turbine control valve closure, the safety relief valves (SRV) behave as expected until 22 s when a SRV fails to close.

The operator responds by manual initiation of the RCIC system 90 s after the start of the transient. Thereafter, the operator leaves the RCIC on as needed to maintain the vessel downcomer water level between 44.8 ft and 48.2 ft (short term: Figure 19, long term: Figure 20), i.e., the indicated water level that is available to the operator via the control room instrumentation.

The vessel continually depressurizes as the transient proceeds until 9765 s when the RCIC system is shutoff (due to the downcomer water level reaching the 48.2 ft elevation). Thereafter the vessel pressure alternately increases and then decreases slowly depending on whether the RCIC is off or on, respectively. The RCIC fails at 25,200 s with the loss of the DC battery and the vessel inventory boiloff begins (see Appendix G).

The RELAP5 calculation was only conducted until 28000 s since the code mass error became excessive as the boiloff proceeded. A hand calculation shows that the core collapsed water level will reach the top of the core at 36000 s.

4.7 The Controlled Depressurization - Sequence V1

The controlled depressurization is discussed by ORNL (see Reference 3) as a means of reducing the heat load from the vessel to the drywell during a prolonged station blackout. The transient can be divided into four distinct phases:

1. Vessel inventory maintenance at operating pressure--the vessel state is maintained at the operating pressure (see Subsection 4.7.1),

2. The controlled depressurization--the operator depressurizes the vessel from the nominal 1000 psia level to the 90/200 psia level (see Subsection 4.7.2),

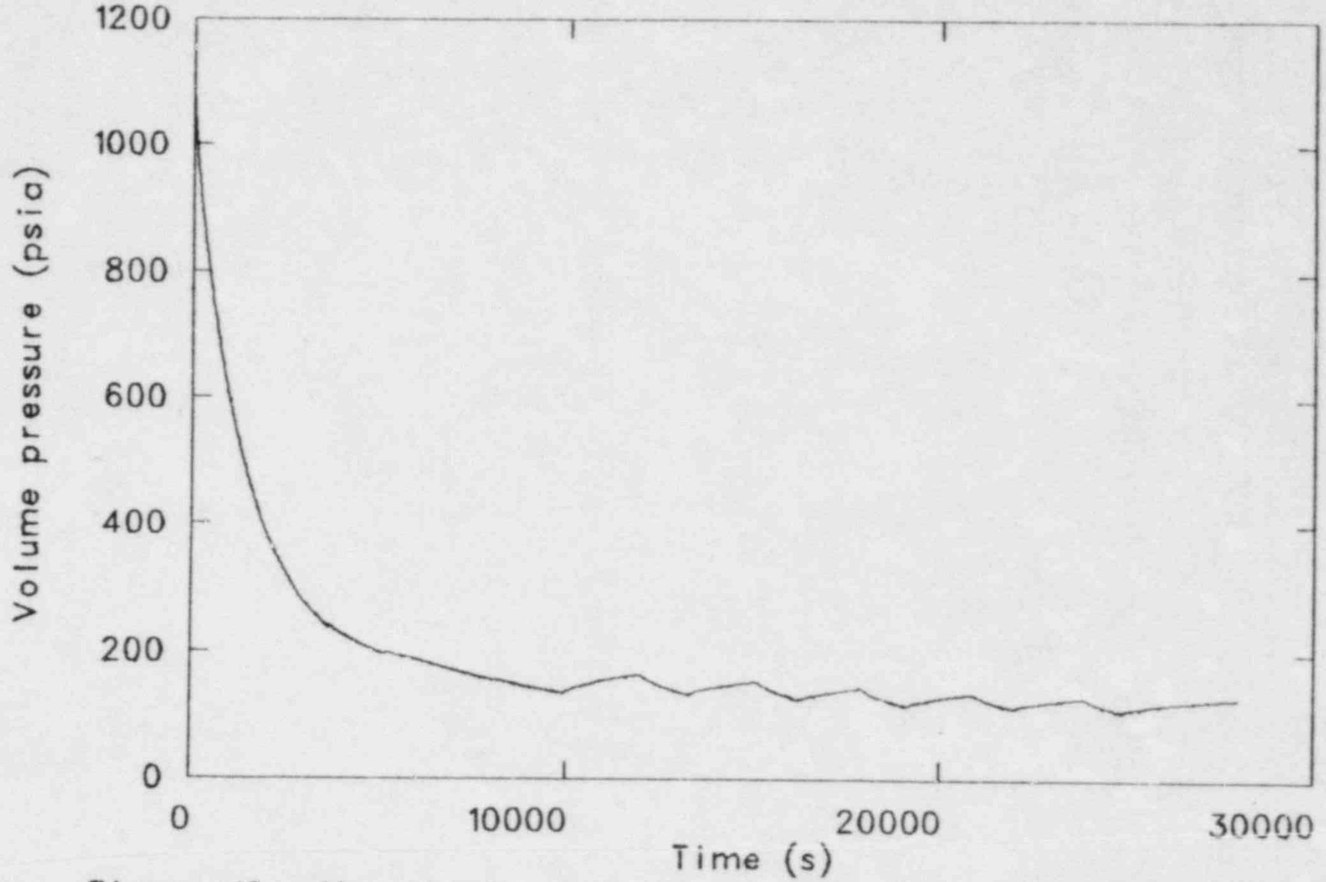


Figure 18. Vessel steam dome pressure - sequence v8.

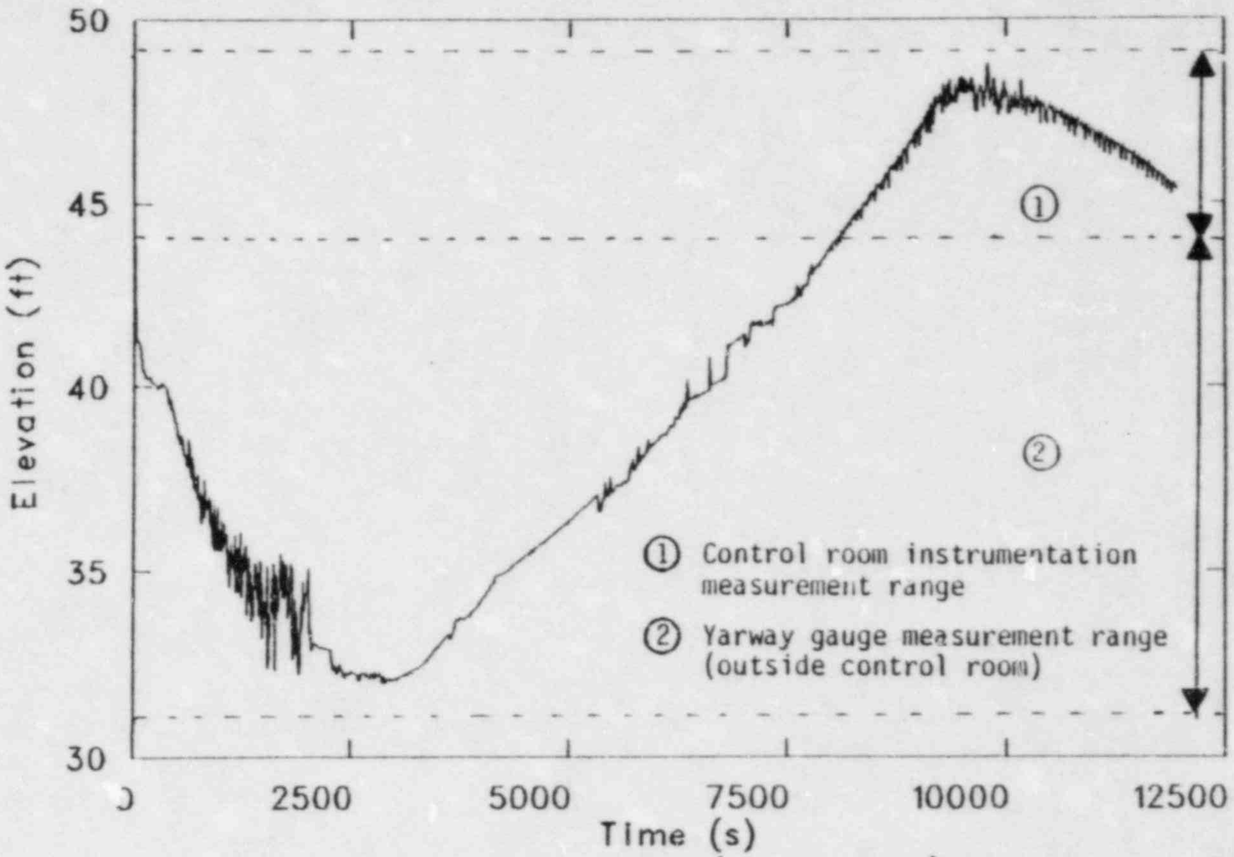


Figure 19. Downcomer water level (short term) - sequence v8.

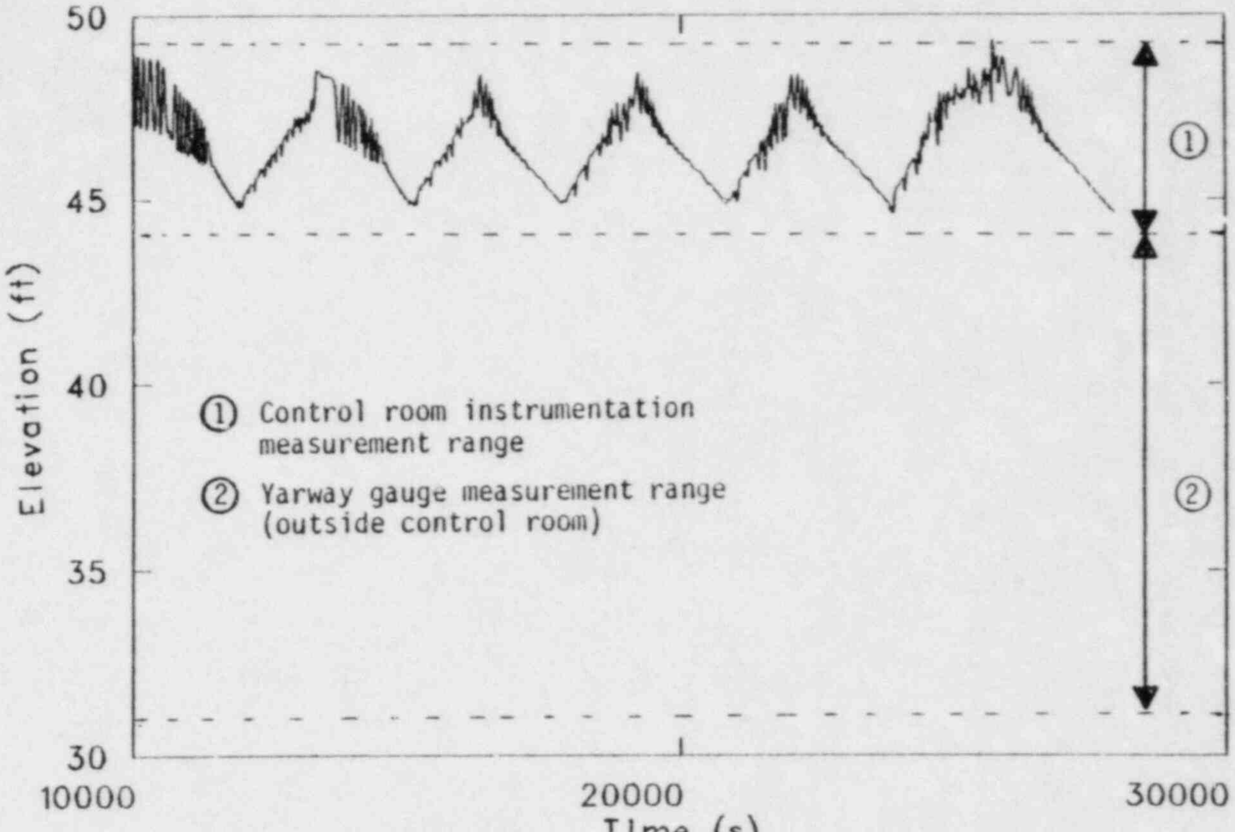


Figure 20. Downcomer water level (long term) - sequence v8.

3. Vessel inventory maintenance at low pressure--the vessel state is maintained at a low pressure level to minimize the heat transfer to the drywell (see Subsection 4.7.3), and

4. Vessel inventory boiloff--the final phase begins at 25,200 s with the DC battery failure. The operator does not have control over the equipment following the battery failure. Thus, the system would repressurize and boiloff over a 15,00C+ s time period (see Reference 3). The core would uncover at a time in excess of 40,000 s.

Phases A, B, and C are discussed in more detail in the remaining subsections.

4.7.1 Vessel Inventory Maintenance at Operating Pressure

The first phase is approximately one hour long and is characterized by the operator initiating the RCIC system 90 s after off-site power is lost. (The HPCI system was assumed inoperative.) For the remainder of the first hour, the operator would maintain the vessel water level between 44.8 and 48.2 ft and the steam dome vessel pressure between 900 and 1100 psia by activating selected SRVs. The vessel thermal- hydraulic behavior during the first hour is identical to that shown in Sequence V2.

4.7.2 The Controlled Depressurization

The second phase of the transient consists of a vessel depressurization which would occur over a 6400 to 8700 s time period. The depressurization time span is determined by the vessel conditions at 3600 s and the Technical Specification Limit (TSL) which does not allow the operator to decrease the vessel temperature at a rate greater than 100°F/hr. Even though the vessel depressurization limit is set by the above TSL, the operator can choose any number of ways to lower the vessel pressure, e.g., use one or two SRVs, the RCIC/HPCI turbines or combinations of these options. Consequently, the depressurization phase of the transient was not calculated using RELAP5. (Reference 3 gives a sample of such a calculation.)

4.7.3 Vessel Inventory Maintenance at Low Pressures

The third phase of the transient consists of the vessel conditions being maintained at a low pressure level. Thus, low vessel temperatures minimize the heat transfer to the drywell (which doesn't have any operable cooling equipment). In the direction of maintaining the vessel temperature at a low level, the RELAP5 calculations have shown several ways in which the operator can accomplish this objective (see Table 4). Important considerations are:

a. Should the operator always follow procedures designed to open one SRV at a time, to decrease the probability of having a SORV?

b. Should the operator follow procedures designed to open and shut SRVs based on a pre-established pressure band e.g., 90 to 140 psia or should the length of time that steam is exhausted through a particular SRV to the pressure suppression pool (PSP) also be considered? A prime consideration is whether localized heatup in the PSP is sufficient to govern the discharge time through a particular SRV. Also, if the discharge time is important, should the pressure limits be defined as the governing parameter for the operator or should the discharge time itself be the operator controlled parameter.

These options can only be judged based on the localized PSP heatup calculations currently underway at the ORNL and an intimate knowledge of operator training procedures. However, given that as a general rule the number of activated SRVs should be minimized, localized PSP heatup should be minimized, the operators could be provided with guidelines which list recommended discharge times into the PSP from a given SRV (rather than just pressure limits) and the vessel pressure should be reduced to the lowest possible level to minimize the drywell heatup; a set of options can be listed (see Table 4).

TABLE 4. STATION BLACKOUT OPERATOR ACTIONS-VESSEL WATER INVENTORY MAINTENANCE AT LOW PRESSURE

Option	Action	Advantage	Disadvantage
A	Maintain vessel pressure between 90 and 140 psia using two safety-relief valves.	Low drywell heat load. Valves discharge for discrete time periods-minimize localized PSP heatup.	Two valves are used. the probability of valve failure is higher. Operator may have to time SRV discharge period to distribute steam to PSP.
B	Maintain vessel pressure below 170 psia using one safety-relief valve.	Low drywell heat load. Only one SRV is used: less probability of valve failure.	Operator must time discharge period to PSP to minimize localized heatup.
C	Maintain vessel pressure below a value established by one SRV operation at discrete time periods to minimize localized PSP heatup.	Valve discharges for discrete time periods-minimize localized PSP heatup. Only one SRV is used.	Higher drywell heat load than Options A and B.

The vessel conditions between 7000 and 8000 s of the Sequence V8 calculation are very similar to the conditions that would exist after a controlled depressurization to 160-170 psia (although the water level is somewhat low). Thus, a controlled depressuriza- tion calculation, assuming the operator would use two SRVs to maintain the vessel pressure beween 90 and 140 psia was begun using the vessel conditions of the Sequence V8 calculation at 7800 s (see Option A--Table 4). The calculation (see Appendix H) demonstrated that the operator could maintain the vessel pressure between 90 and 140 psia with 2100+/700+ s open/shut periods. The primary disadvantage of Option A is that two SRVs are used.

Option B (Table 4) is virtually the same as the SORV calculation (see Appendix G) from 9700 s until DC battery failure (25,200 s) if the operator allowed the vessel pressure to increase to 170 psia at isolated times in the transient. The virtues of Option B are that only one SRV is activated and the drywell heat load is low. However, the operator should meter the discharge times to the PSP from each SRV to prevent localized PSP heatup.

Option C represents the action which an operator would take if only one SRV was activated and the discharge periods to the PSP were defined (i.e. timed) to limit localized PSP heatup. The disadvantage of Option C is that a larger heating load is imposed on the contain- ment drywell since the the vessel pressure level is higher than in Options A and B.

Of the options listed in Table 4. Option B is recommended because only one SRV is used and the vessel temperature is kept at a minimum level. However, these options should be reviewed by TVA operations personnel to determine their feasibility and desirability considering the equipment characteristics and their operator training philosophy.

4.8 Summary of the Station Blackout Calculation

Inasmuch as the station blackout calculations using the RELAP5 Browns Ferry model were conducted to provide boundary conditions for the analyses conducted at the Oak Ridge National Laboratory (Reference 3), the results of special interest are summarized. Finally, the equipment needs and actions which are necessary to keep the plant from catastrophic failure are summarized.

4.8.1 Key Events and Quantities

Key events and quantities calculated during the station blackout analyses are listed in Table 5. The table includes:

1. The time at which the core collapsed water level reached the top of the heated core i.e., t_{CWL}

2. The time at which the uppermost core volume began to heatup or reached a void fraction of 1 i.e., t_α

3. Total condensate storage tank mass delivered to the reactor vessel i.e., M_{CST}

4. The total reactor vessel mass delivered to the pressure suppression pool (PSP) by t_α i.e., M_{PSP}

5. The total reactor vessel energy delivered to the PSP by t_α i.e., Q_{PSP}.

4.8.2 Plant Equipment Required to Prevent Catastrophic Failure

As mentioned in subsections 4.2 through 4.7, the HPCI and RCIC systems are sufficient (Sequences V2 and V4) to prevent core uncovery given that their water source is not too hot to pump or that inventory is available in the CST. These systems will have power for 7 h (Reference 3, Appendix G). However if the HPCI and RCIC are not available, core uncovery will occur as indicated on Table 5 for Sequences V6 and V12.

TABLE 5. KEY EVENTS AND QUANTITIES

Sequence[a]	Core Uncovery: Time(s)[b] t_{CWL}	Core Uncovery: t_{α}[c]	CST Mass[d] To Vessel M_{CST} (lbm)	Mass to[e] PSP M_{PSP} M_{PSP} (lbm)	Energy to[f] PSP Q_{PSP} (BTU)
V6	1630.0	2300.0	0.0	245,000	2.90×10^8
V12	69.0	1600.0	0.0	260,000	3.1×10^8
V2	31700	34000[g]	1.11×10^6	1.21×10^6	12.6×10^8
V4	31700	34000[g]	1.11×10^6[h]	1.21×10^6[h]	12.6×10^8[h]
V8	36000	37840	1.28×10^6	1.48×10^6	17.8×10^8
V1	Not calculated	40000+	1.28×10^6[i]	See Ref. 3	See Ref. 3

a. See Figure 1.

b. Time when the water level reaches the core top i.e. elevation = 30.2 ft above the vessel bottom.

c. Time when the uppermost volume void fraction equals 1 or core heatup begins.

d. Mass from the condensate storage tank (CST) to the reactor vessel by t_{α}.

e. Mass from the vessel to the pressure suppression pool (PSP) by t_{α}.

f. Energy from the vessel to the PSP by t_{α}.

g. Intermittent heatup occurs after 33200 s.

h. Same as Sequence V2.

i. Same as Sequence V8.

