MONTICELLO

INDIVIDUAL PLANT
EXAMINATION
(IPE)

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Executive Summary

This Individual Examination (IPE) report is a summary of an extensive study done to meet the requirements of Generic Letter 88-20. This generic letter requires utilities to address important contributors to risk and implement improvement that they believe are appropriate for their plant. The IPE is one part required for closure of the severe accident issues. The severe accident issues require the following:

Individual Plant Examination (IPE)
This includes all internal events and internal flooding which is considered an external event for PRA proposes.

Containment Performance Improvements (CPI)
This is the development of generic containment performance improvements with respect to severe accidents. This has been concluded with the request to put in the hardened piped vent for Mark I BWRs and the rest has become part of the IPE.

Individual Plant Examination External Events (IPEEE)
This is an extension of the IPE to include external events. The main external event are seismic and internal fires, but all external events are to be considered.

Accident Management (AM)
This is a development of a program to use the IPE and IPEEE to enhance the utilities accident management capabilities. This program is still under development.

The IPE is a full scope level 2 PRA consisting of two major parts, level 1 and level 2. The level 1 or front end analysis determined an estimate of the core damage frequency. The level 1 results were then used as inputs to the level 2 or back end analysis. The level 2 analysis determined an estimate of the probability and type of releases which could potentially result from a severe accident.

The level 1 analysis resulted in a total core damage frequency (CDF) of 1.9E-5/yr excluding internal flooding. Internal flooding is estimated to contribute 6.8E-6/yr to the CDF. Figure 1.4-1 and Table 1.4-1 shows the core damage frequency separated by accident class. The major core damage sequences are shown on Table 1.4-2. The results of the level 2 analysis were grouped by release modes and are shown on Figures 1.4-2 through 1.4-6.

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LIST OF ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient(s) Without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
C	Reactivity Control
CAFTA	Computer Assisted Fault Tree Analysis
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CRD	Control Rod Drive
CS	Core Spray
CSF	Critical Safety Function
CST	Condensate Storage Tank
DC	Direct Current
DG	Diesel Generator
DHR	Decay Heat Removal
DLTRM	Delete Term (command in PCSETs)
ECCS	Emergency Core Coolant System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedures
EFT	Emergency Filtration Train
EPRI	Electric Power Research Institute
ESW	Emergency Service Water
FMEA	Failure Modes and Effects Analysis
G	Containment Control - Temperature
HCTL	Heat Capacity Temperature Limit
HELB	High Energy line Break
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection
IDCOR	Industry Degraded Core Rulemaking Program
IORV	Inadvertently Open Relief Valve
IPE	Individual Plant Examination
IPEM	IDCOR's Individual Plant Examination Methodology

LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection (a mode of RHR)
M	Primary Pressure Control - SRVs Open
MCC	Motor Control Center
MG	Motor Generator
MGL	Multiple Greek Letter
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valves
NPRDS	Nuclear Power Reliability Data System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OSP	Offsite Power
P	Primary Pressure Control - SRVs Closed
PCS	Power Conversion System
PCSETS	Personnel Computer Set Equation Transformation System
PCV	Pressure Control Valve
P&ID	Piping and Instrumentation Drawing
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
QA	Quality Assurance
QU	Reactor Coolant Inventory High Pressure
RBCCW	Reactor Building Closed Cooling Water System
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
RHR SW	Residual Heat Removal Service Water
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean-up
SBO	Station Blackout
SCET	Streamlined Containment Event Tree
SLC	Standby Liquid Control
SORV	Stuck Open Relief Valve
SRI	Safety Review Item
SRO	Senior Reactor Operator

SRV	Safety Relief Valve
SW	Service Water
TBV	Turbine Bypass Valves
V	Reactor Coolant Inventory - Raw Pressure
W	Containment Control - Pressure
X	Reactor Coolant Inventory - Depressurization

1. SUMMARY AND CONCLUSIONS

1.1 Background and Objective

In July and August of 1985, the NRC published its policy statement on issues related to severe accidents in NUREG-1070 and 10CFR Part 50. The Severe Accident Policy states that on the basis of currently available information, existing plants pose "no undue risk" to the health and safety of the public. Therefore, the NRC sees no justification to take immediate action on generic rulemaking or other regulatory changes for existing plants because of issues related to severe accidents. The Commission's conclusion of "no undue risk" is based upon actions taken as a result of the Three Mile Island action plan (NUREG-0737), information that resulted from NRC and industry sponsored research, information obtained from published Probabilistic Risk Assessments (PRAs) and operating experience, and the results of the IDCOR technical program.