Sequences V8 and V1 deserve mention in that the amount of water inventory required to maintain a well cooled core for 7 h was calculated to be 1.28×10^6 lbm or 156,000 gal. Thus if only the 135,000 gal guaranteed reserve were available in the CST, the HPCI/RCIC systems would have to switch suction to the PSP between 5.4 and 5.9 h. The question as to whether the HPCI or the RCIC pump net positive suction head limit would be the limiting factor if the pump suction were transferred to the PSP has not been addressed. However, since the CST capacity is 375,000 gal., such a pump suction transfer probably would not occur.

In general, none of the station blackout sequences can recover on the long term (after 7 hr) unless the plant residual heat removal (RHR) system is activated such that the containment can be cooled and water inventory pumped into the vessel.

5. CONCLUSIONS AND OBSERVATIONS

Several significant conclusions and observations, based on the analyses described in Section 4 are:

a. A station blackout transient is improbable. Equipment unavailabilities calculated in the Reference 1 analyses give the station blackout event frequency to be 5.7×10^{-4} events/year at most.

b. The emergency equipment cooling water (EECW) system is the most important contributor to a station blackout sequence. As listed in Table 1, the event frequency of a station blackout scenario could be decreased by a factor of 10^{-3} if the EECW system were eliminated as a contributor to the failure of the diesel generators.

c. The vessel water inventory equipment available to the operator during a station blackout is sufficient to maintain the vessel inventory over the time frame when the station battery is available, even when a SORV is present.

d. Ultimate shutdown of the plant can only be accomplished if AC power is restored together with the residual heat removal system.

e. RELAP5 can be used to model BWR long term transients. The code has completed 7.8 and 9.7 h transients at 4.9 and 4.4 h calculated times respectively. Thus RELAP5 is an ideal code for conducting prolonged transients due to it's fast running nature.

6. REFERENCES

1. S. E. Mays, J. P. Poloski, W. H. Sullivan, J. E. Trainer, R. C. Bertucio, and T. J. Leahy, Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant, NUREG/CR-2802, July 1982.

2. G. W. Burnette, D. W. Danielson and K. A. Nilsson BWR Blowdown Heat Transfer Program Task C4 Report Preliminary System Design Description of Two Loop Test Apparatus Revision 1 GEAP-13276-1, November 1973.

3. D. H. Cook, S. R. Greene, R. M. Harrington, S. A. Hodge and D. D. Yue, Station Blackout at Browns Ferry Unit One-Accident Sequence Analyses, NUREG/CRT-2182, November 1981.

4. J. A. Dearien ltr to R. E. Tiller "Categorization of BWR and PWR Plants for SASA Work (A6048C)", JAD-220-80, August 28, 1980.

5. Browns Ferry Nuclear Plant Final Safety Analysis Report.

6. R. R. Schultz personal communication with D. Caraher, Spring 1981.

7. P. W. Baranowsky ltr to E. Davidson, "SASA Station Blackout Analyses", June 19, 1981.

8. S. L. Forkner, D. L. Bell and E. N. Winkler, Calculation of a Generator Load Regection Transient in a 3293 MW BWR/4 with RETRAN-01-RET12B, July 1978.

9. R. R. Schultz, personal communication with T. Burton, September 11, 1981.

10. R. W. Griebe and J. W. McConnell, Boiling Water Reactor Full Length Emergency Core Cooling Heat Transfer (BWR-FLECHT) Tests Project Final Report on Atmospheric Pressure Stainless Steel Experiments, ANCR-1115, November 1973.

APPENDIX A

U.S. BWRs CATEGORIZED BY DESIGN SIMILARITIES

(Target Plants Designated)

TABLE A-1. U.S. BWR's GROUPED INTO MAJOR DECKS

Group	Plant Name	BWR Mark	Containment (1) (1) Mark	Vessel ID/No. Fuel Bundles	Rated MWt Power	Expected or Commercial (c) Date (2)	Comments/Target Plant (T)(3)
1	Browns Ferry 1	4	1	251/764	3293	C, 8/74	
	Browns Ferry 2	4	1	251/764	3293	C, 3/75	
	Browns Ferry 3	4	1	251/764	3293	C, 7/77	
	Peach Bottom 2	4	1	251/764	3293	C, 7/74	
	Peach Bottom 3	4	1	251/764	3293	C, 12/74	
	Fermi 2	4	1	251/764	3293	11/83	
	Hope Creek 1	4	1	251/764	3293	12/86	
	Hope Creek 2	4	1	251/764	3293	5/86	
	Limerick 1	4	2c	251/764	3293	4/85	
	Limerick 2	4	2c	251/764	3293	4/87	
	Susquehanna 1	4	2c	251/764	3323	5/83	
	Susquehanna 2	4	2c	251/764	3323	5/84	
2	La Salle 1	5	2c	251/764	3293	6/82	
	La Salle 2	5	2c	251/764	3293	10/83	
	Nine Mile	5	2c	251/764	3293	10/86	
	Point 2	5	2	251/764	3323	9/81	T
	(Hanford-2)						
3	Bailly N-1	5	2c	201/444	1931	?/84	T
4	Clinton 1	6	3	218/624	2894	8/83	T
	Clinton 2	6	3	218/624	2894	Indef.	
	River Bend 1	6	3	218/624	2894	4/84	Motor Driven
	River Bend 2	6	3	218/624	28984	Indef.	Feed Pumps
5	Allens Creek 1	6	3	238/732	3579	?/91	
	Black Fox 1	6	3	238/732	3579	7/91	Simulator
	Black Fox 2	6	3	238/732	3579	7/94	
	Grand Gulf 1	6	3	238/732	3579	12/82	T
	Grand Gulf 2	6	3	238/732	3579	12/82	
	Hartsville A1	6	3	238/732	3579	Indef.	
	Hartsville A2	6	3	238/732	3579	4/91	
	Hartsville B1	6	3	238/732	3579	4/92	
	Hartsville B2	6	3	238/732	3579	Indef.	
	Phipps Bend 1	6	3	238/732	3579	Indef.	
	Phipps Bend 2	6	3	238/732	3579	Indef.	
	Perry 1	6	3	238/732	3579	5/84	
	Perry 2	6	3	238/732	3579	5/88	

TABLE A-1. (continued)

Group	Plant Name	BWR Mark	Containment (1) (1) Mark	Vessel ID/No. Fuel Bundles	Rated MWt Power	Expected or Commercial (c) Date (2)	Comments/Target Plant (T)(3)
	Skagit 1	6	3	238/732	3579	9/91	
	Skagit 2	6	3	238/732	3579	9/93	
6	Brunswick 1	4	1c	218/560	2436	C, 3/71	
	Brunswick 2	4	1c	218/560	2436	C, 11/77	
	Cooper	4	1	218/458	2381	C, 7/74	T
	FitzPatrick	4	1	218/560	2436	C, 7/75	
	Hatch 1	4	1	218/560	2436	C, 12/75	
	Hatch 2	4	1	218/560	2436	C, 8/79	
	Shoreham	4	2c	218/560	2436	3/83	
	Zimmer	5	2c	218/560	2436	?/83	
7A	Dresden 2	31	1	251/724	2526	C, 8/70	(IC), T Simulator
	Dresden 3	3	1	251/724	2527	C, 10/71	
7b	Quad Cities 1	3	1	251/724	2511	C, 8/72	(RHR), T
	Quad Cities 2	3	1	251/724	2511	C, 10/72	
8	Millstone 1	3	1	224/580	2011	C, 12/70	
	Pilgrim 1	3	1	224/580	1912	C, 12/72	T
9	Monticello	3	1	205/484	1464	C, 7/71	
	Vermont Yankee	4	1	205/368	1593	C, 11/72	T
	Duane Arnold	4	1	183/368	1593	C, 5/74	
10	Oyster Creek	2	1	213/560	1930	C, 12/69	T No Jet Pumps
	Nine Mile Point 1	2	1	213/532	1583	C, 12/69	No Jet Pumps
11	"Others"						
11A	Dresden 1	1	Steel Sphere	146/448	630	C, 8/60	T No Jet Pumps
11B	Big Rock Point	1	Steel Cylinder	106/84	230	C, 12/62	No Jet Pumps
	Humbolt Bay	1	Steel Cylinder	120/184	240	C 8/63	No Jet Pumps
	LaCross BWR	Allis	?	?	?	C, 11/69	No Jet Pumps

(1) Containment Mark Number:
1 is drywell and free standing torus.
1 is drywell and concrete torus with steel liner.

TABLE A-1. (continued)

Group	Plant Name	BWR Mark	Containment (1) (1) Mark	Vessel ID/No. Fuel Bundles	Rated MWt Power	Expected or Commercial (c) Date (2)	Comments/Target Plant (T)(3)

2 is over/under with free standing steel pressure vessel.
2c is over/under steel liner surrounded by concrete.
3 is suppression pool type.

(2) Expected or Commercial Date:
A "C" notes commercial operation on the date shown.
A date without "C" indicates expected commercial operation date (Source: Nuclear News 1979 World List of Nuclear Power Plants).

(3) Target Plants:
A "T" indicates a plant will receive first attention in setting up decks (see letter text).

Sources and References

1. Nuclear News 1979 World List of Nuclear Power Plants.

2. GE/BWR experience list (by date of commercial operation), October 1973.

3. Various Personal Conversations.

APPENDIX B

THE INTERIM BROWNS FERRY MODELS

APPENDIX B

THE INTERIM BROWNS FERRY MODELS

A RELAP5 model of the Browns Ferry plant was constructed to meet the needs of the Severe Accident Sequence Analysis (SASA) Program. From the start, the objective was to construct a detailed thermal-hydraulic model capable of use with virtually any transient of interest in SASA. However, such a goal was unrealistic and in fact impossible on the current time schedule. Road blocks encountered very early in the model building process limited the amount of information. In fact, only a limited number of blueprints were available. These together with the Browns Ferry Final Safety Analysis Report (FSAR) and a RETRAN input listing were provided by TVA (see References B.1 and B.2). More detailed hardware layout blueprints and thermal-hydraulic specifications were unavailable since General Electric proved to be uncooperative. Thus, only an interim model could be constructed.

While all general information was provided to the Idaho National Engineering Laboratory (INEL) by TVA via the above information sources, there are at least three shortcomings inherent in constructing a RELAP5 deck using an intermediate source of information or model as a baseline:

1. Errors present in the baseline model can be carried into the new model by transposition.

2. Modeling philosophy and information needs consistent with the objectives of the TVA baseline model may not be consistent with the objectives of the RELAP5 model development effort.

3. Earlier efforts to produce a Brown Ferry thermal-hydraulic model using RETRAN did not have the more sophisticated modeling capability available in codes like RELAP5 (e.g., detailed core nodalization including counter current flow limiting at the bundle entrance and exit). Thus, information useful for constructing a RELAP5 model was not present in earlier

thermal-hydraulic models simply because the earlier generation codes do not have the ability to calculate many important phenomena to the same degree of complexity.

Thus, even though the general information needs have been met by TVA, further needs have been defined (Reference B.3) which should be provided by the manufacturers (General Electric Co.). These needs are required before the model will become final.

Two basic variations of the Browns Ferry model were used to conduct the station blackout calculations (Reference B.4). Of the four complete transients, three were analyzed using a thermal-hydraulic model with two recirculation loops. The stuck open relief valve (SORV) transient, assuming the reactor core isolation cooling (RCIC) system was available, was analyzed using a simplified model with only one recirculation loop.

The following three subsections describe the assumptions and basis for (1) the hydrodynamic components (2) the heat slabs and (3) the RELAP5 code options used to form the Browns Ferry Model. Subsection B.4 discusses the assumptions inherent to the one recirculation loop model and Subsection B.5 is devoted to the limitations of the interim Browns Ferry models.

B.1 Hydrodynamic Nodalization of the RELAP5 Browns Ferry Model

The manner in which RELAP5 is used to hydrodynamically model a system is similar to RELAP4 and RETRAN in many respects. However, there are important differences which often impact the model nodalization directly.

Unlike RELAP4 and RETRAN, RELAP5 has pipe volumes which can be subdivided to form subvolumes using a common volume number. Thus, the axial pressure profile can be examined at distinct locations within a volume to investigate transient behavior of interest. However, whenever a volume is attached to three or more stream tubes, a RELAP5 branch volume must be used. Such requirements often define a system component type.

B.1.1 Hydrodynamic Nodalization: The Recirculation Lines

The Browns Ferry recirculation lines were nodalized (see Figure B.1 and Table B.1) to represent each of the two recirculation loops present on the plant. Each model loop represents the plant piping from the vessel to the pump suction side (Volumes 200 through 215--NOTE: The volume geometry in the other recirculation loop is identical except the volumes are numbered in the three hundreds), each recirculation pump (Volume 220), the discharge piping from the pump to the recirculation manifold (Volumes 230 through 250), the manifold (Volume 260) and the jet pump drive line risers (Volume 280).

The suction side of the recirculation lines were modeled to exit the vessel at the 13.46 ft[a] elevation by aligning the pipe centerline with the top of Volume 600. The recirculation suction line was also nodalized to include the reactor water cleanup (RWCU) system at the 1 ft elevation and the suction side isolation valve at the -27.1 ft elevation.

The recirculation pump was modeled using the physical dimensions identified by TVA (see Reference B.2). The homologous curves and two phase degradation curves were taken directly from a Hope Creek Model (see Reference B.5). The Hope Creek data was used for the Browns Ferry analysis since (1) a full set of pump data is not currently available for the Browns Ferry recirculation pumps and (2) Hope Creek is a sister plant of Browns Ferry.

The recirculation pump discharge line (Volumes 230 through 250) was nodalized to include the downstream isolation valve, the isolation valve bypass and the residual heat removal (RHR) system [low pressure coolant injection (LPCI) mode] inlet location. The isolation valve bypass, and the RHR system were not used in the station blackout analysis (Reference B.4), consequently the valves remained closed.

---

a. All elevations are referenced to the vessel inside bottom.

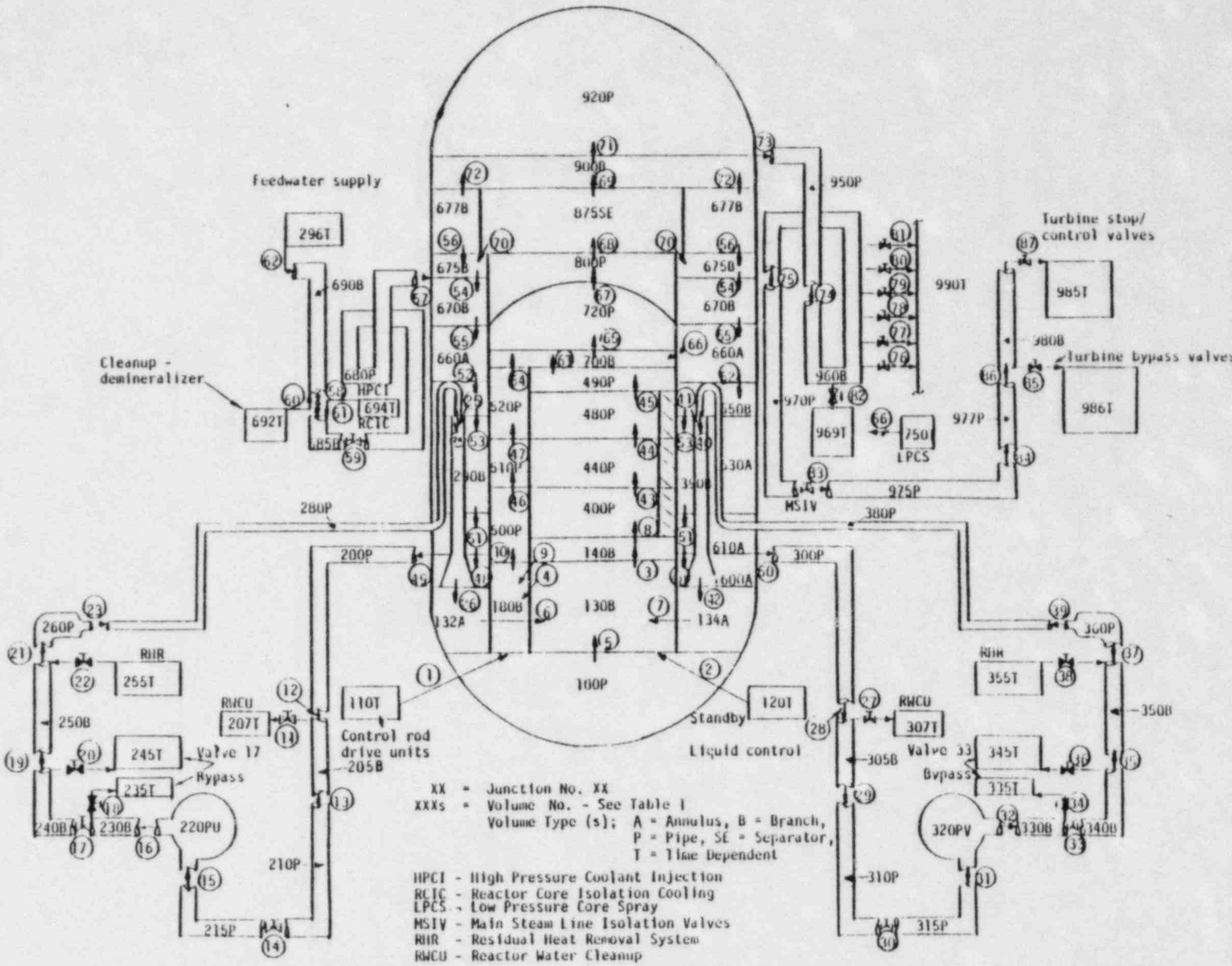


Figure B1. The Interim Browns Ferry RELAP5 model.

Because the recirculation flow is distributed from the manifold (Volume 260) to feed five jet pump risers, Volume 260 was subdivided into three subvolumes. The first represents the manifold feed line from the RHR inlet to the manifold tee. The second subvolume was sized to give the correct velocity given that 40 percent of the recirculation flow is present. The third subvolume was sized to represent the correct pressure (and velocity) given that 20 percent of the recirculation flow is present.

The jet pump risers i.e., Volume 280 represents the recirculation line from the manifold to the jet pump drive line nozzle. The riser was sized to represent five jet pump risers and was subdivided into 9 subvolumes.

B.1.2 Hydrodynamic Nodalization: The Jet Pumps

The jet pump was treated as a special component in the Browns Ferry model. Such treatment was forced since the momentum mixing necessary for the drive jet to transfer energy to the pump suction flow was not present in the RELAP5 code. Thus, a code update was added (see Figure B.2) to simulate the presence of momentum mixing.