In November 1988, the NRC staff issued Generic Letter 88-20 which formalized the requirement for an Individual Plant Examination (IPE) under 10CFR50.54(f). This generic letter requires utilities to perform their IPEs, identify potential improvements to address important contributors to risk and implement improvements that they believe are appropriate for their plant. In August 1989, the NRC issued its guidance for utility IPE submittals (NUREG-1335). That document specified the information that should be reported in the IPE submittal as well as a recommended format for the utility reports.

In anticipation of the NRC's staff issuing a Generic Letter on the IPE, NSP initiated an examination of the Monticello Plant in November 1987 using IDCOR's IPE Methodology (IPEM). This decision was made to provide NSP with an interim PRA tool for application to plant design, identify recommendations for safety improvements, and focus work on the full scope PRA. The IDCOR's IPEM analysis was completed late in 1988. A series of recommendations or insights were determined from the IDCOR IPEM. These recommendations were not changed by the IPE study. These insights are listed in Section 6 of this report under their respective accident classes. The plant engineering staff is considering the recommendations and has implemented many of them through procedure changes or modifications. The only modification installed to date is the fire water to

RHRSW crossconnect. Upon receipt of Generic Letter 88-20, NSP elected to fulfill the IPE requirement by performing a full scope level 2 PRA, which is documented in this report.

The IPE of NSP's Monticello Nuclear Generating Plant was performed to develop an improved understanding of the plant response to potential accident conditions and to identify any significant vulnerabilities to severe accidents. The specific objectives are summarized as follows:

- Establish a realistic estimate of the frequency of a core damage event at Monticello.
- Identify the potential accident sequences that contribute to the overall core damage frequency.
- Determine the timing and nature of any radionuclide releases to the environment that might be associated with these dominant accident sequences.
- Identify any dominant accident sequence that occurs with a frequency significantly higher than similar sequences at the other plants that have been judged to be acceptably safe.
- Identify any instance of unusually poor containment performance for these dominant accident sequences.
- Identify cost effective modifications to the plant design, operating procedures, training or maintenance practices that would reduce the likelihood of any accident sequence outliers which are identified.
- Maximize participation in the evaluation process by NSP personnel and maximize the technology transfer from the consultant to NSP to ensure the IPE can be maintained and understood by NSP personnel.

- Provide a well organized and clearly written summary of the Monticello IPE to facilitate communication of the results to both the NRC and NSP, as well as to serve as a tool for communicating the results to interested members of the public.

- Develop the risk based tools and documentation to support resolution of future regulatory, safety, or operational issues for Monticello.

1.2 Plant Familiarization

The Monticello Nuclear Generating Plant is a low power density BWR. General Electric Company designed the plant and supplied the nuclear steam supply system, and turbine - generator unit and its related systems. Bechtel Corporation constructed the plant. The design is identified by General Electric as a "BWR-3" with a Mark I containment. The reactor core produces 1670 MWt with an electrical output of 545 MWe, using 484 fuel assemblies. The plant is located northwest of Monticello, Minnesota. Construction started on June 19, 1967, and full commercial operation began on June 30, 1971.

1.3 Overall Methodology

NSP has elected to perform a full scope level 2 PRA as a basis for the IPE. NSP analysts performed most of the work, using consultants primarily for training, guidance and review.

The level 1 event trees are similar to those used in the IDCOR IPE Methodology and other PRAs which are functionally oriented with functions patterned after the BWROG EOPs. The accident sequence binning is also similar to the IDCOR IPE Methodology and other PRAs. The six accident classes and subclasses are shown in Table 1.4-1.

Level 2 event trees were developed for each of the six accident classes and are also patterned after the functions of the BWROG EOPs. Phenomenological papers were developed for each of the containment failure modes and mechanisms found in Section 7 of NUREG-2300. The phenomenological papers were used to:

- Determine the applicability of the phenomena to Monticello, given specific design features and operating characteristics.

- Identify system success criteria for prevention and mitigation of the various phenomena.

- Assign the phenomena to the containment event tree branches or identify the headings into which the phenomena should be included if appropriate.