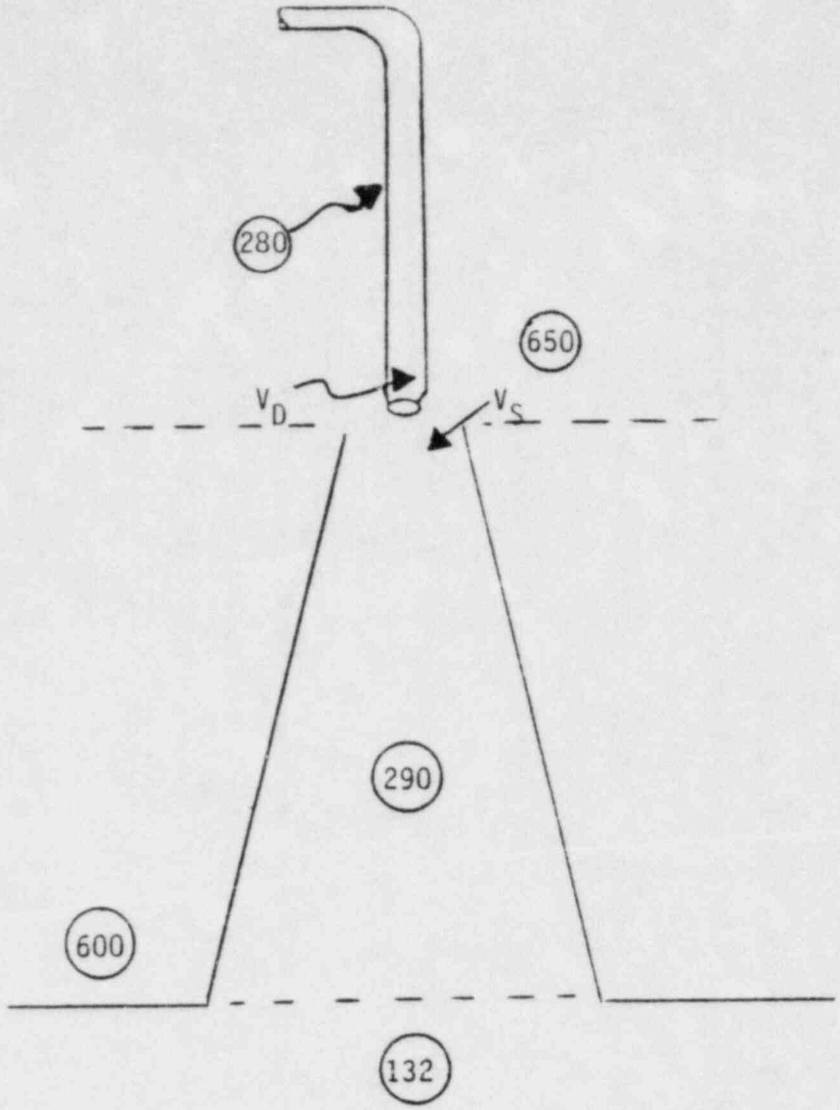
The code update (Reference B.6) simulated the presence of momentum mixing by adding a pressure increase (ΔP) to the flow between the downcomer and jet pump:

$$\Delta P = K \frac{\rho \, V_D^2}{2 \, g_c}$$

where

K = input factor to simulate momentum mixing

g_c = proportionality constant



$$\Delta P_{JP} = K \frac{\rho V_D^2}{2}$$

$$\Delta P_{290/650} = P_{650} - P_{290} + \Delta P_{JP}$$

Where :

K = const if:

$\alpha_{290} \leq 0.2$

$V_D \geq 1$ m/s

K = 0 if :

$\alpha > 0.2$

$V_D < 1$ m/s

V_D = drive velocity

V_S = suction velocity

P_{xxx} = pressure in volume xxx

ΔP_{JP} = differential pressure addition (momentum mixing term)

K = input constant

ρ = drive fluid density

O = component volume number

α = volume void fraction

Figure B2. The Interim Browns Ferry RELAP5 Jet Pump Model.

ρ = suction flow density

V_D = drive line flow velocity.

The code update was constructed to be included as a contributor in the calculation as long as V_D is greater than 1 m/s and the jet pump void fraction is greater than 0.2. Thereafter, K is set to zero and only the geometry form losses and frictional pressure loss relationships govern the natural circulation through the jet pumps.

B.1.3 Hydrodynamic Nodalization: The Vessel

The Browns Ferry vessel was nodalized as shown in Figure B.1. With the exception of the steam dome, separator, upper plenum and the downcomer, the vessel was nodalized as in Reference B.1.

The primary sources of information for the downcomer nodalization were the Figures 4.1.1 and 3.3.5 in Reference B.1. The downcomer was nodalized in more detail than given in Reference B.2 to account for the sudden changes in downcomer water level resulting from:

1. Abrupt changes in the downcomer flow area.

2. Flashing water in downcomer subvolumes during transients of interest.

Thus the downcomer was nodalized as described in Table B.1. In conjunction with the downcomer nodalization, the water level was tracked by summing the liquid elevation in each downcomer subvolume:

$$EL = \Sigma_i (EL_i * VOIDF_i)$$

where

EL = total downcomer water elevation (ft)

EL_i = geometrical elevation of downcomer volume i (ft)

$VOIDF_i$ = liquid fraction in downcomer volume i

i = volume 600, 610, 630, 650, 660, 670, 675 or 677

The water level in the core shroud was tracked in a similar way.

The vessel separator was modeled by including all the volume contained within the vane assemblies in model volume 875. The volume logic was updated to permit only liquid to leave the lower separator exit unless the volume became completely steam filled. However, flow into the separator could be any mixture of steam and liquid at all junctions.

The steam dome (Volumes 900 and 920) and upper plenum (Volumes 700 and 720) nodalization was defined in part by the location of the steam lines and the low pressure core spray sparger (LPCS) injection plane respectively. Provisions for inclusion of the LPCS were factored into the model even though the low pressure emergency core cooling systems (ECCS) were never used in the station blackout analyses.

The guide tube volume (component 180) was sized to model the control rods when fully inserted in the core shroud. Flow paths into the guide tubes were modeled to simulate the control rod drive flow (from component 110), leakage from the lower plenum (from Volume 130) and leakage from the core inlet pieces (from Volume 140). Flow areas for the latter leakage paths were sized in conjunction with the core inlet flow area to pass approximately ten percent of the total jet pump discharge flow.

Provisions were made to model the standby liquid control system (SLCS) by including Volume 120 in the model. The SLCS was modeled to inject into the bottom of Volume 130.

B.1.4 Hydrodynamic Nodalization: The Feedwater and Steam Lines

The feedwater and steam lines were modeled to duplicate the work reported in Reference B.2 when possible. Exceptions were only made to accomodate the model structure required by RELAP5.

The feedwater line was modeled to include all the volume from the feedwater heaters to the vessel feedwater sparger. The two feedwater lines and spargers were represented by three component volumes (numbers 690, 685 and 680). Thermodynamic conditions of the feedwater was set by time dependent Volume 696. The reactor core isolation cooling (RCIC) system and/or the high pressure coolant injection (HPCI) system was modeled by time dependent Volume 694 which fed liquid water into Volume 685 downstream of a check volume (located between Volumes 690 and 685). An additional check valve was located between Volumes 685 and 680 to represent the inboard containment isolation valve. The feedwater line was plumbed to the upper end of Volume 670 in the vessel downcomer. The cleanup demineralizer flow was modeled using time dependent Volume 692 which fed fluid into the upstream end of Volume 685.

The steam lines were modeled with a set of component volumes as shown in Figure B.1. The lines were modeled from the vessel to the turbine control valves. The main steam line isolation valves (MSIVs) were located between Volumes 970 and 975. The thirteen safety-relief valves (SRV) were simulated by junctions 76 through 81 (see Table B.2) with setpoints at one percent above the FSAR listed values. The higher setpoints were used to examine the maximum pressure and power spike that would occur following a loss of off-site power (LOSP) given that the SRV setpoint uncertainty is one percent. Both the SRVs and the RCIC/HPCI steam turbine lines were plumbed to Volume 960.

The turbine bypass valves were modeled by sizing valve 85 (see Figure B.1) to pass 3.99×10^6 lbm/hr at an upstream pressure of 950 psig (Reference B.7). The bypass flow was nodalized to exit the component Volume 980.

B.2 The Heat Structure Nodalization of the RELAP5 Browns Ferry Model

The heat structures in the RELAP5 Browns Ferry model were taken directly from Reference B.1. The RELAP5 input format is such that heat structures can be described as a cylindrical or rectangular geometry. In addition the slab can be programmed to interface with a thermal-hydraulic volume on both the left and right side.

The heat structures were distributed to model the presence of the vessel wall, the core shroud, the core, the upper plenum, jet pumps and the recirculation loop pipes. Heat structures representing the vessel wall and the recirculation pipes were given an adiabatic boundary condition. The core channel pipes were given an adiabatic boundary. The core channel walls were also modeled as adiabatic heat structures. Attempts to model the core channel slabs as two sided bodies detracted from the steady-state initialization.

The heat structures representing the vessel internals have a total mass of 4.88×10^5 lbm. The vessel heat structures have 1.25×10^6 lbm total mass and the core fuel weight is 3.62×10^5 lbm. Midway through the station blockout analyses, the heat slab 16702000 was found to be misdimensioned. The slab width was a factor of 21.7 too large i.e., width equaled 1.02 ft instead of 0.047 ft. This error resulted in excessive vessel metal sensible energy. Thus whenever the system depressurized, the slab released too much energy to the vessel inventory. Conversely, when the vessel was repressurizing, the erroneous heat slab required more energy than the real system to reach the same thermodynamic condition. Thus the presence of the erroneous slab meant that the model's time response lagged the real system behavior.

For the purposes of calculating the required boundary conditions for the ORNL back end analysis, the heat slab error is almost unnoticeable. Given that the total energy available in the vessel is the sum of the fuel decay heat, the vessel fluid sensible energy and the vessel metal sensible energy; the heat slab error will cause an uncorrected model to contribute approximately 6 percent too much energy for a 7 hour transient with a system depressurization from 1000 to 100 psi. However, the RELAP5 decay heat model was found to contribute 7 to 8 percent less energy than the correct value (see Subsection 3.1, Reference B.4). Thus the two errors are self compensating.

B.3 RELAP5 Code Options

The RELAP5 code options can be divided into three general groups: (a) the main program control, (b) component options and (c) the power assumptions. The options discussed herein are those used for the final calculation set. However, the options were somewhat different at the start and when significant are noted as changed parameters. All the code input are stored under configuration Control Number F00966.

B.3.1 Main Program Control

All calculations were conducted in British units since the input used from Reference B.2 was British and since the plant personnel are not accustomed to metric. It should be noted that RELAP5 conducts all internal calculations in metric such that when the British option is used, some output appears in British and some in metric. Further information on this option is available in the RELAP5 description manual (Reference B.8).

The minimum time step used throughout was 1.E-7 s. The maximum time step was variable, but usually chosen to allow the code to select the appropriate maximum. The time step control was set at 00002 such that the heat structure time step was the same as the hydrodynamic time step.

B.3.2 Component Options

The component options can be divided into two areas based on whether the component was a volume type e.g., pipe, annulus or branch or a junction type e.g., a single junction or a valve. In both cases the recommended options were used as a rule.

The volume input was programmed to use the code wall friction and nonequilibrium calculations. Further, the initial conditions were input in data sets of pressure, internal energy and static quality.

The junctions were all input to calculate choking with the full inertial treatment. In general the junctions were treated as smooth and forward/reverse form loss coefficients input. The form loss coefficients were either calculated based on the known geometry, taken directly from Reference B.2, input based on a needed component pressure drop e.g., the separator (Volume 875) and the jet pumps (Volumes 290 and 390) or taken from an existing BWR/6 model (Reference B.9). The fluid phases were analyzed with two distinct velocities at all junctions except the separator (Volume 875 junction 68) inlet junction which was forced to have no phase slippage.

B.3.3 Power Assumptions

The core was nodalized and the power/reactivity characteristics were taken directly from Reference B.2. In general, the core was power weighted toward the bottom with 39, 38 and 23 percent of the heat being generated in the lower third, the middle third and the upper third respectively. The steady-state power was 3293 MWt (100 percent rated power for Browns Ferry). Point kinetics together with the RELAP5 fission product decay including actinides were used in all calculations. The scram reactivity table was constructed to give a full control rod insertion by 3.9 s. The fission product and actinide yield factors were set at 1.0. Default input was used for delayed neutron, fission product decay and the actinide decay constants.

B.4 Simplified Interim Browns Ferry RELAP5 Model

The basis for simplifying the Browns Ferry model was to reduce the two recirculation loops to one. Thus the number of volumes, junctions and heat structures were minimized. In such a way the total number of volumes were reduced from 115 to 66, the total number of junctions from 121 to 70 and the total number of heat structures from 56 to 33 (see Figure B.3).

The manner in which the recirculation lines were simplified was to not only model just one line, but also to reduce the number of subvolumes in each component volume. In addition the pump was nodalized to have the same rated head and velocity as the two loop model but double the flow, torque and inertia. The homologous curves and the two phase degradation curves were not changed. The pump geometrical input was doubled.

The one recirculation loop model is shown in Figure B.3. Note that the single recirculation loop does not have the isolation valves or the bypass valves (junctions 14, 17, 18 or 20) that are included in the more detailed nodalization. The volume descriptions for the simplified model conform to that described in Table B.1.

B.5 Limitations of the Interim Browns Ferry RELAP5 Models

As discussed in the introductory paragraphs of Appendix B, the Browns Ferry model construction task was first approached as an opportunity to assemble a detailed model usable for any transient. However important information were missing. Moreover, the information provided were not sufficient to allow a model to be constructed that matched the Nuclear Regulatory Commission's (NRCs) quality assurance (QA) standards. Specifically, much of the information was out of date and second-hand.

To meet the NRCs QA standards and to expand the usefulness of the model, the following key items are needed.

B.5.1 Detailed Core Information

Core information sufficient to allow detailed modeling was not available. The missing information and impact of not having it is listed:

1. Core inlet and channel inlet leakage paths geometry--these leakage paths have been shown to have an important contribution during core reflood following a large loss-of-coolant accident (LOCA) when the upper tie plate is countercurrent-flow-limited (CCFL)--see Reference B.10.

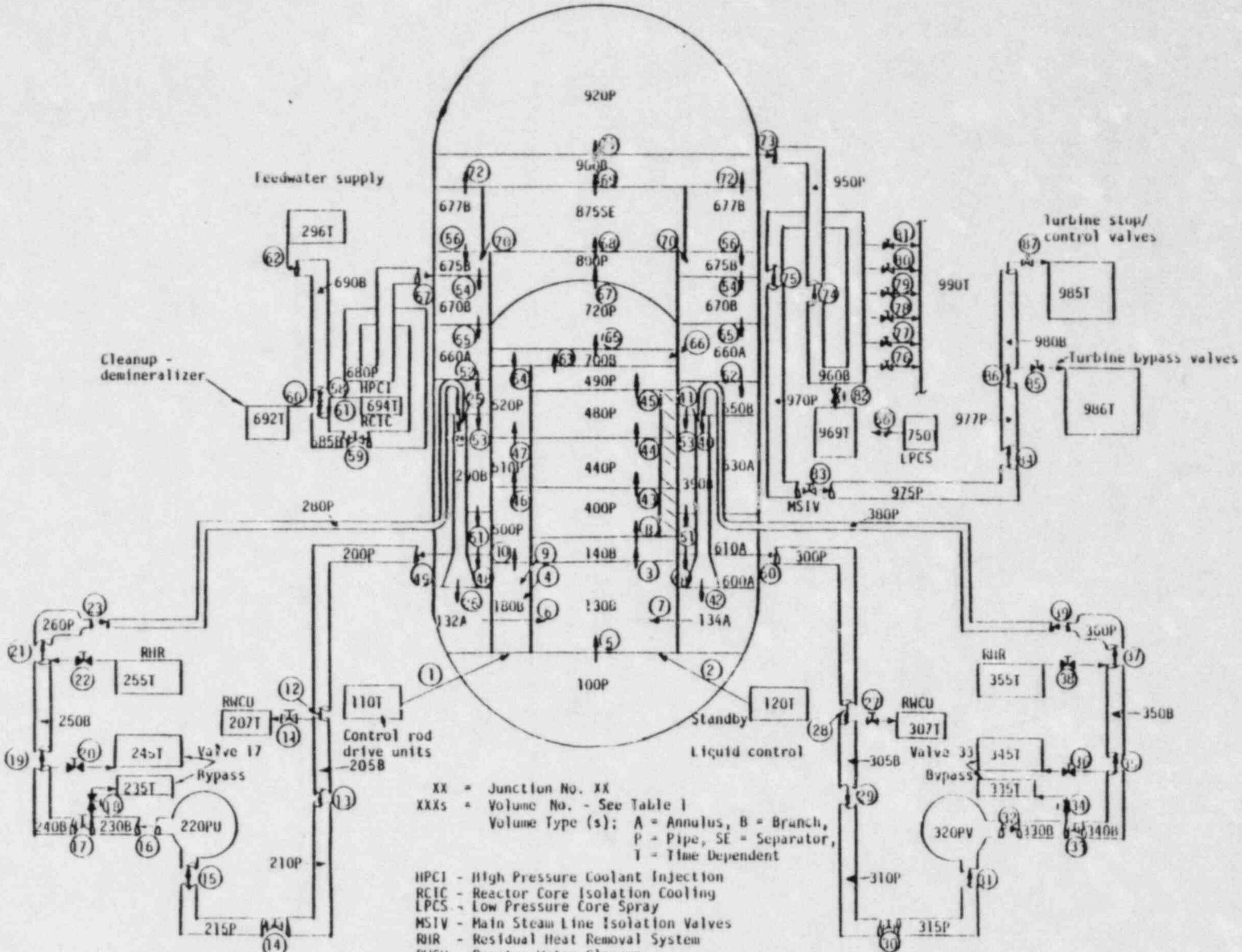


Figure B3. The simplified Browns Ferry Unit One RELAP5 model.

2. Core exit geometry--CCFL can only be predicted and accounted for when the proper loss coefficients and flow area at the upper tie plates are available.

3. Core inlet geometry--Accurate modeling of a CCFL situation at the core inlet of a high, medium and low (peripheral) bundle cannot be modeled. To date only information concerning an average bundle was provided.

4. Fuel rod spacer hydraulic description--The pressure loss distribution throughout the fuel bundles cannot be calculated without an accurate spacer representation.

5. Core wide power distribution--Peaking factors i.e., radial, local and axial for representative high, medium and low power and steaming rates during LOCA transients.

6. Reactivity information--Reactivity information for high, medium and low power bundles over the full operational envelope are needed to accurately predict core power behavior at all conditions including anticipated transients without scram (ATWS).

B.5.2 Recirculation and Jet Pumps

Information describing the recirculation and jet pump behavior at normal operating conditions have been provided. However, many of the transients will involve predicting the system behavior under very abnormal conditions e.g., large LOCAs. Thus further off design pump information is needed:

1. Recirculation Pump Homologous Curves and Two Phase Degradation Characteristics--Pump behavioral characteristics for all quadrants excluding the normal quadrant need further definition. Some large LOCA scenarios postulate recirculation loop failure on the pump suction side--pump reversal occurs.

2. Jet Pump Off Design Behavior--Information describing the jet pump M-N characteristics for drive-forward/suction-forward, drive-forward/suction-reverse, drive-reverse/suction-forward and drive-reverse/suction-reverse are needed for the same reason as in item 1 above. Also detailed geometry layouts are needed.

B.5.3 Other Information

In addition, detailed descriptions of the following components are needed to provide better model simulations or to permit QA checks to be completed.

1. Up-to-Date Vessel Information--Other than the RETRAN input (Reference B.2) the only other source of information describing the vessel and most of its internals was a figure in the FSAR (Reference B.1) showing the vessel (Figure 4.2-1) and marked "Not Updated". Complete QA checks require the most up-to-date information concerning all vessel components.

2. Separator Behavior--In addition to more detailed drawings of the separators, carryunder/carryover behavior descriptions of the separators are needed to model the full spectrum of operational transients.

3. Valves C_v's--Valve C_v descriptions are needed to properly model the system operational transient behavior.

4. Control Systems Descriptions--Although control systems descriptions have been provided in the form of RETRAN input (Reference B.2), the reference hardware descriptions, layouts, and controls tuning/balancing/adjusting data are needed to QA and model the system proper. Without the raw information, the model will be using "biased" input i.e., Tennessee Valley Authority engineers judged what information was needed to properly model the plant.

REFERENCES

B.1 The Browns Ferry Safety Analysis Report.