There was an extensive data collection effort to develop plant specific initiating event frequencies and component failure rates. This data was used in both the level 1 and level 2 event trees and fault trees. The same analysts that performed the level 1 sequence quantification developed the level 2 models and quantified the CET sequence. Having the same analysts throughout the project ensured the proper integration of the level 1 and 2 analyses. CAFTA software from EPRI and PCSETS software from Logic Analysts were used as the principal tools for fault tree management and cutset generation. MAAP 3.0B was the principal tool used for deterministic best estimate analysis of reactor and containment response during severe accident sequence conditions. Best estimate analysis was performed for both the front end and back end portions of the assessment. Deterministic or probabilistic sensitivity studies were conducted to assess the impact of all key assumptions.

1.4 Summary of Major Findings

The level 1 analysis resulted in a total Core Damage Frequency (CDF) of 1.9E-5/yr excluding flood initiators. Internal flood initiators are estimated to contribute 6.8E-6/yr to the CDF. Figure 1.4-1 shows the Core Damage Frequency (CDF) for Monticello separated by accident class. Refer to Table 1.4-1 for a breakdown of the CDF by accident class.

The top core damaging sequences are listed in Table 1.4-2 along with general descriptions. This list includes all sequences with a frequency greater than once per 10 million years (1E-7/yr).

Containment event trees were developed around the major accident classes of level 1 sequence quantification. CET sequences were binned into categories or plant damage states. Binning of CET sequences with respect to source term characteristics was also performed. The results are grouped by Release Modes as shown below.

Release Mode	Description
A	Containment or reactor vessel intact or controlled vented release at accident termination.
B	Containment failure resulting in releases scrubbed through the suppression pool.
C	Containment failure precedes or is coincident with reactor vessel failure. Suppression pool bypassed.
D	Containment failure is delayed after reactor vessel failure. Suppression pool bypassed.
E	Radionuclides exit containment directly through an unisolated penetration or LOCA outside containment.

Release modes were further subdivided to represent the effects of the operation of containment systems on the magnitude of the release. Three plant damage states were developed and consist of reactor status, containment status, and timing of release with respect to the initiator. Section 4 provides a definition of the identifiers for each plant damage state.

After all the possible sequences were quantified, they were sorted by release mode, reactor failure pressure, containment failure mode, release timing and accident class. These results were then used to create Figures 1.4-2 through 1.4-6 for internally initiated accidents.

No new or unusual means were discovered by which core damage or containment failure could occur. The Monticello CDF for internal events and internal flood initiators is well below the NRC's proposed safety goal of 1E-4/yr.