B.2 S. L. Forkner, D. L. Bell and E. N. Winkler, Calculation of a Generator Load Rejection Transient in a 3293 MWt BWR/4 with RETRAN-01-RET12B, July 1978.

B.3 B. F. Saffell ltr to J. E. Solecki, "Information Requirements for Construction of a Detailed Browns Ferry Model for TRAC-BD1" Saff-175-82, May 11, 1982.

B.4 R. R. Schultz and S. R. Wagoner, The Station Blackout Transient at the Browns Ferry Unit One Plant--A Severe Accident Sequence Analysis, EGG-NTAP-6002, August 1982.

B.5 C. E. Hendrix, Hope Creek BWR Evaluation Model-Behavior Analysis, PG-R-05-77, February 1977.

B.6 R. R. Schultz, Personal Communication with D. Caraher, Spring 1981.

B.7 R. R. Schultz, Personal Communication with T. Burton, September 11, 1981.

B.8 R. J. Wagner et al., RELAP5/MOD1 Code Manual Volume 2: Users Guide and Input Requirements, NUREG/CR-1826, November 1980.

B.9 J. S. Miller, BWR/6 "Best-Estimate" RELAP4 Model-Blowdown Analysis, PG-R-77-32, July 1977.

B.10 R. W. Shumway and R. R. Schultz, "A Boiling Water Reactor/G Small Break Analysis Using TRAC-BD1," ANS Specialists Meeting on Small Break Loss-of-Coolant Accident Analyses in LWRs Conference Papers, August 25-27, 1981.

TABLE B.1 INTERIM BROWNS FERRY RELAP5 MODEL VOLUME DESCRIPTION

Volume Number	Volume[a] Type	Volume Description
100	P	Lower plenum (lower portion)
110	T	Control rod drive feed volume
120	T	Standby liquid control system
130	B	Lower plenum (upper portion)
132	A	Lower plenum
134	A	Lower plenum
140	B	Core inlet volume
180	B	Control rod drive plena
200/300	P	Recirculation line--suction
205/305	B	Recirculation line--suction
207/307	T	Reactor water cleanup volume
210/310	P	Recirculation line--suction
215/315	P	Recirculation line--suction
220/320	PU	Recirculation line--discharge
235/335	T	Recirculation line isolation valve bypass
240/340	B	Recirculation line--discharge
245/345	T	Recirculation line isolation valve bypass
250/350	B	Recirculation line--discharge
255/355	T	Residual Heat Removal System
260/360	P	Recirculation line--discharge plenum
280/380	P	Recirculation line jet pump drive
290/390	B	Jet pumps
400	P	Core--lower third
440	P	Core--middle third
480	P	Core--upper third
490	P	Core--upper unheated volume
500	P	Bypass-lower volume
510	P	Bypass-middle volume
520	P	Bypass-upper volume
600	A	Downcomer--bottom to recirculation line suction midplane
610	A	Downcomer--recirculation line suction to jet pump diffuser inlet
630	A	Downcomer--jet pump diffuser inlet to jet pump suction
650	B	Downcomer--jet pump
660	P	Downcomer--jet pump drive line top to upper plenum
670	B	Downcomer--upper plenum to feedwater line sparger

TABLE B.1 (continued)

Volume Number	Volume[a] Type	Volume Description
675	B	Downcomer--feedwater line sparger to the steam separator bottom
677	P	Downcomer--steam separator bottom to top
680	P	Feedwater line--inboard to drywell check valves
685	P	Feedwater line--check valves to HPCI/RCIC system inlets
690	B	Feedwater line--HPCI/RCIC inlets to the feedwater heaters
692	T	Cleanup-demineralizer volume
694	T	High pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems
696	T	Feedwater supply volume
700	B	Upper plenum--between the core bundle upper tie plate and the core spray sparger injection plane
720	P	Upper plenum--upper portion
750	T	Low pressure core spray
800	P	Separator standpipes
875	SE	Separator
900	B	Steam dome--top of separators to the steam lines
920	P	Steam dome
950	P	Steam lines
960	B	Steam line
969	T	HPCI and RCIC systems turbine dump volume
970	P	Steam line
975	P	Steam line
977	P	Steam line
980	B	Steam line between the turbine bypass and control valves
985	T	Steam line beyond the turbine control valve
986	T	Condenser volume
990	P	Containment

a. Volume type:
A = Annulus,
B = Branch,
P = Pipe,
PU = Pump,
SE = Separator,
T = Time Dependent

TABLE B.2 THE SAFETY-RELIEF VALVE MODELS

Junction	Number of Valves[a] Modeled	Open Setpoint ~ psia	Close Setpoint ~ psia
76	1	1131.0	1081.0
77	1	1131.2	1081.2
78	2	1131.5	1081.5
79	4	1141.0	1091.0
80	3	1151.0	1101.0
81	2	1278.0	1228.0

a. Junctions 76 through 80 were modeled to pass 236.4 lbm/s value of steam at 1130.7 psia. Junction 81 was modeled to pass 257.1 lbm/s value at 1281.7 psia.

APPENDIX C
A SUMMARY DISCUSSION OF THE RELAP5 MASS ERROR PROBLEM RESOLUTION

W. C. Jouse

APPENDIX C

A SUMMARY DISCUSSION OF THE RELAP5 MASS ERROR PROBLEM RESOLUTION

Severe Accident Sequence Analysis (SASA) calculations were often inhibited by excessive[a] RELAP5 mass error. An investigation was initiated to isolate and correct the source of this error. Appendix C is a summary of the activities undertaken.

The SASA calculations which exhibited excessive RELAP5 mass error simulated (a) a stuck open relief valve, and (b) a lower plenum small break using the Browns Ferry (BWR) model. Both calculations began with a system mass inventory of approximately 0.75×10^6 lbm, and were to run 8 to 10 hours of simulated time until core recovery.

The mass error of any system volume is computed thusly: let ρ_s denote the state of density (obtained from the equation of state), and let ρ_m denote the mixture density (obtained from the continuity equation). The mass error ε is obtained by differencing the densities as

$$\varepsilon = (\rho_s - \rho_m) * V$$

where V is the volume of the component.

The time advancement is rejected if the mass error is excessive.

Phase boundary crossings occur whenever a system volume is carried into a single phase state wherein the static quality is, by definition, either exactly zero or unity. Since the thermal-hydraulic/numeric conditions of RELAP5 execution are rarely such that the new-time static

---

a. Customarily, mass errors greater than 5% are deemed excessive (Reference C.1). Mass errors as high as 20% were observed during some SASA calculations.

quality is either exactly zero or unity, a static quality overshoot occurs. Subsequent truncation of the overshoot results in an apparent loss or gain in system mass, i.e., a mass error. Large mass errors accompany such crossings.

The results of the investigation for the lower plenum break calculation are shown in Figures C-1 and C-2. Figure C-1 shows the corrected and uncorrected system mass error as accumulated over the first 5000 seconds of the calculation. The magnitude of the mass error was reduced from 7% to 1% by a set of code modifications. Figure C-2 shows the effect of the modifications to system pressure and transient timing. The safety relief valve cycling is periodically interrupted by cold ECC injection.

Having identified sources of mass error, modifications to RELAP5 have systematically and effectively mitigated the effect of the error on the transient simulation. Within the scope of the investigation, efforts to assess and correct the problem have been successful.

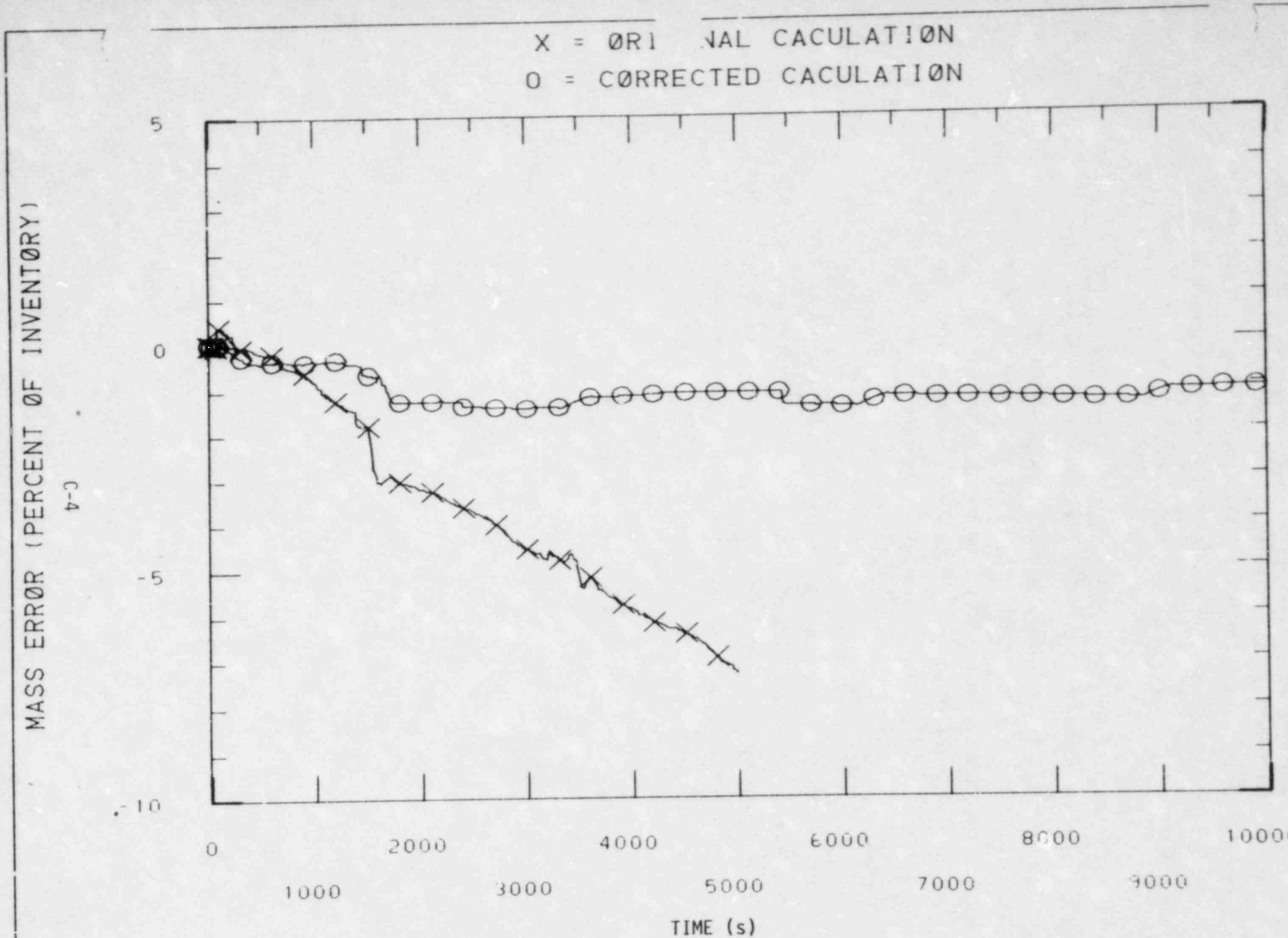


Figure C-1. Relative mass error versus time.

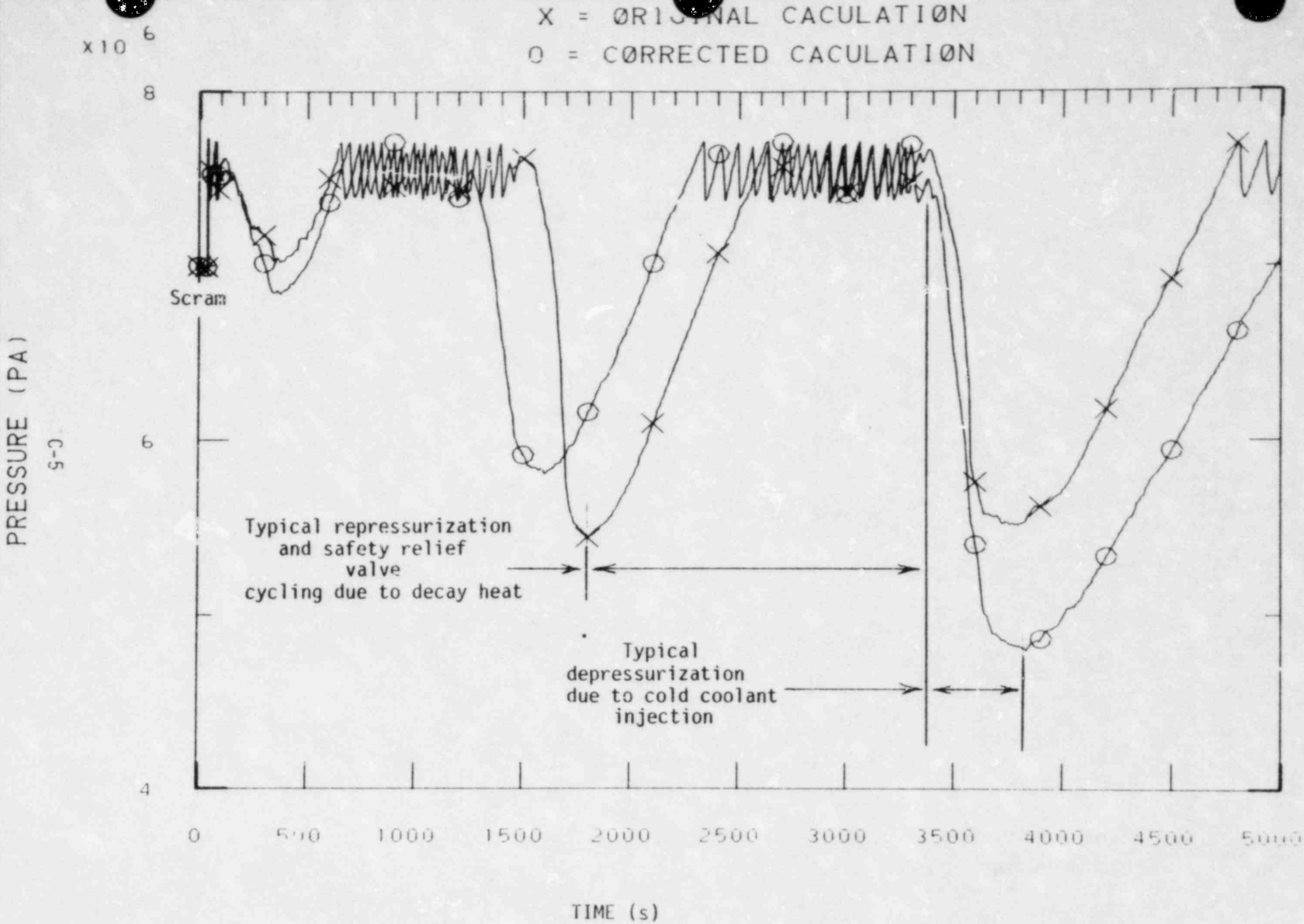


Figure C-2. Steamdome pressure versus time.

REFERENCES

C.1 V. H. Ransom, et al., "The RELAP5 Two-Phase Fluid Model and Numerical Scheme for Economic LWR System Simulation", 3rd CSNI Specialist Meeting in Transient Two-Phase Flow Calculations Techniques, Pasadena, CA, March 1981.

APPENDIX D

THE BOILOFF TRANSIENT WITH STUCK OPEN RELIEF VALVE (SORV): SEQUENCE V12

APPENDIX D

THE BOILOFF TRANSIENT WITH STUCK OPEN RELIEF VALVE (SORV): SEQUENCE V12

The sequence V12 calculation was based on the same set of assumptions as sequence V6 (see Section 4.2) except a safety-relief valve was assumed to fail open after it lifted the first time. Thus, without any systems available to replenish the vessel inventory, the vessel depressurizes (see Figure 11) until core uncovery occurs intermittently as early as 1580 s (see Figure 12). Core uncovery is sufficient by 1680 s to allow a sustained heatup of over 20 s.

The additional figures presented in this appendix show the behavior of the water level in the downcomer and inside the core shroud (Figures D.1 and D.2, respectively). In addition the total mass and energy delivered to the pressure suppression pool are presented in Figures D.3 and D.4, respectively. Finally, the mass injected into the vessel is shown in Figure D.5.