TABLE 1.4-1

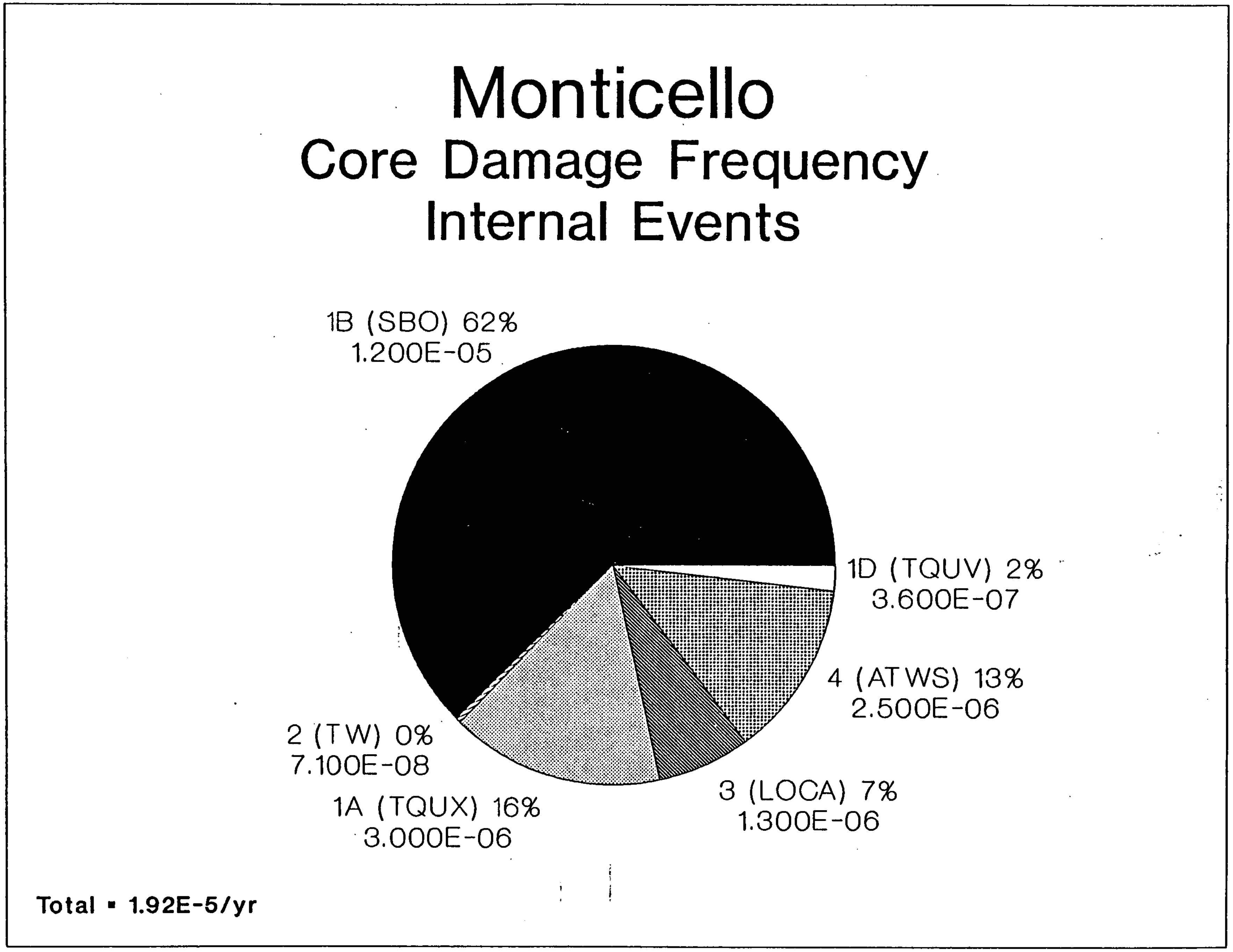
SUMMARY OF CDF RESULTS

ACCIDENT CLASS	DESCRIPTION	CORE DAMAGE FREQUENCY
1A	Loss of Coolant Make-up, High Pressure Core Melt.	3.0E-6/yr
1B	Loss of All AC Power.	1.2E-5/yr
1D	Loss of Coolant Make-up, Low Pressure Core Melt.	3.6E-7/yr
2	Loss of Decay Heat Removal.	7.1E-8/yr
3A	Failure of SRVs to Open Causes RPV Rupture.	1.1E-7/yr
3B	LOCA, High Pressure Core Melt.	4.7E-7/yr
3C	LOCA, Low Pressure Core Melt.	3.9E-7/yr
3D	LOCA, Failure of Vapor Suppression.	2.9E-7/yr
4	ATWS.	2.5E-6/yr
5	Unisolated LOCA Outside Containment.	3.2E-10/yr
--	TOTAL (internal events)	1.92E-5/yr
6	Core Damaging Accidents Initiated by Internal Floods.	6.8E-6/yr
--	TOTAL (with flooding)	2.60E-5/yr