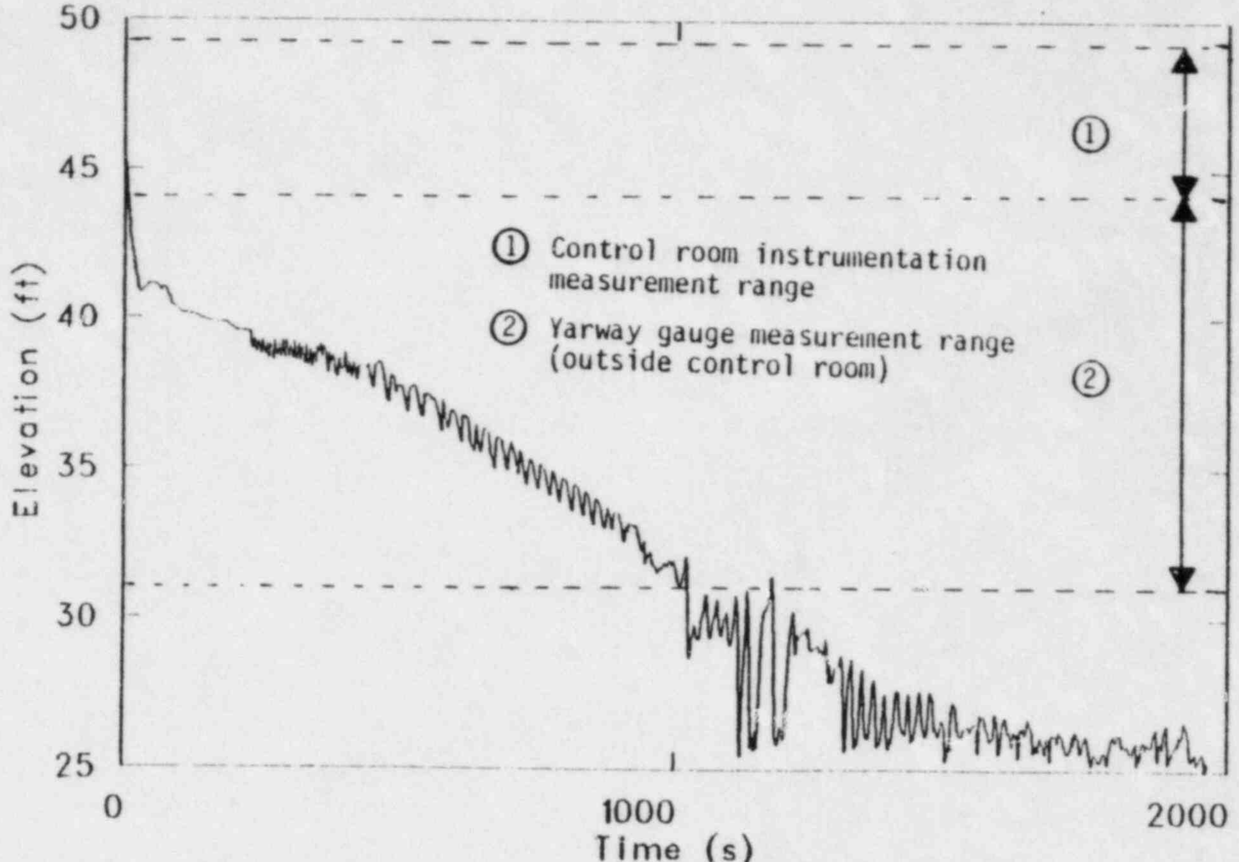


Figure D-1. Downcomer water level - sequence v12

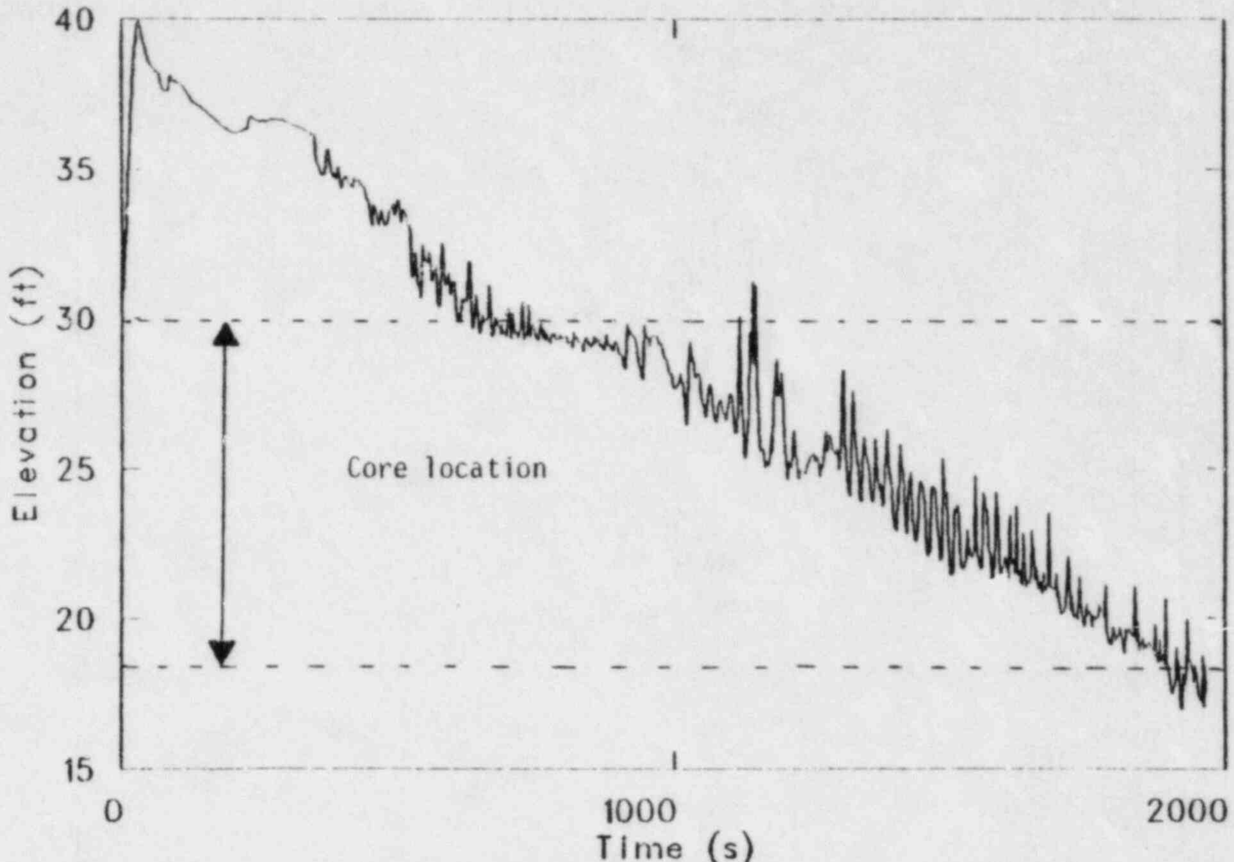


Figure D-2. Core collapsed water level - sequence v12

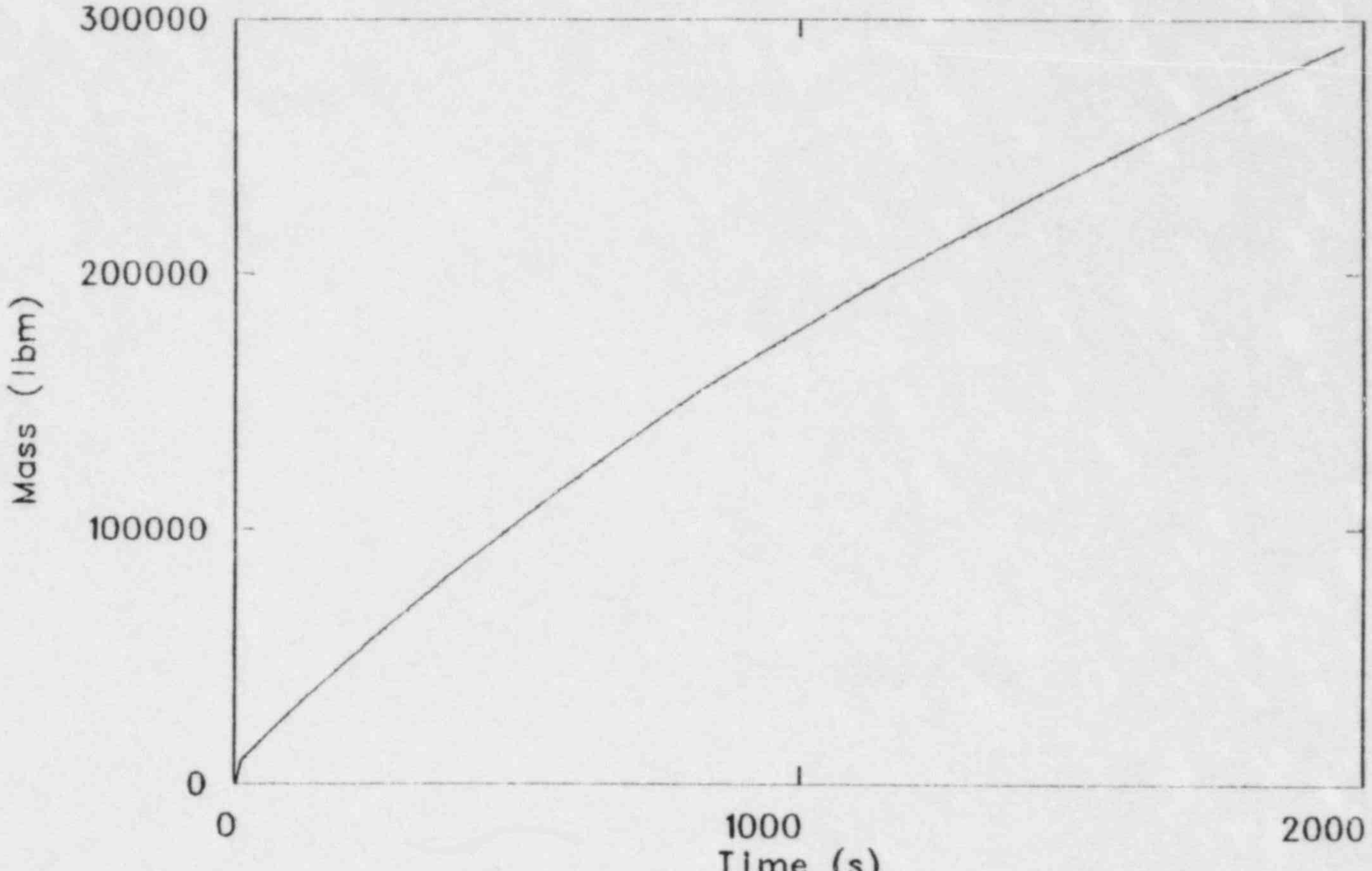


Figure D-3. Total mass transfer to the pressure suppression pool – sequence v12

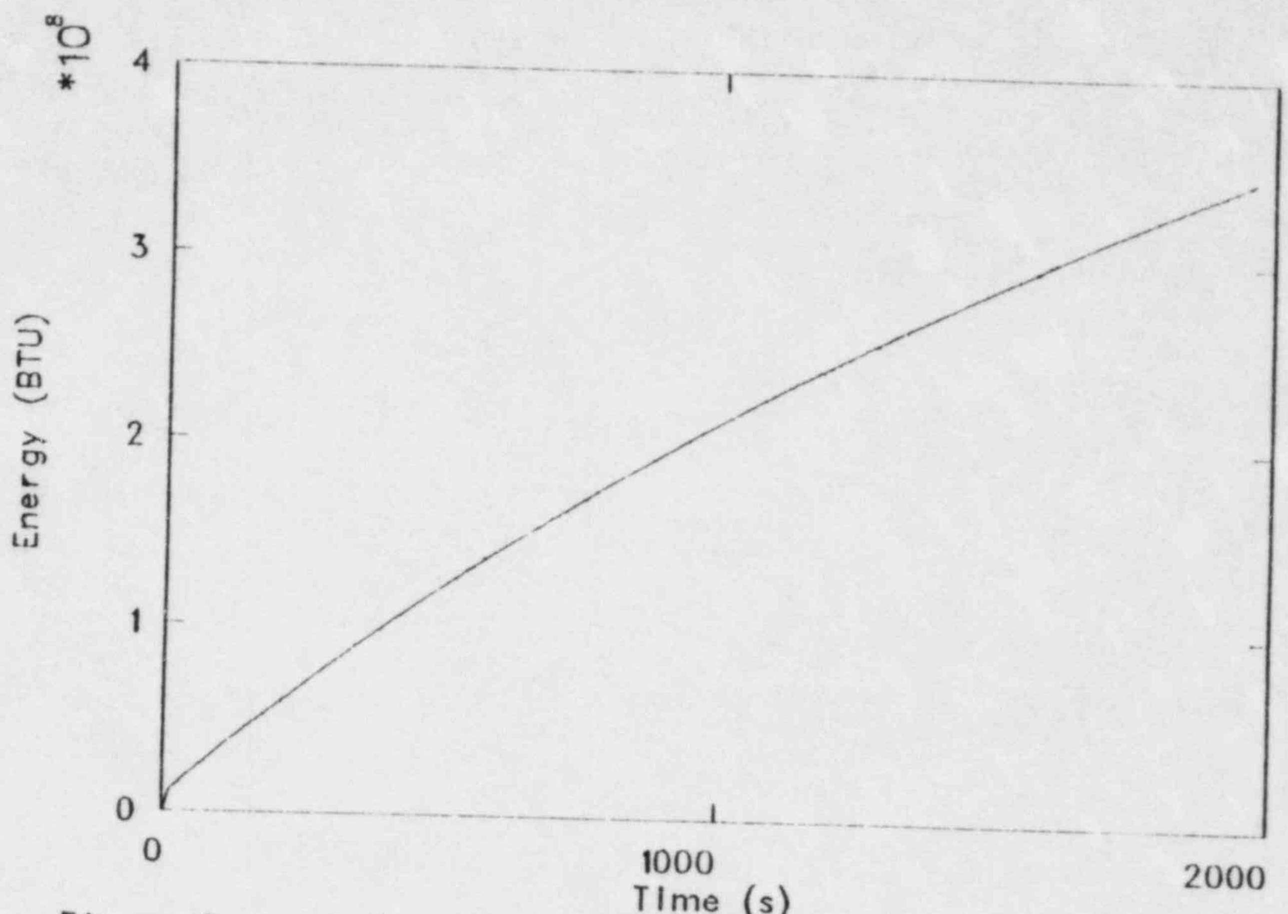


Figure D-4. Total energy transfer to the pressure suppression pool – sequence v12

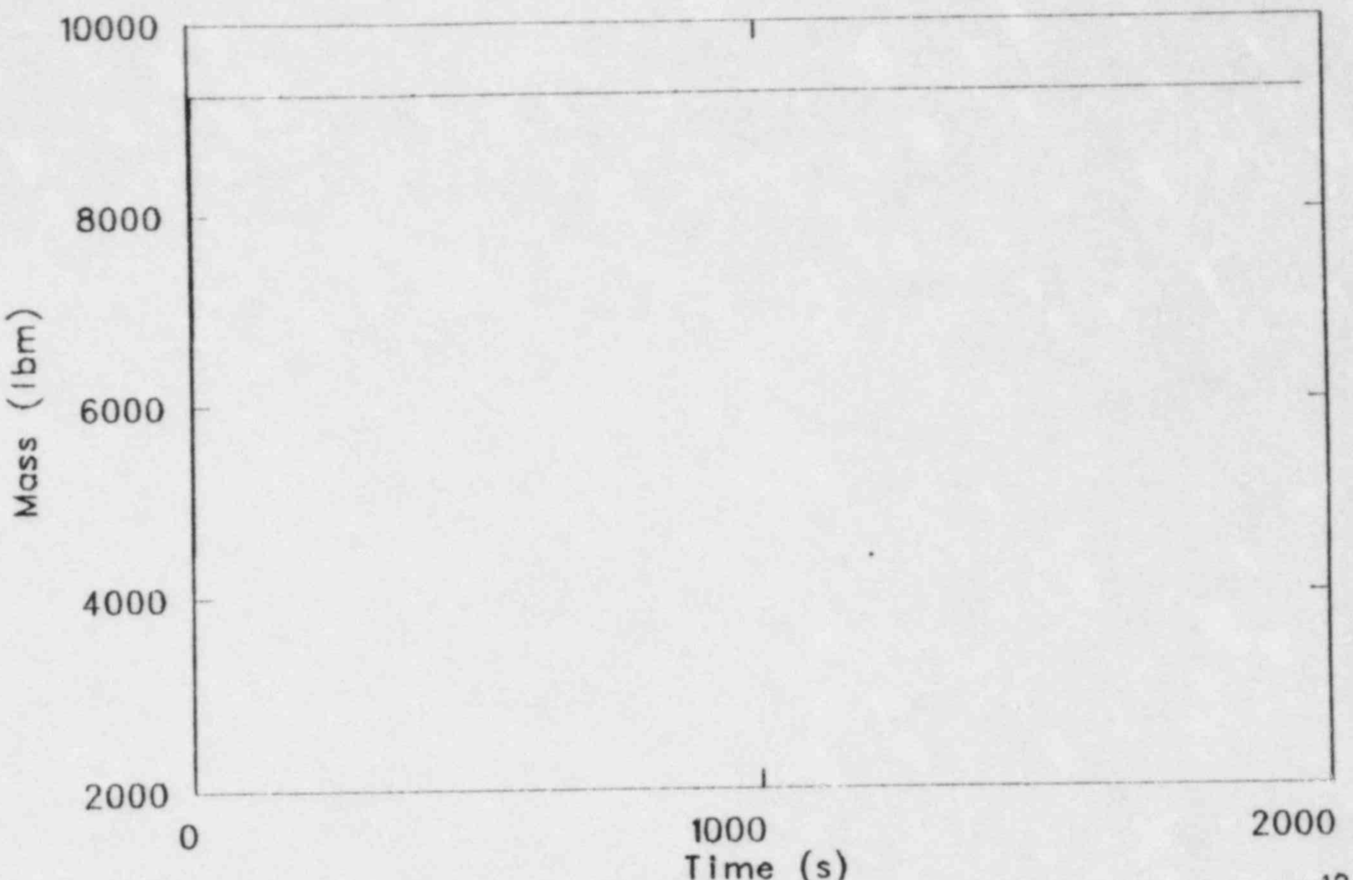


Figure D-5. Total mass transfer to the vessel – sequence v12

APPENDIX E

THE STATION BLACKOUT WITH THE RCIC SYSTEM AVAILABLE: SEQUENCE V2

APPENDIX E

THE STATION BLACKOUT WITH THE RCIC SYSTEM AVAILABLE: SEQUENCE V2

The calculated results represent the sequence that would occur if Unit 1 experienced a station blackout when the high pressure coolant injection (HPCI) system was unavailable. The operator would respond as soon as possible to the large drop in vessel water level (a 90 s response time is assumed in the calculation) by turning on the reactor core isolation cooling (RCIC) system (short term: Figure 13, long term: Figure 14). The SRVs would be opening and shutting at steam line pressures defined by their setpoints (see Figure 15). However, to distribute the steam discharge from the valves to the pressure suppression pool (PSP) uniformly, the operator was assumed to manually operate the valves 120 s after the station blackout occurred. The operator would open a valve at 1100 psig and discharge steam to the PSP until the vessel pressure was relieved to 900 psig. At the same time, the operator would use the RCIC to maintain the downcomer water level in the control room instrumentation measurement range i.e., between 44.8 and 48.2 ft (measured with respect to the vessel zero).

The above actions (defined in Reference 3) were assumed to continue until complete loss of the station batteries occurred at seven hours. Thereafter, the SRVs began cycling between their automatic setpoints. The RCIC system was assumed to no longer be functional. The vessel water inventory boiled off until core heatup began at 34,000 s.

The additional figures presented in this appendix show the behavior of the upper elevation core temperature (Figure E.1) and the core collapsed water level (Figure E.2). In addition, the total mass and energy delivered to the pressure suppression pool are presented in Figures E.3 and E.4 respectively. Finally, the mass injected into the vessel is shown in Figure E.5.

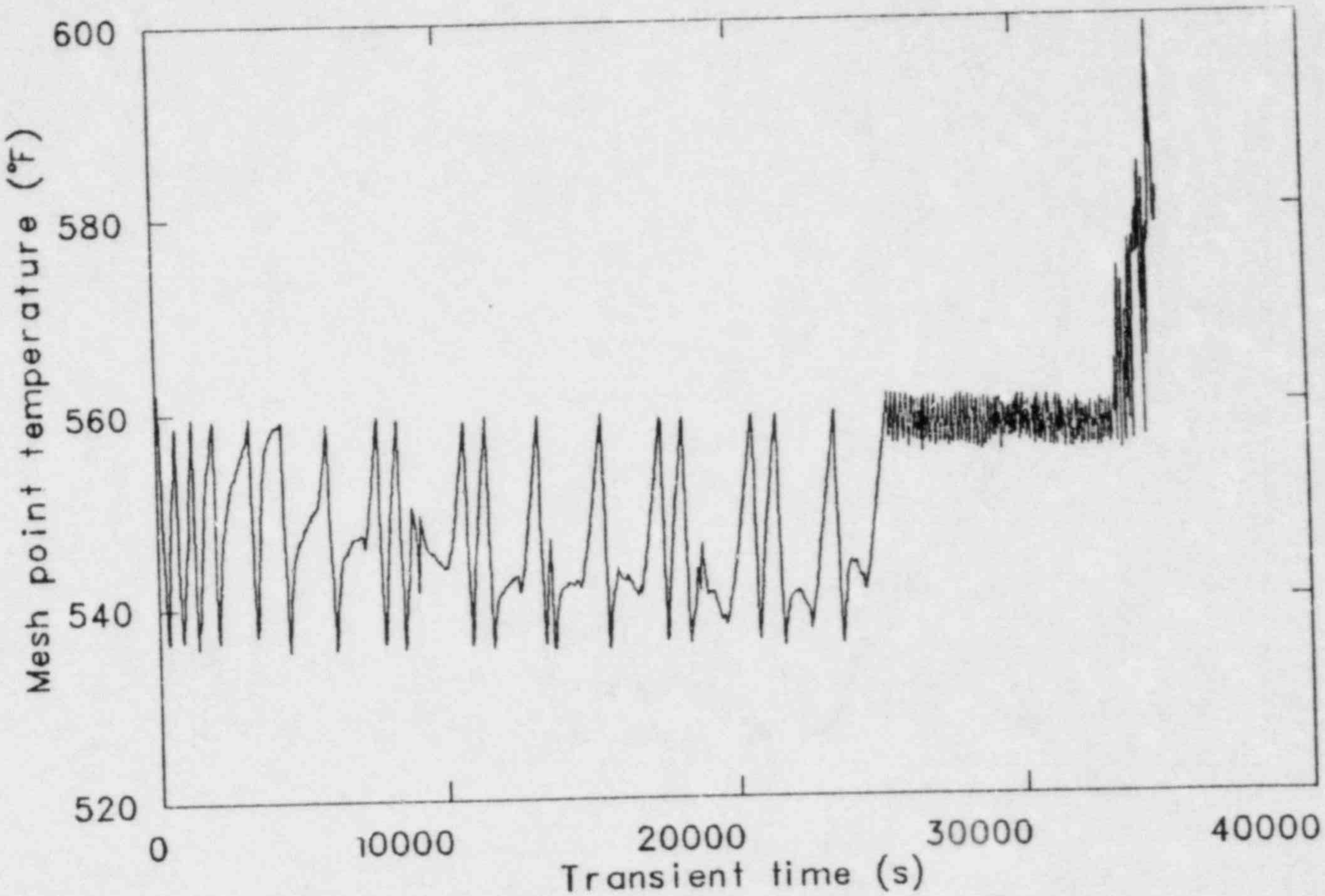


Figure E-1. Upper elevation core temperature - sequence v2.