TABLE 1.4-2

MAJOR CORE DAMAGING SEQUENCES

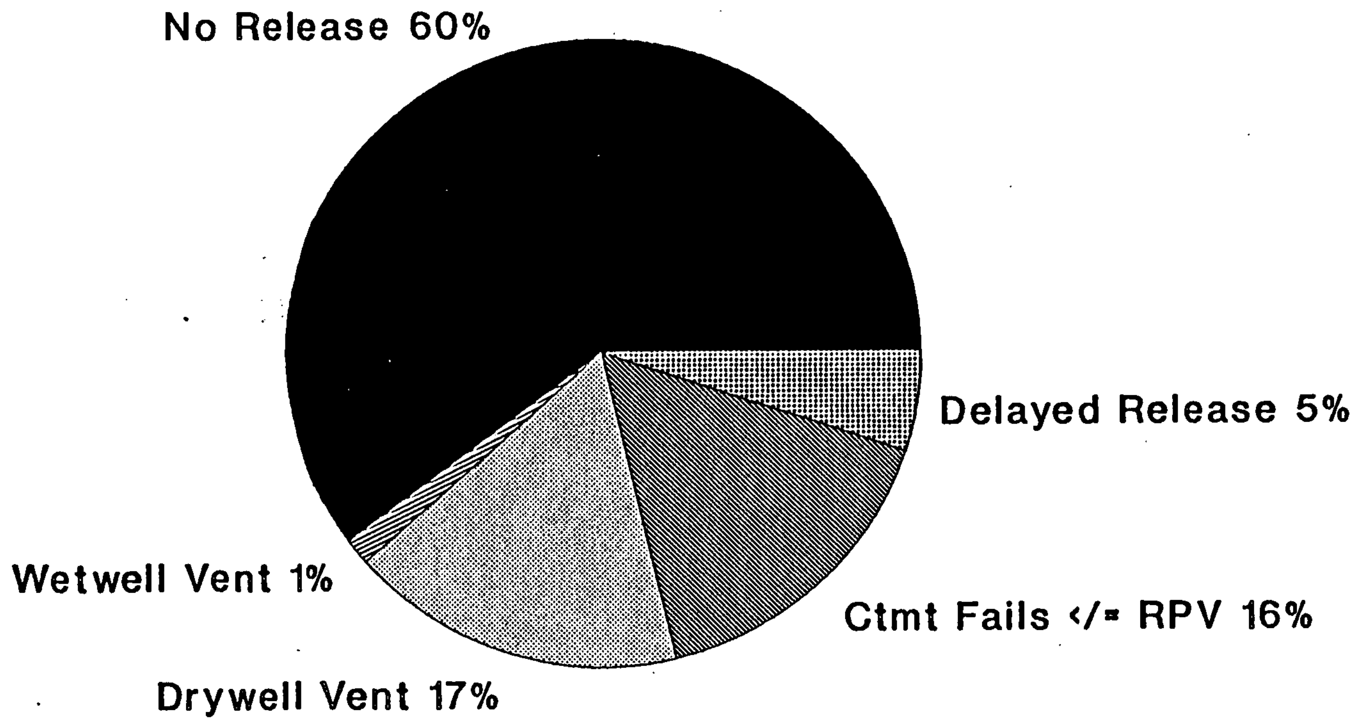
SEQUENCE	DESCRIPTION	FREQUENCY
SBO-LONG	Loss of all AC Power, Unrecovered > 6 Hours.	9.8E-6/yr
SBO-QU	Loss of all AC Power, with High Pressure Coolant Makeup Failure.	1.4E-6/yr
TE,QU,X	Loss of Offsite Power, Failure of Coolant Makeup, Melt at High Pressure.	8.2E-7/yr
TMS,QU,X	Manual Shutdown, Failure of Coolant Makeup, Melt at High Pressure.	8.1E-7/yr
TF,QU,X	Loss of Feedwater, Failure of Coolant Makeup, Melt at High Pressure	6.0E-7/yr
SBO-P	Loss of all AC Power with SORV.	6.0E-7/yr
TT,QU,X	Turbine Trip, Failure of Coolant Makeup, Melt at High Pressure	4.4E-7/yr
S1, V	Medium LOCA, Failure of Low Pressure Coolant Makeup.	3.2E-7/yr
TE,QU,V	Loss of Offsite Power, Failure of Coolant Makeup, Melt at Low Pressure.	2.4E-7/yr
A,D,QUV	Large LOCA, Failure of Vapor Suppression, Failure of Coolant Makeup.	2.1E-7/yr
S2,QU,X	Small LOCA, Failure of Coolant Makeup, Core Melt at High Pressure	1.5E-7/yr
S1,QU,X	Medium LOCA, Failure of Coolant Makeup, Core Melt at High Pressure.	1.4E-7/yr



Monticello
Core Damage Frequency
Internal Events
1B (SBO) 62%
1.200E-05
1D (TQUV) 2%
3.600E-07
4 (ATWS) 13%
2.500E-06
3 (LOCA) 7%
1.300E-06
1A (TQUX) 16%
3.000E-06
2 (TW) 0%
7.100E-08
Total = 1.92E-5/yr

FIGURE 1.4-1

Monticello Level II PRA
Internal Events By Release Mode



(Other Release Modes < 1%)

FIGURE 1.4-2

Monticello Level II PRA
Internal Events By Vessel Failure Press.

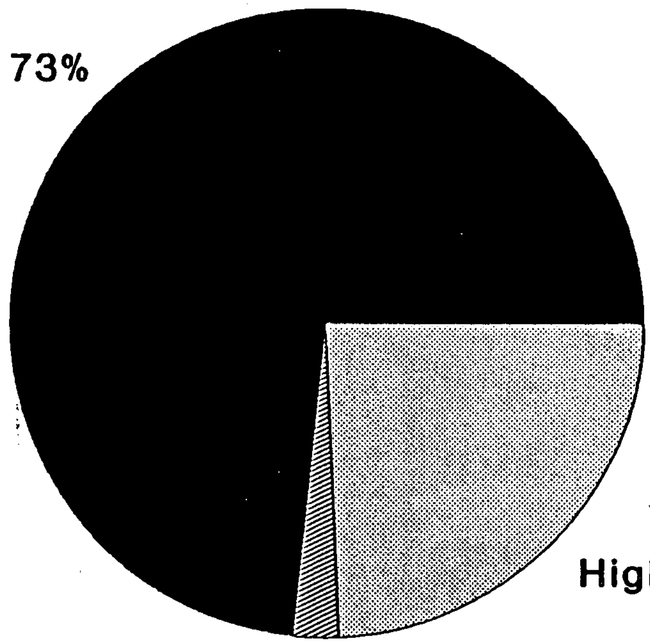
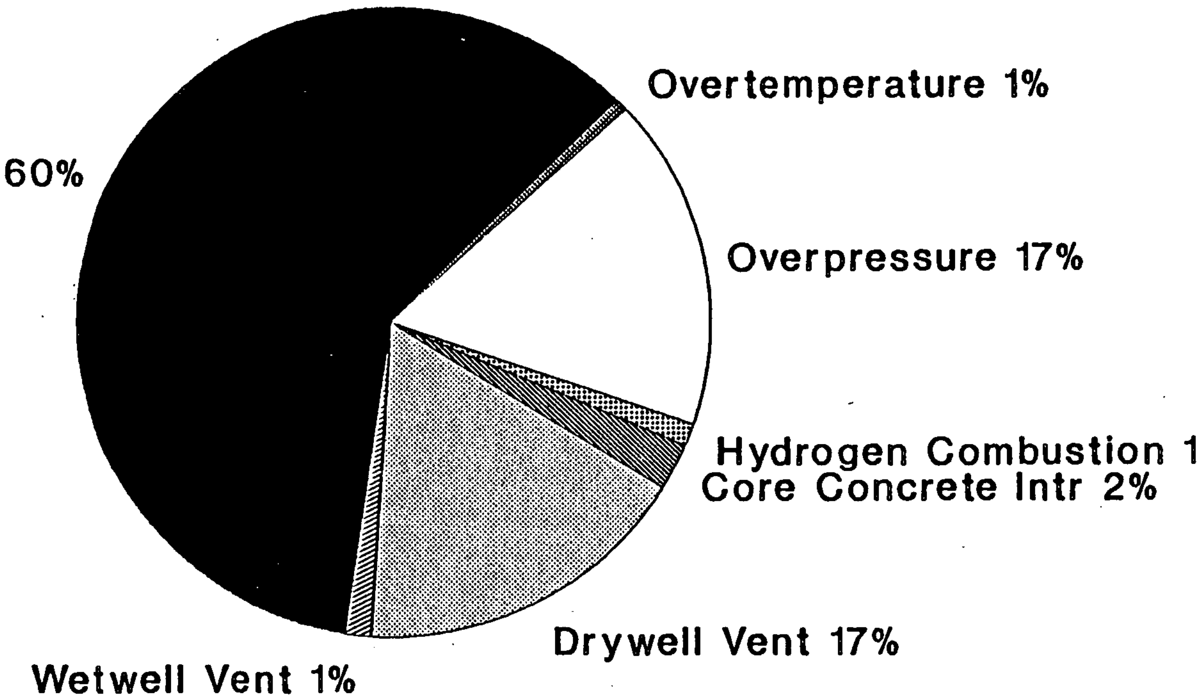


FIGURE 1.4-3

Monticello Level II PRA
Internal Events By Ctmt Failure Mode



(Other Failure Modes < 1%)