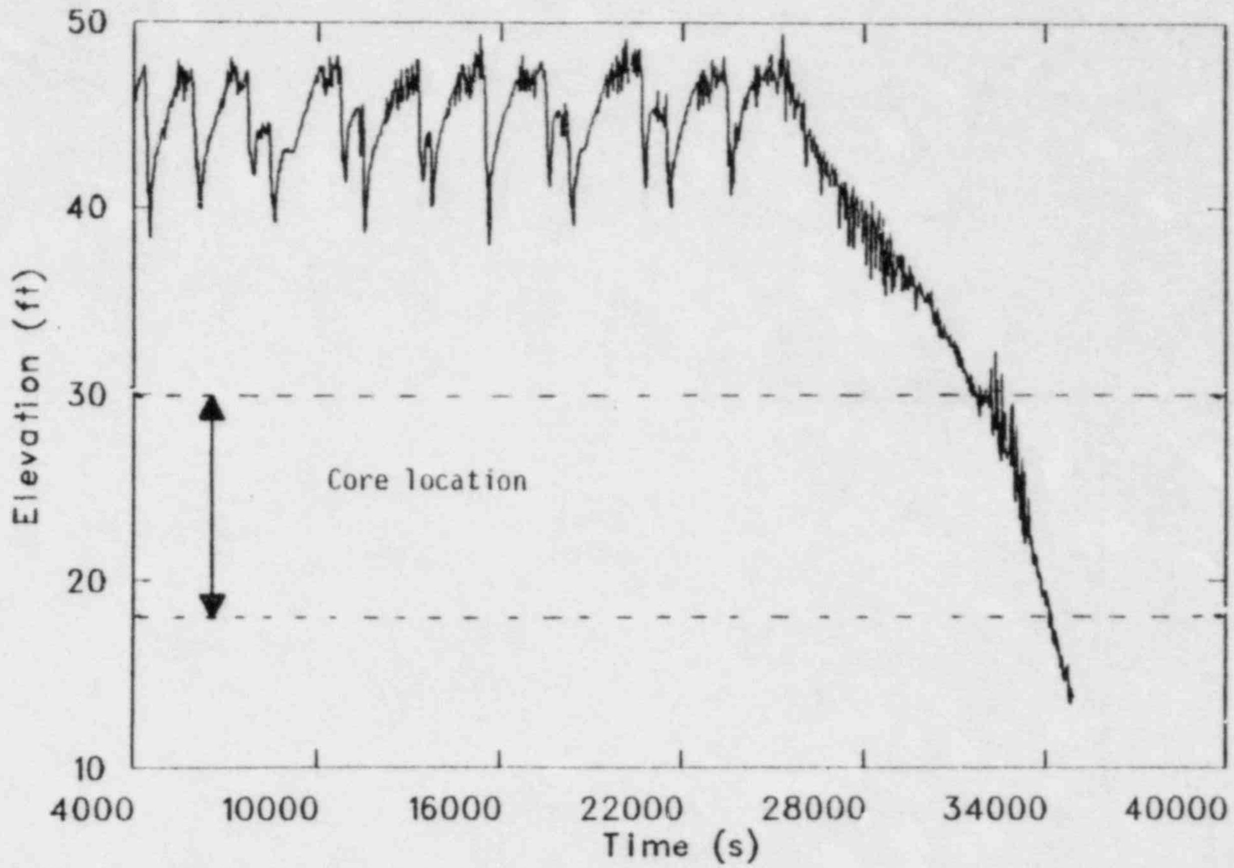


Figure E-2. Core collapsed water level – sequence v2.

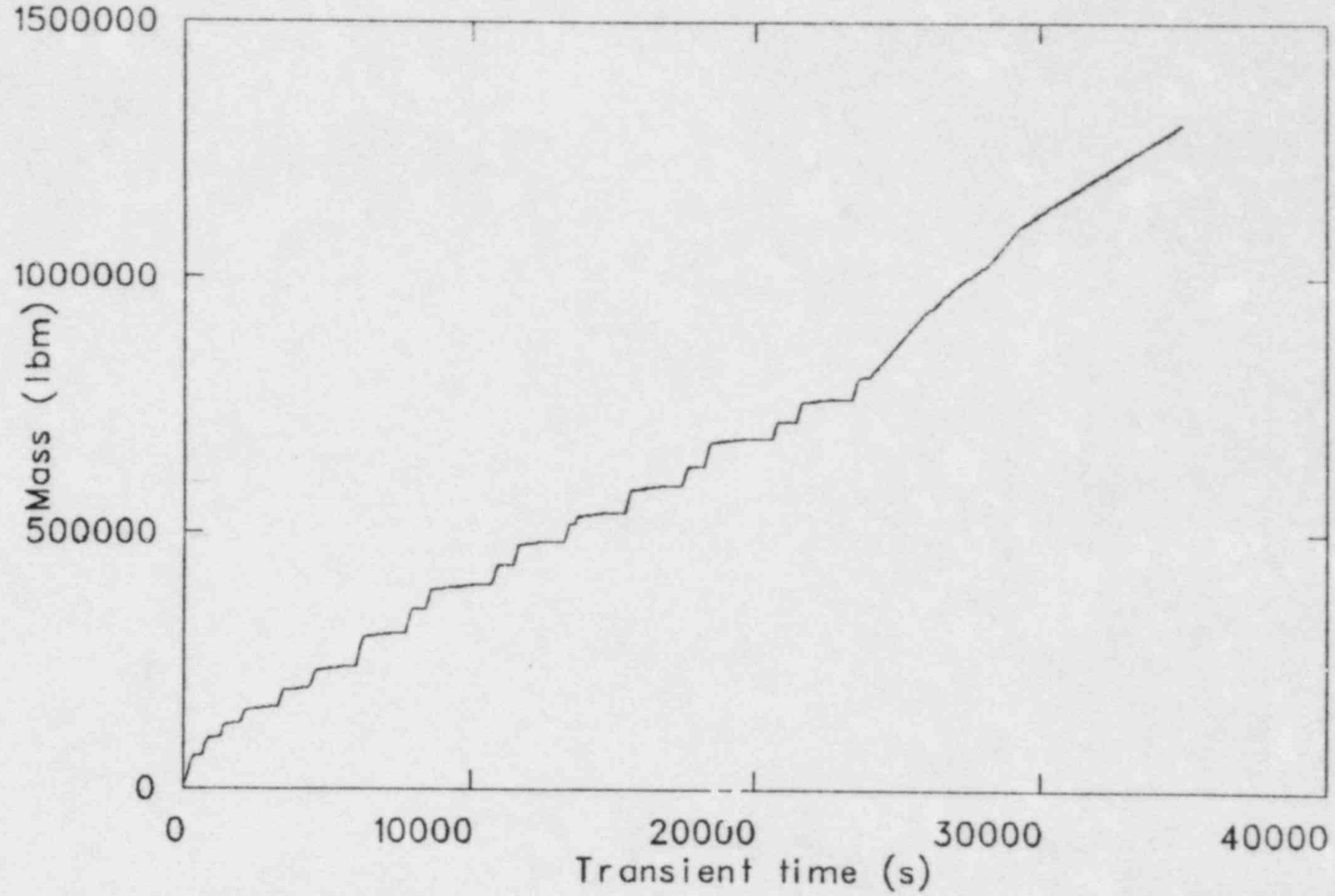


Figure E-3. Total mass transfer to the pressure suppression pool - sequence v2.

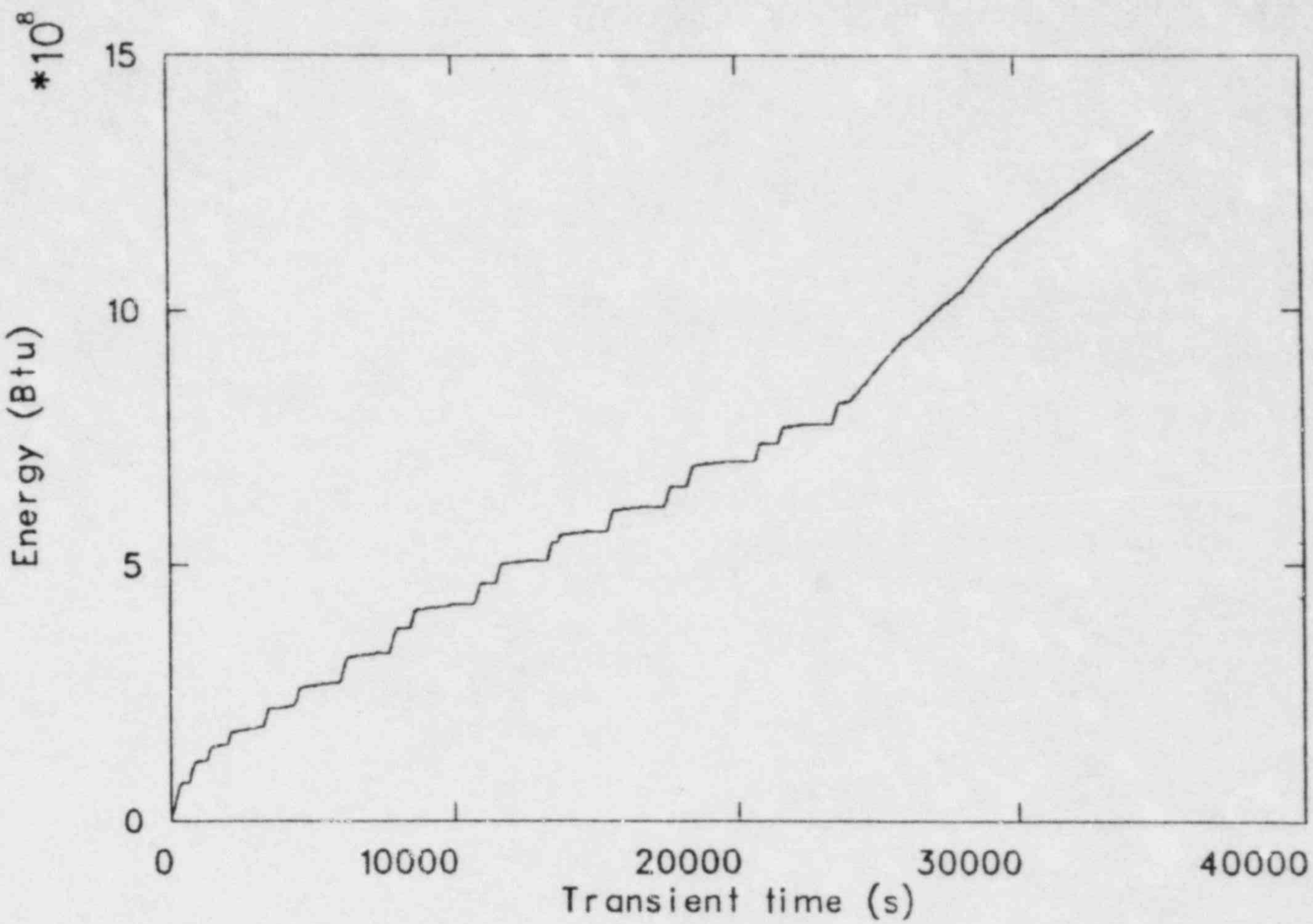


Figure E-4. Total energy transfer to the pressure suppression pool - sequence v2.

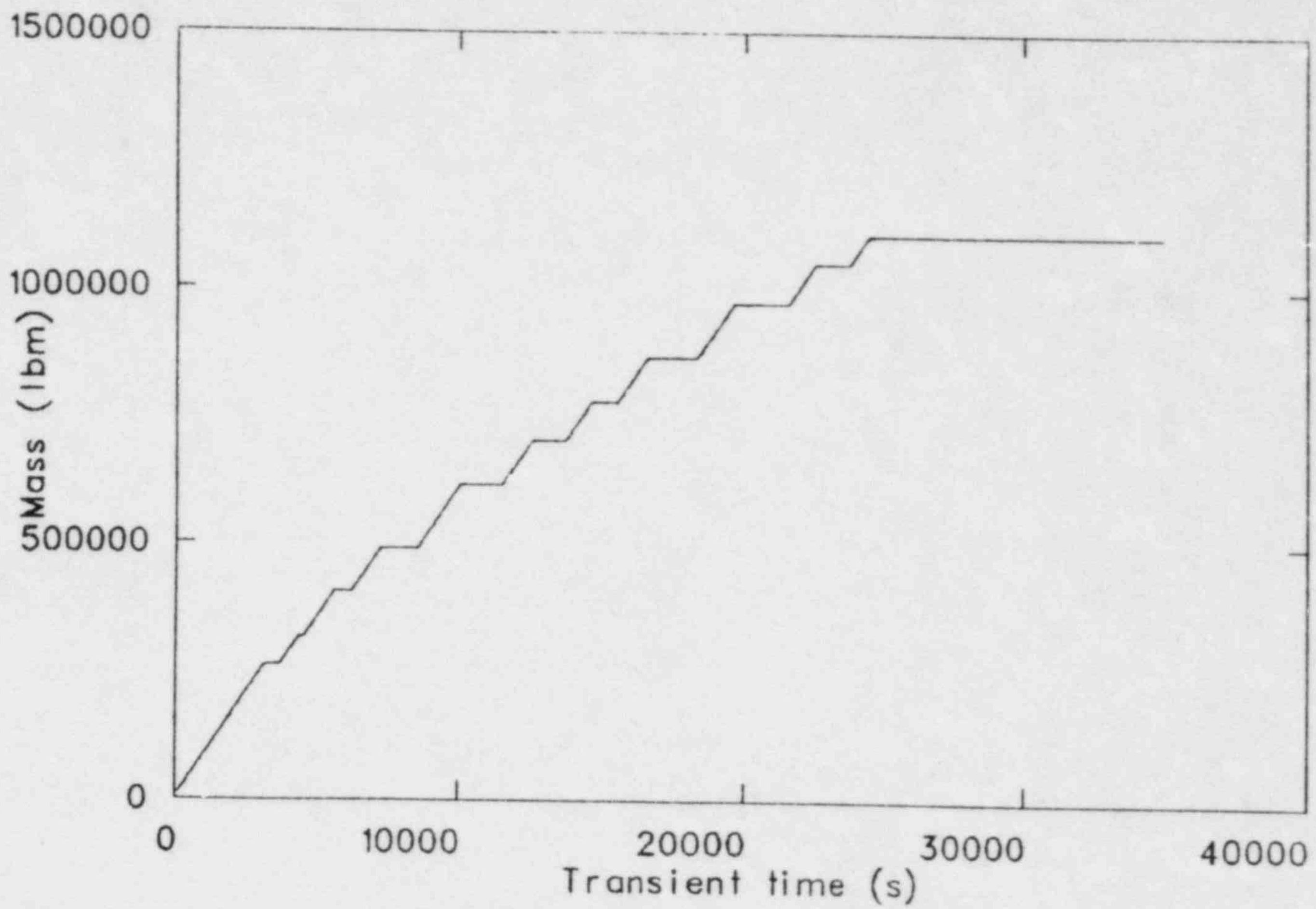


Figure E-5. Total mass flow to the vessel - sequence v2.

APPENDIX F

THE STATION BLACKOUT TRANSIENT WITH THE HPCI SYSTEM AVAILABLE: SEQUENCE V4

APPENDIX F

THE STATION BLACKOUT TRANSIENT WITH THE HPCI SYSTEM AVAILABLE: SEQUENCE V4

The calculation was based on the same set of assumptions as sequence V6 except that the HPCI system was assumed available. The calculation was conducted principally to establish the system behavior when the HPCI is available in the automatic mode. The RCIC system is assumed unavailable.

System behavior was the same as in the first calculation until 295 s (see Figure 16) when the HPCI was initiated by a low-low level trip (see Figure 17). The system provided flow until 525 s when the system was tripped off. The HPCI was initiated a second time at 2385 s and pumped fluid to the vessel until 2645 s. Core uncovery did not occur during the seven hour period that the station batteries were available. However, following station battery failure, the core should uncovery in the same time frame as Sequence V2 (see Subsection 4.4 and Appendix E).

The additional figures presented in this appendix show the behavior of the upper elevation core temperature (Figure F.1) and the core collapsed water level (Figure F.2). In addition, the total mass and energy delivered to the pressure suppression pool are presented in Figures F.3 and F.4 respectively. Finally, the mass injected into the vessel is shown in Figure F.5.

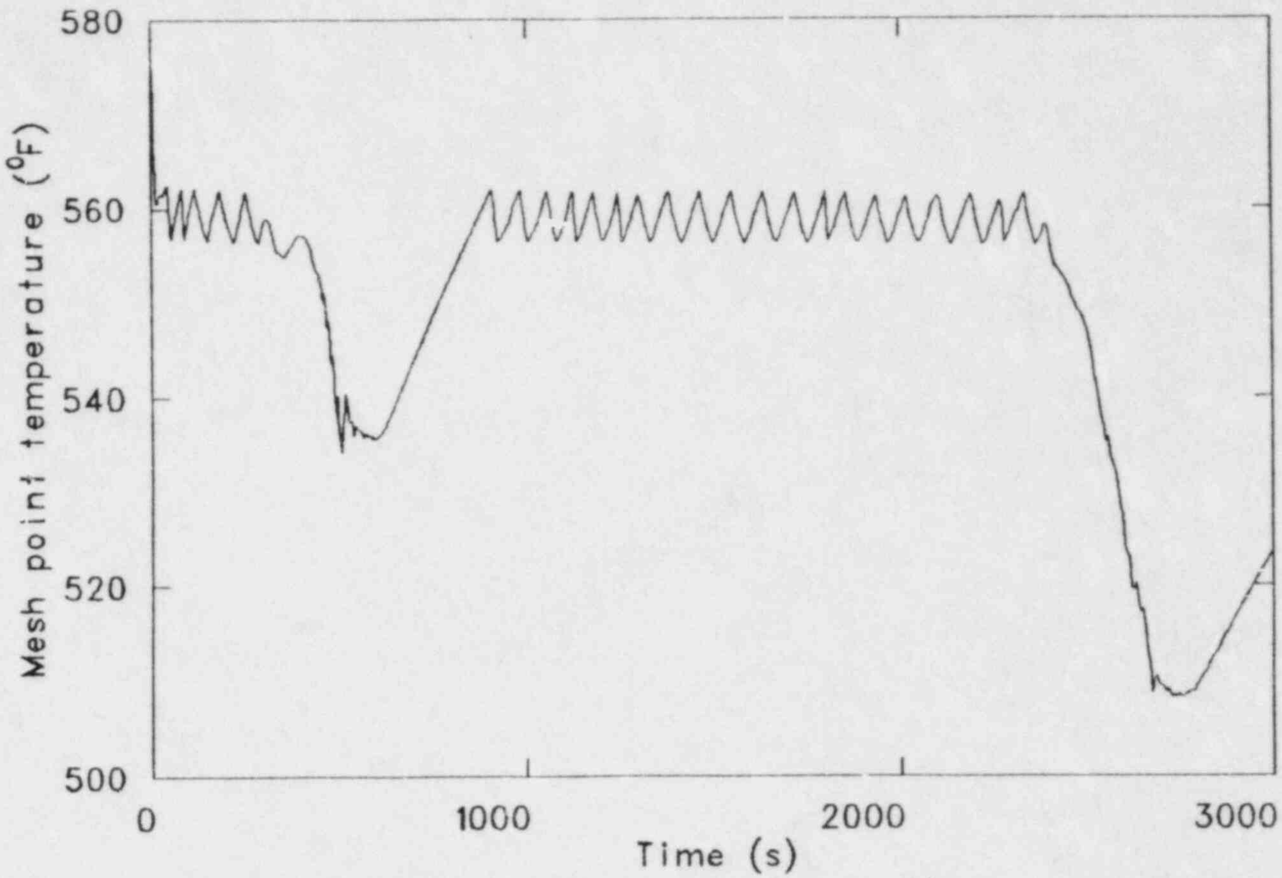


Figure F-1. Upper elevation core temperature - sequence v4.