FIGURE 1.4-4

Monticello Level II PRA
Internal Events By Release Timing

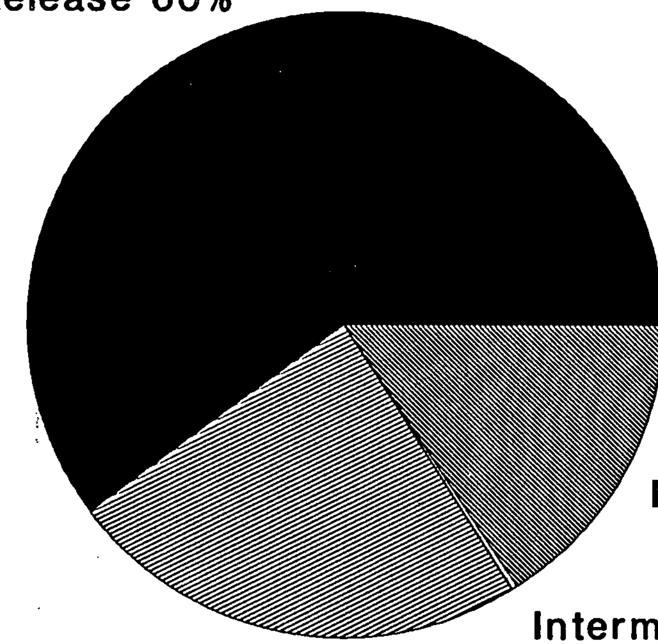
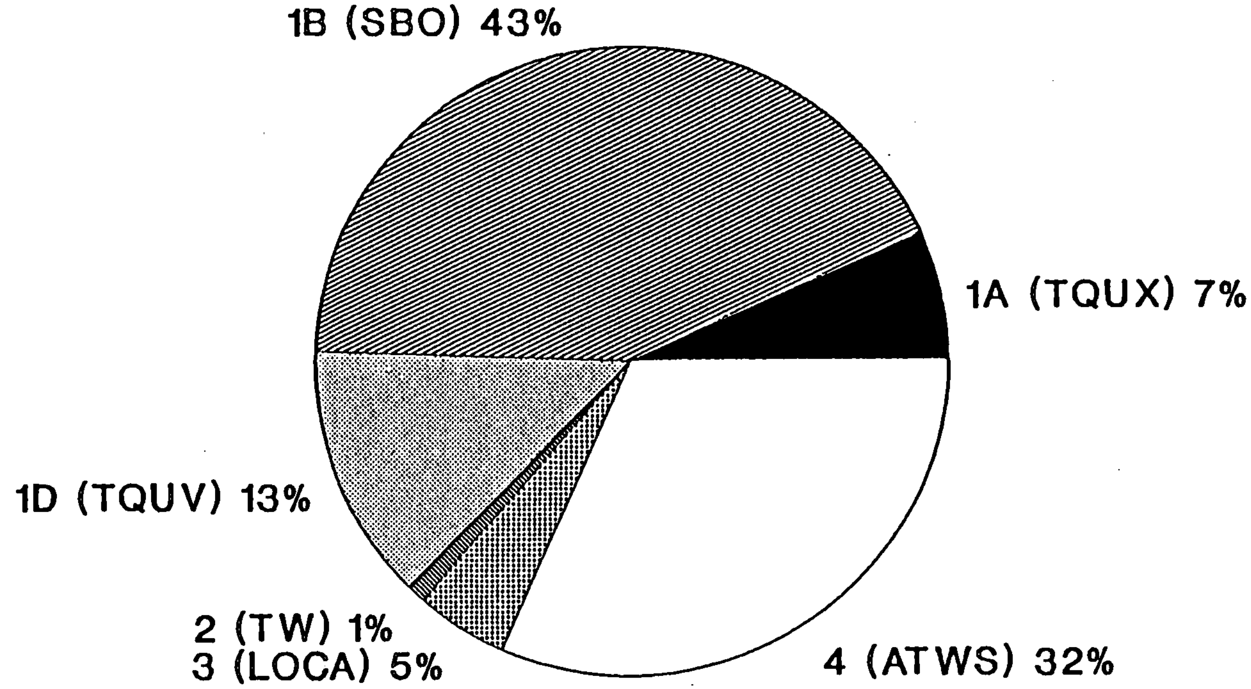


FIGURE 1.4-5

Monticello Level II PRA
By Accident Class



(Other Failure Modes < 1%)

FIGURE 1.4-6

2. EXAMINATION DESCRIPTION

2.1 Introduction

This section describes how the IPE ensures that the primary objectives of generic letter 88-20 are met and that the methods used to perform the IPE conform with the provisions of the generic letter.

The primary objectives of the IPE, as stated by the NRC in the generic letter, are for each utility to: develop an overall appreciation of severe accident behavior; understand the most likely severe accident sequences that could occur at its plant; gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and, if necessary, to reduce the overall probabilities of core damage and fission product releases by modifying hardware and procedures.

The method used for the IPE was a full scope level 2 PRA with containment analysis meeting the intent of Appendix 1 to the Generic Letter.