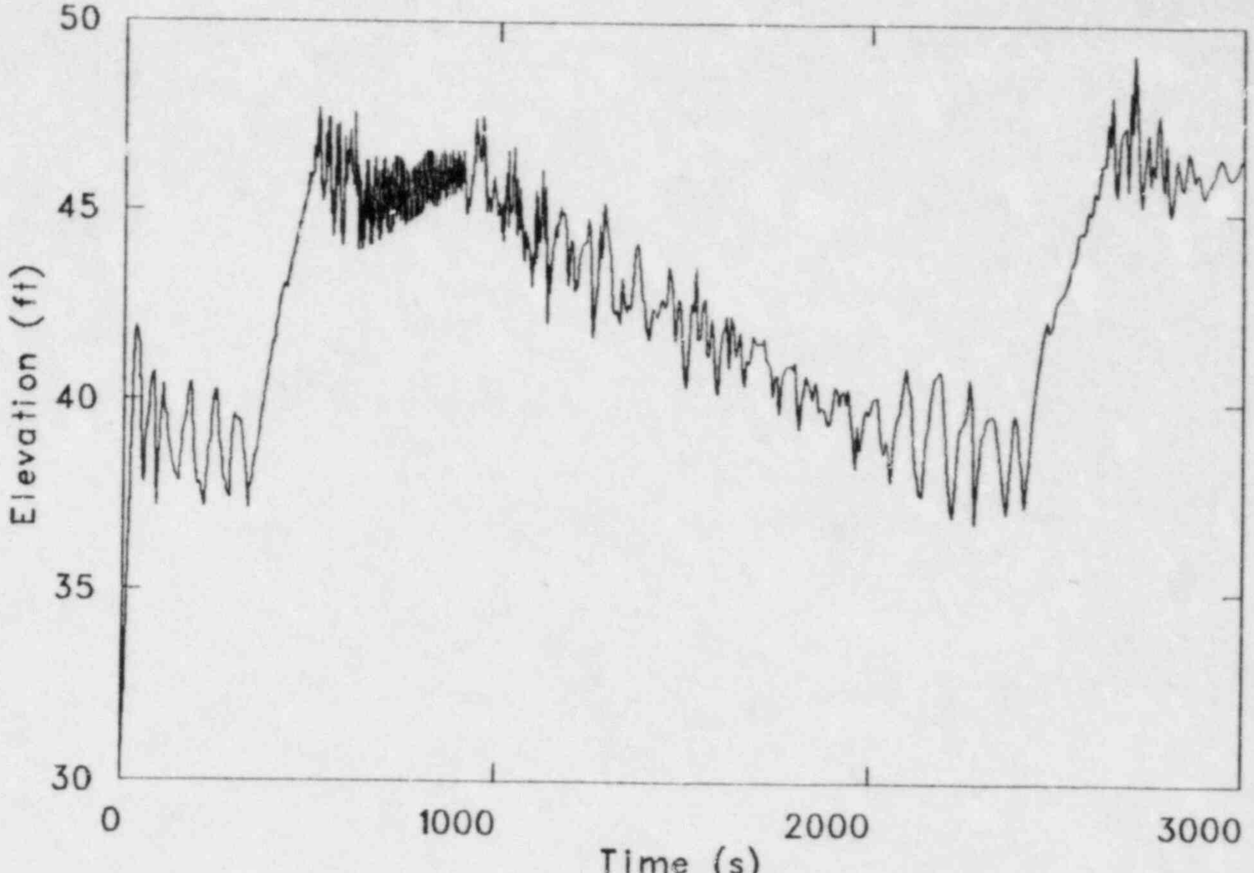


Figure F-2. Core collapsed water level - sequence v4.

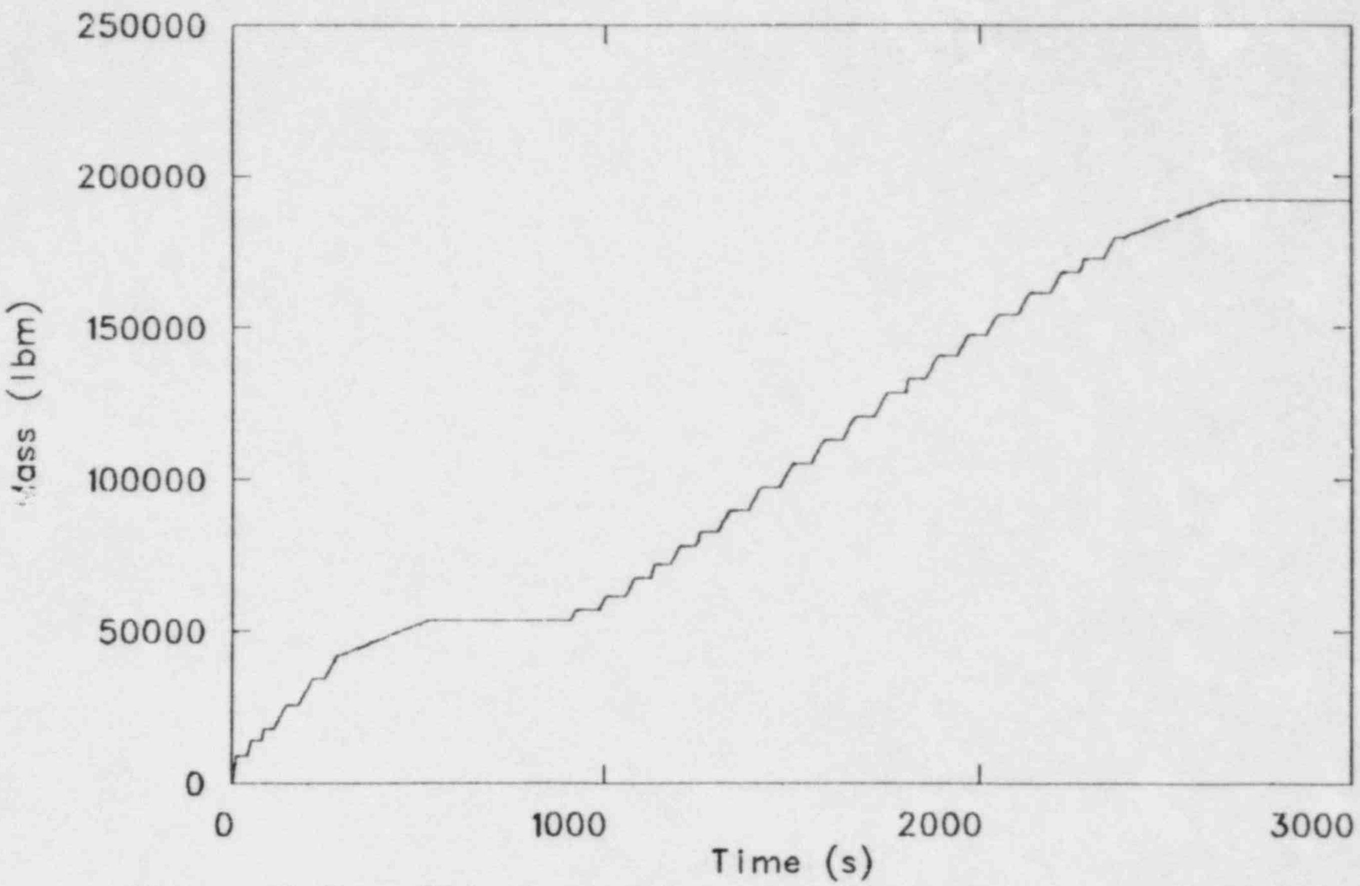


Figure F-3. Total mass transfer to pressure suppression pool – sequence v4

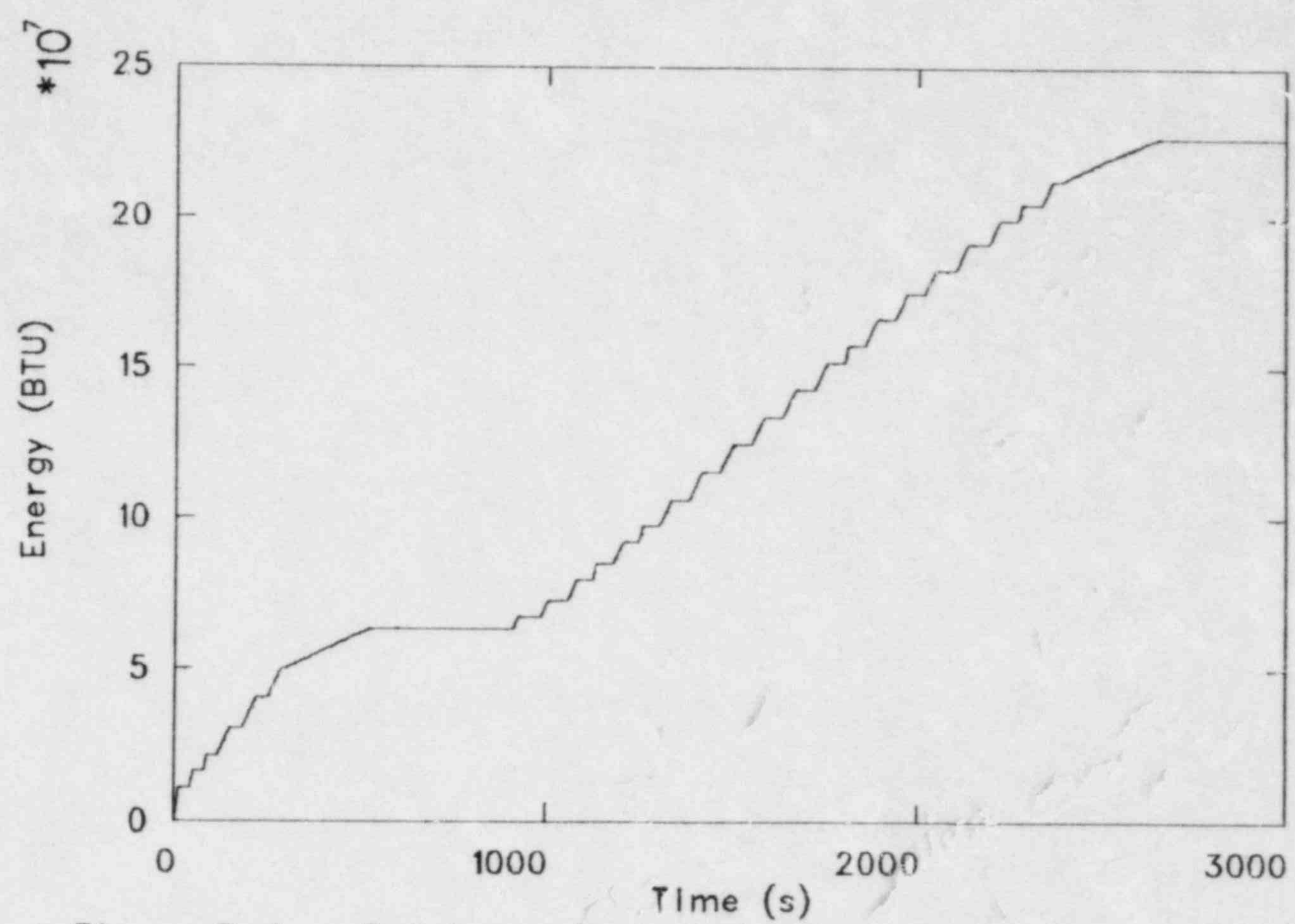


Figure F-4. Total energy transfer to pressure suppression pool -- sequence v4

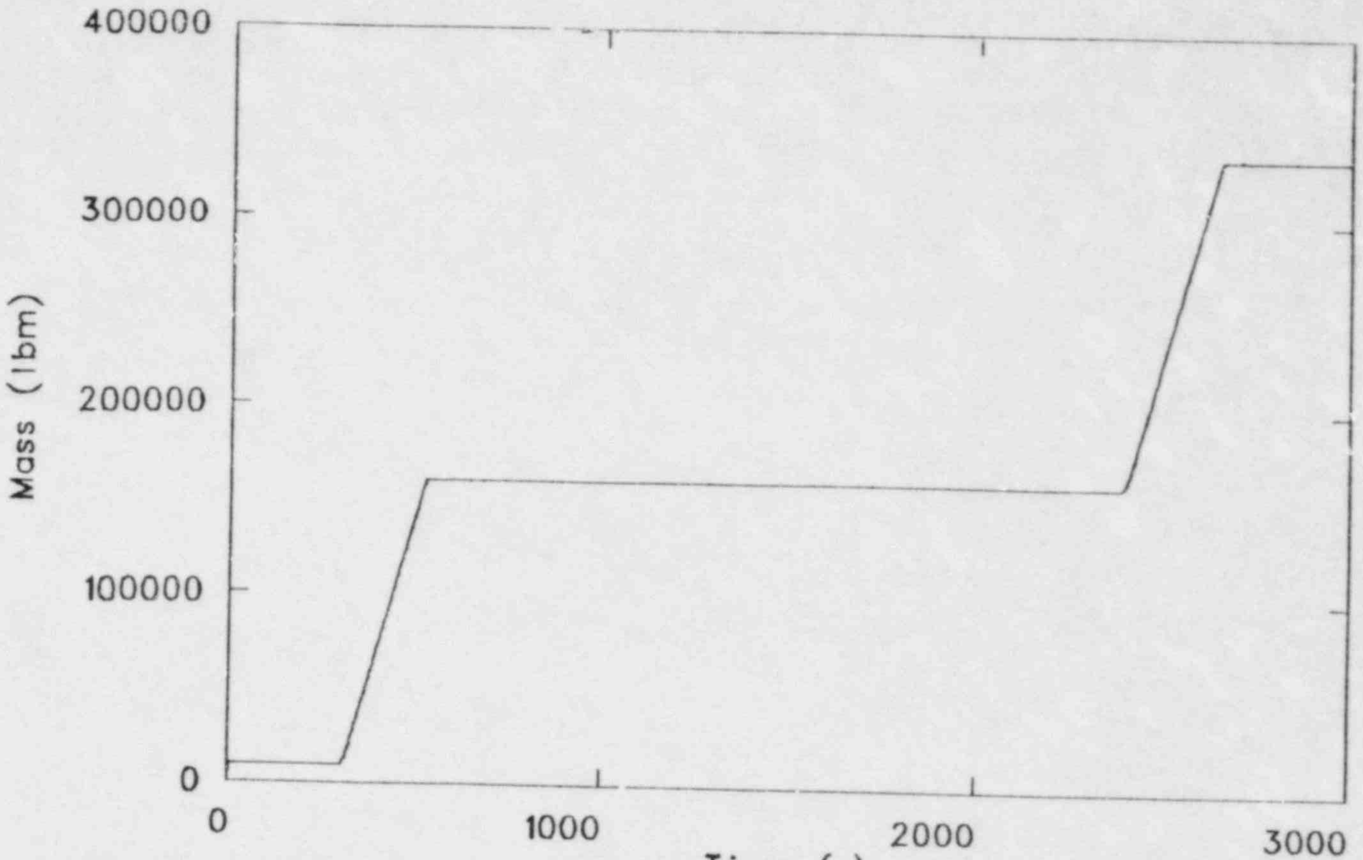


Figure F-5. Total mass transfer to the vessel - sequence v4

APPENDIX G

THE SORV TRANSIENT WITH RCIC AVAILABLE: SEQUENCE V8

APPENDIX G

THE SORV TRANSIENT WITH RCIC AVAILABLE: SEQUENCE V8

Assumptions inherent in the sequence V8 stuck open relief valve (SORV) transient are that following LOSP, only the RCIC system is available to replenish the vessel water inventory. Following the scram and the vessel pressurization (see Figure 18), which accompanies the turbine control valve closure, the safety relief valves (SRV) behave as expected until 22 s when a SRV fails to close.

The operator responds by manual initiation of the RCIC system 90 s after the start of the transient. Thereafter, the operator leaves the RCIC on as needed to maintain the vessel downcomer water level between 44.8 ft and 48.2 ft (short term: Figure 19, long term: Figure 20), i.e., the indicated water level that is available to the operator via the control room instrumentation.

The vessel continually depressurizes as the transient proceeds until 9765 s when the RCIC system is shutoff (due to the downcomer water level reaching the 48.2 ft elevation). Thereafter the vessel pressure alternately increases and then decreases slowly depending on whether the RCIC is off or on, respectively. The RCIC fails at 25,200 s with the loss of the DC battery and the vessel inventory boiloff begins. A RELAP5 calculation was only conducted until 28000 s since the code mass error became excessive as the boiloff proceeded.

The additional figures presented in this appendix show the behavior of the upper elevation core temperature (Figure G.1) and the core collapsed water level (Figure G.2). In addition, the total mass and energy delivered to the pressure suppression pool are presented in Figures G.3 and G.4 respectively. Finally, the mass injected into the vessel is shown in Figure G.5.

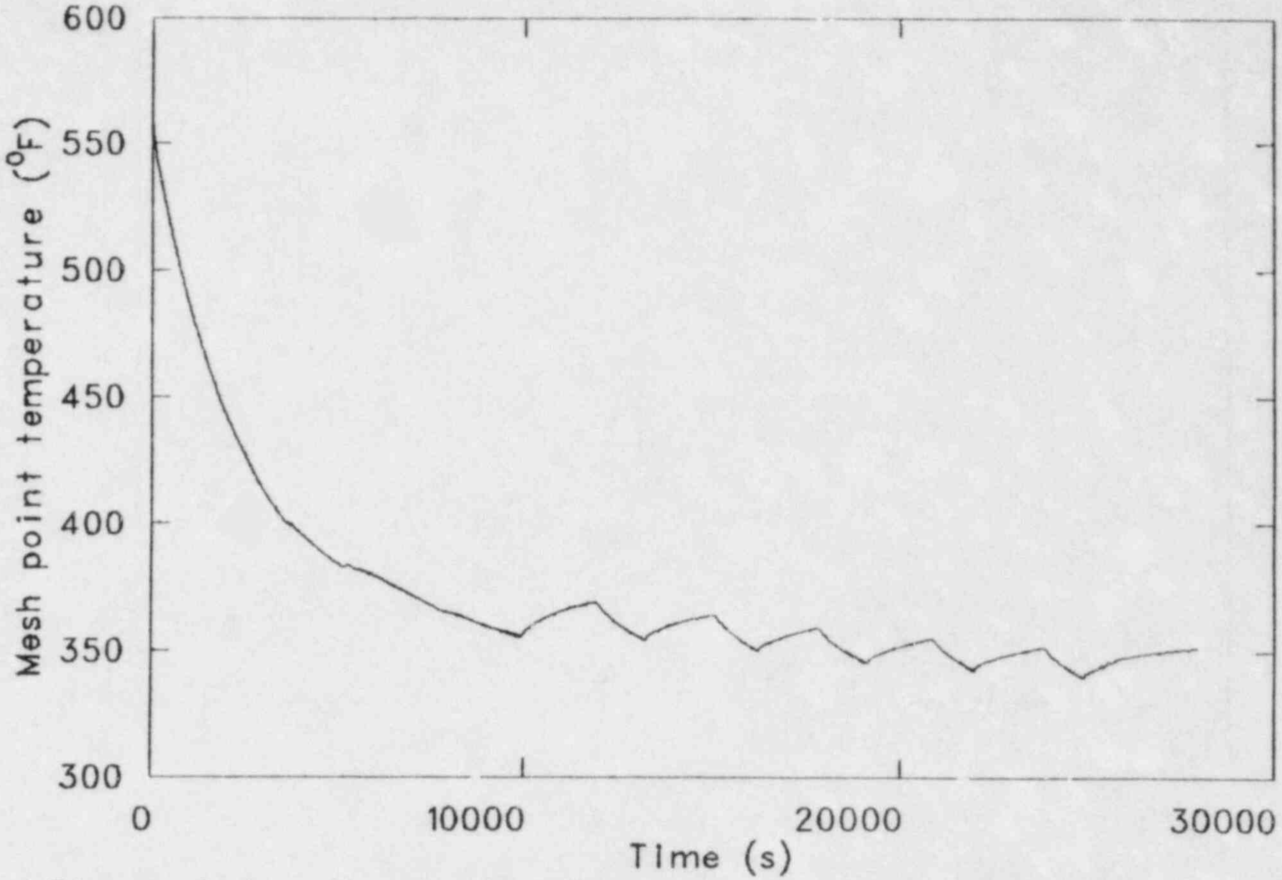


Figure G-1. Upper elevation core temperature -- sequence v8.

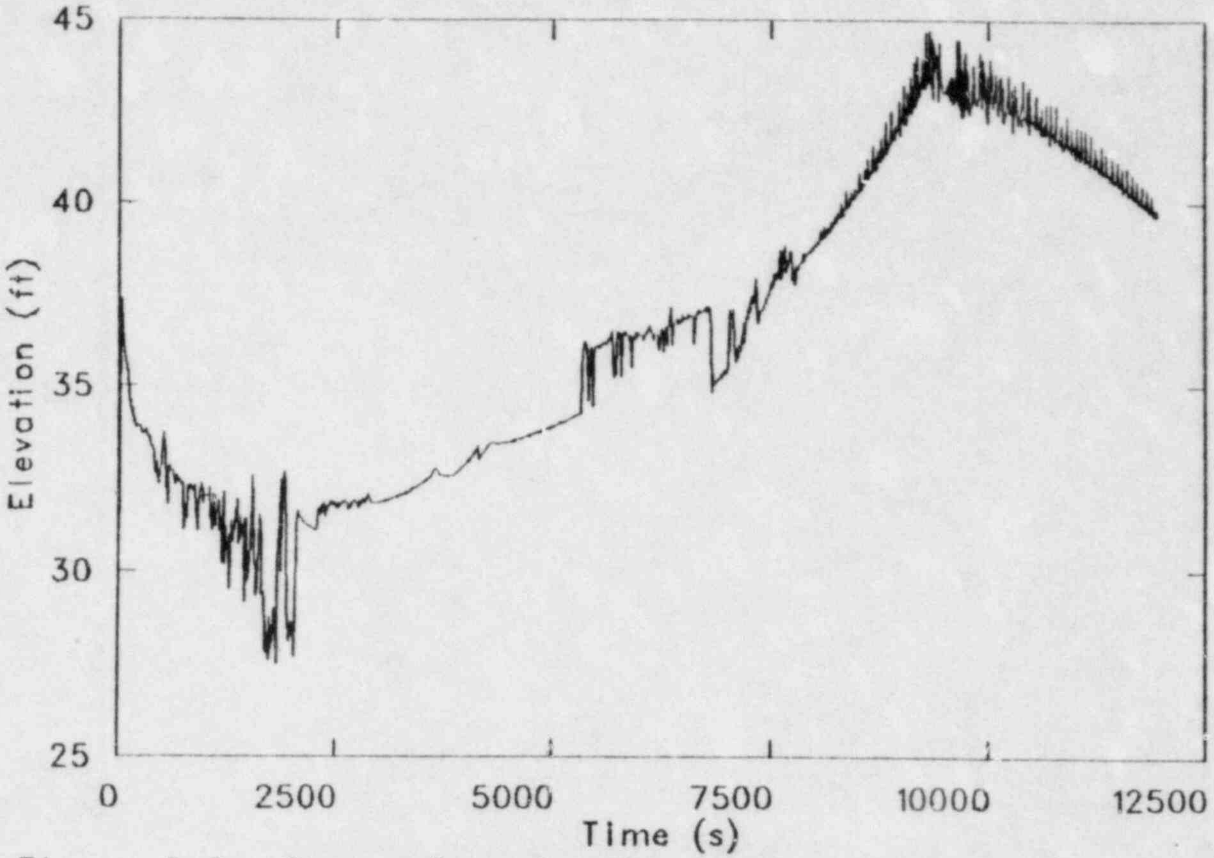


Figure G-2. Core collapsed water level - sequence v8.

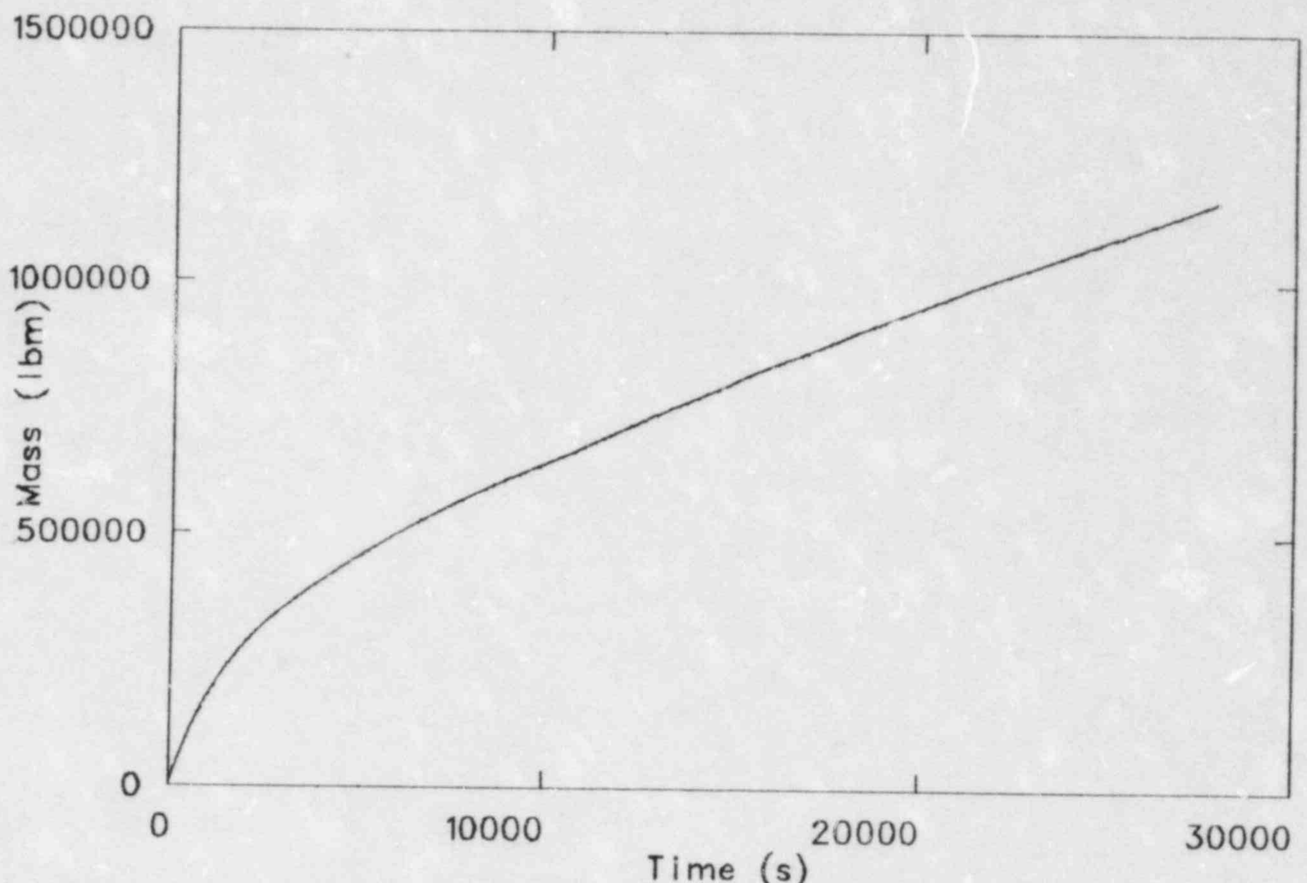


Figure G-3. Total mass transfer to the pressure suppression pool - sequence v8.

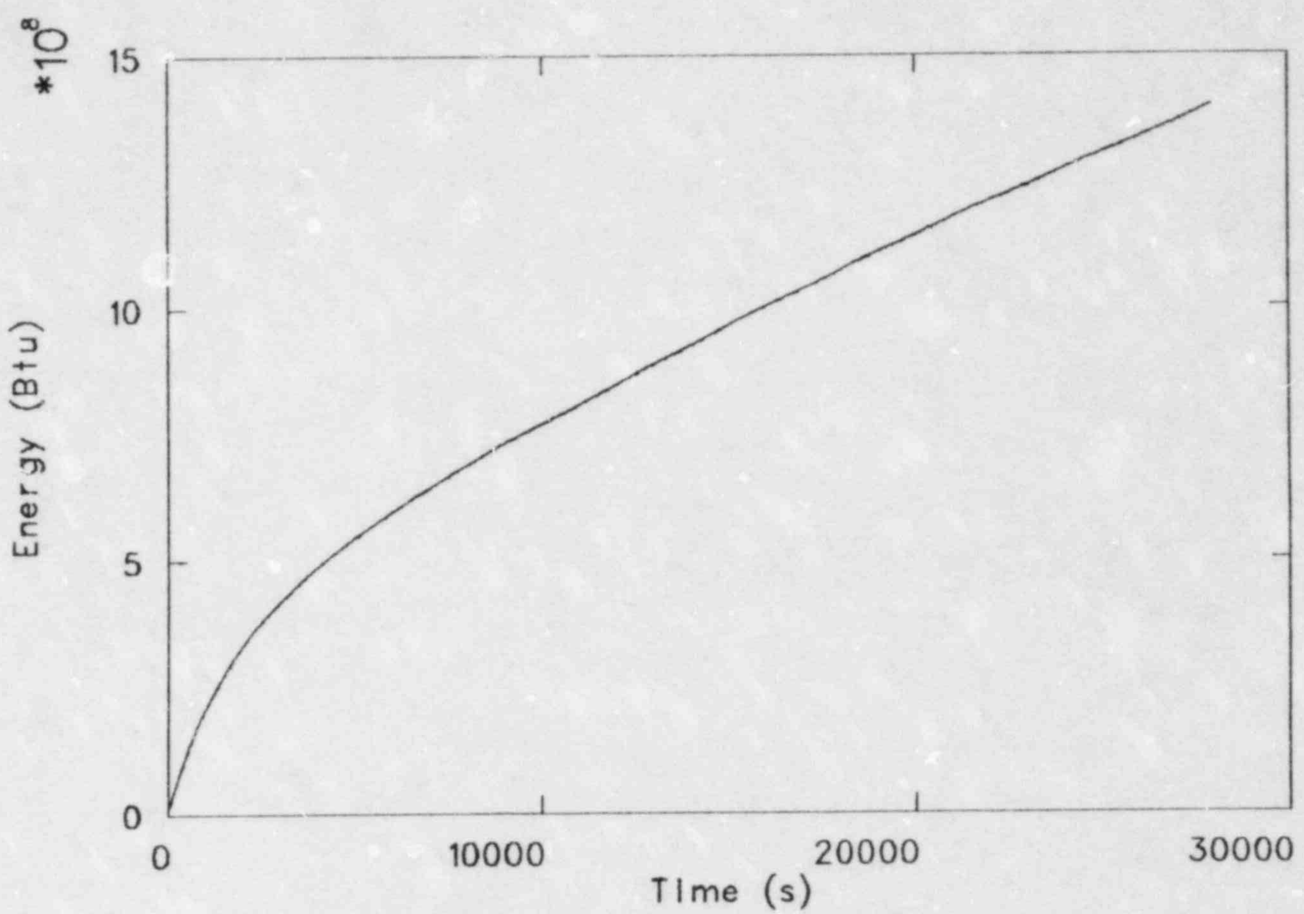


Figure G-4. Total energy transfer to the pressure suppression pool – sequence v8.

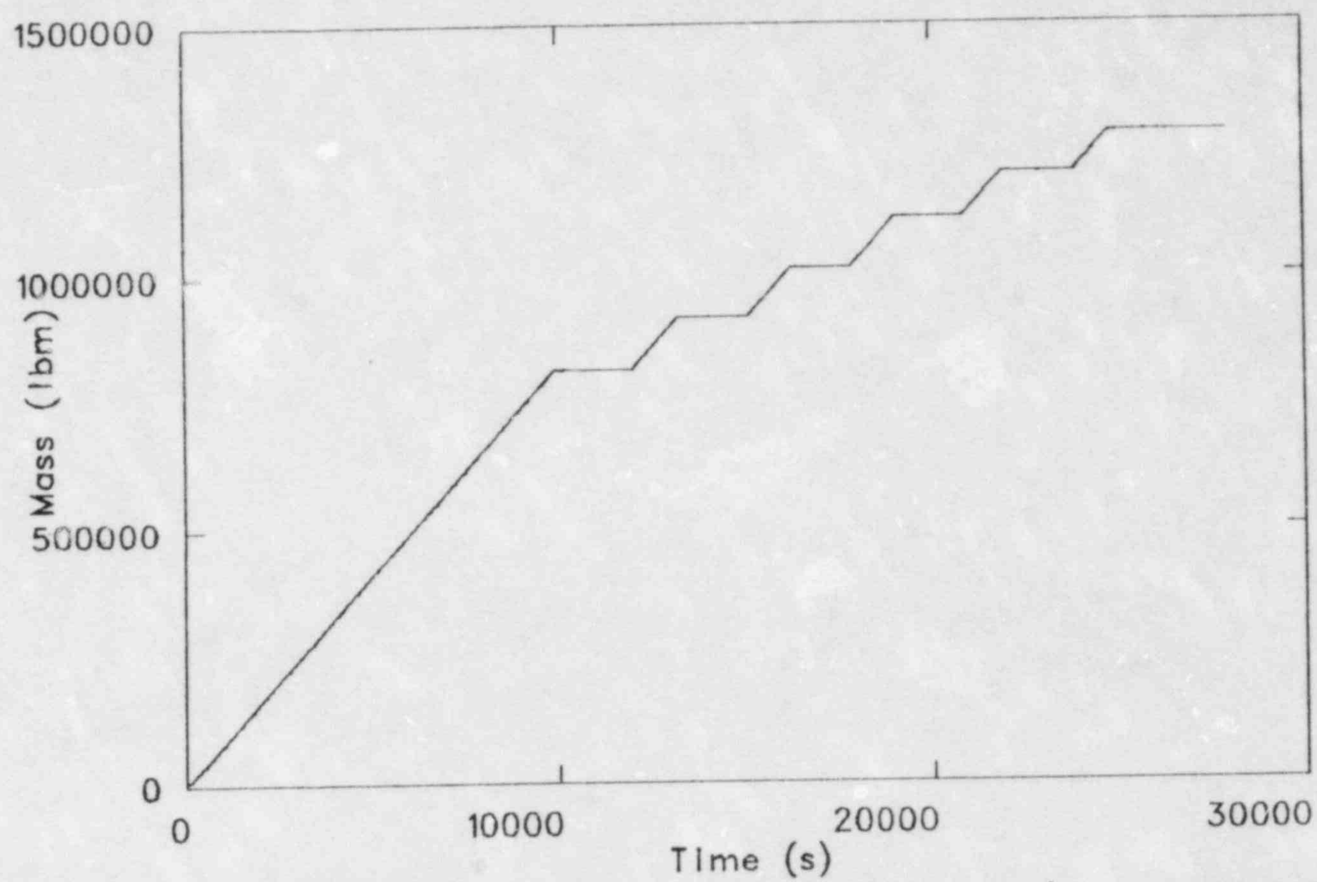


Figure G-5. Total mass transfer to the vessel – sequence v8.

APPENDIX H

THE CONTROLLED DEPRESSURIZATION: SEQUENCE V1

APPENDIX H

THE CONTROLLED DEPRESSURIZATION: SEQUENCE V1

The controlled depressurization is discussed by ORNL (see Reference 3) as a means of reducing the heat load from the vessel to the drywell during a prolonged station blackout. The transient can be divided into four distinct phases:

1. Vessel inventory maintenance at operating pressure--the vessel state is maintained at the operating pressure (see Subsection 4.7.1),

2. The controlled depressurization--the operator depressurizes the vessel from the nominal 1000 psia level to the 90/200 psia level (see Subsection 4.7.2),

3. Vessel inventory maintenance at low pressure--the vessel state is maintained at a low pressure level to minimize the heat transfer to the drywell (see Subsection 4.7.3),

4. Vessel inventory boiloff--the final phase begins at 25,200 s with the DC battery failure. The operator does not have control over the equipment following the battery failure. Thus, the system would repressurize and boiloff over a 15,000+ s time period (see Reference 3). The core would uncover at a time in excess of 40,000 s.

Figure H.1 shows the behavior of the vessel steam dome pressure in Phase 3 of the controlled depressurization transient. The operator is assumed to be opening two safety-relief valves.

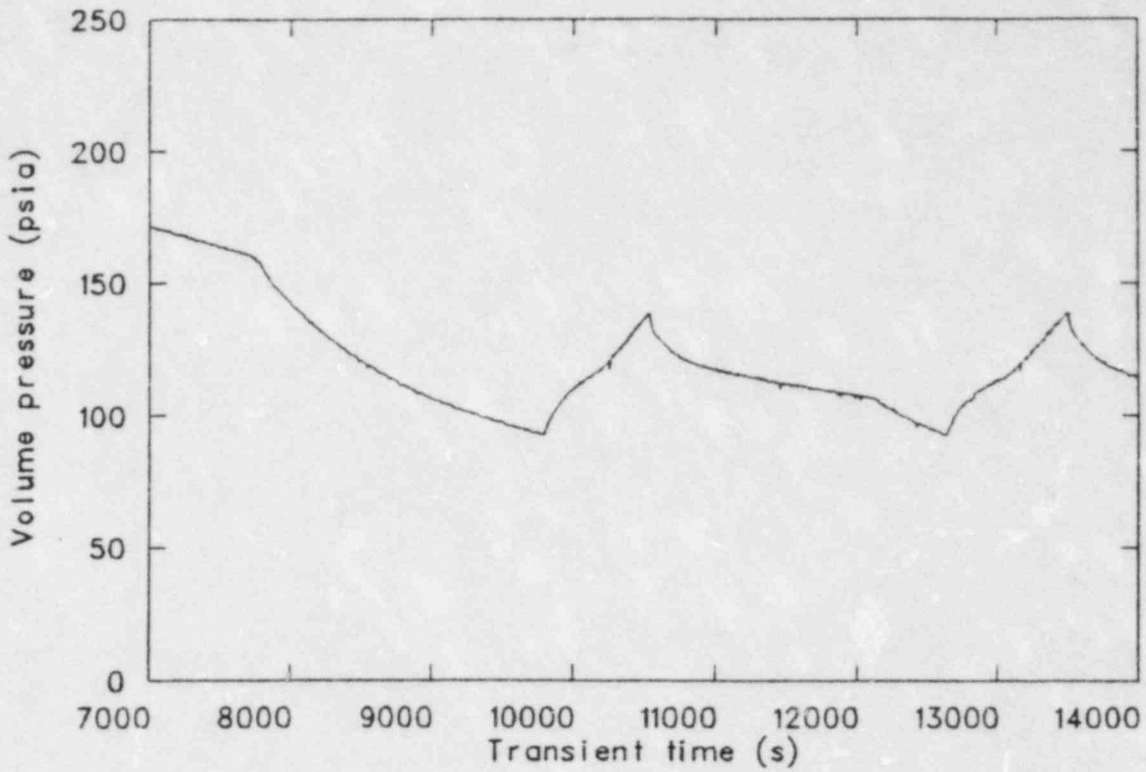


Figure H-1. Vessel steam dome pressure - sequence v8.