2.2 Conformance with Generic Letter and Supporting Material

The NSP plant and general office engineering staff have been involved with the IPE process since its inception. They directed all aspects of the analysis with consulting services provided by TENERA, L.P., and Fauske & Associates. This was done to insure the knowledge gained from the examination would become an integral part of plant procedures and training programs and allow future activities to be performed with limited involvement by consultants. Further details of the organization are provided in Section 5.0.

Several comprehensive reviews of the IPE work were performed by NSP personnel in addition to the standard practice of calculation verification. The NSP Quality Assurance Department performed a comprehensive audit at the request of the Nuclear Analysis Department. A review team composed of a multidisciplinary group of plant and corporate staff members reviewed this report prior to publication. Plant system engineers reviewed the system notebooks which formed the foundation of the level 1 analysis. Operations personnel and plant technical staff were trained on the IPEM results which provided an additional review.

The internal events are covered in Section 3. A level 2 PRA was used for the containment release analysis that is presented in Section 4. An analysis of the reliability of decay heat removal (USI A 45) was performed and is documented in Section 3.4. An evaluation of internal flooding was performed and is provided in Section 3.3.8. The general review of results to determine the insights is covered in Section 6.

2.3 General Methodology

2.3.1 Event Trees

The level 1 event trees were functionally oriented, based on critical safety functions used in the EPGs. This allowed for comparison of the level 1 results with the IDCOR conclusions as well as those from other PRAs. The event tree structure includes:

- Reactivity Control
- Reactor Pressure Control - Pressure Relief/Depressurization
- Reactor Level Control - High Pressure Injection/Low Pressure Injection
- Containment Pressure/Temperature Control

The event tree initiators are grouped by similarity of the resulting accident sequences and by their effect on mitigation systems. Event trees used for the analysis are shown in Sections 3 and 4. No support state event trees were necessary in this analysis, since fault tree linking was used to accomplish sequence quantification. Fault tree linking explicitly accounts for the success and failure of frontline systems in the quantification process as well as the interrelationships among frontline systems and support systems.

The level 1 analysis was used as direct input to the level 2 sequence quantifications. The level 2 event trees were functionally oriented. The focus of the level 2 analysis was on containment response to core damage. Since the BWR containment performance is integrally linked with ECCS, many of the same functions and systems appear in the containment event trees. The functions on level 2 event trees are listed below and have been structured to reflect functions in the plant EOPs.

- Containment Isolation
- Reactor Pressure Control
- Reactor Level Control
- Containment Pressure/Temperature Control
- Combustible Gas Controls
- Release Control

Level 2 containment event trees (CETs) are structured around the major accident classes of the level 1 PRA. These CETs were used to determine the containment response and ultimately the release mode, given a core damage event has occurred.

Class 1 (transient) and class 3 (LOCA) CETs represent containment response to events in which core damage occurs with intact containment. The various challenges to containment that might occur as a result of phenomena associated with core melt progression are examined as part of these CETs. The subclasses of 1 and 3 are variations of the same basic CET. Refer to Section 4 for the CETs.

Classes 2 (the containment heat removal failure) and class 4 (ATWS) represent plant response to potentially severe accident events in which containment failure may precede core damage. The only difference between these classes is the relative speed of the event. Class 2 events evolve slowly over several days, while class 4 events are much faster developing and more energetic. The CETs for classes 2 and 4 are shown in Section 4.

Class 5 accident sequences represent bypass of the containment as part of the initiator, and therefore the need for a separate containment event tree is not required.

2.3.2 System Analysis

2.3.2.1 Systems List for PRA by Function

The level 1 PRA functions were discussed in Section 2.3.1. This section will summarize the plant systems analyzed under each function.

PRA Function	Plant Systems Credited	
Reactivity Control	RPS/CRD	Reactor Protection System Control Rod Drive System
	ARI	Alternate Rod Insertion
	SLC	Standby Liquid Control
	RPT	Recirc Pump Trip
Reactor Pressure Control	SRVs	Safety Relief Valves (open and close for pressure relief function)
	SRVs	Safety Relief Valves (ADS)
Reactor Level Control		
(High Pressure Makeup)	FW	Feedwater
	HPCI	High Pressure Coolant Injection
	RCIC	Reactor Core Isolation Cooling
	CRD	Control Rod Drive (low volume makeup)
(Low Pressure Makeup)	Cond	Condensate
	LPCI	Low Pressure Coolant Injection
	CS	Core Spray
	RHRSW	Residual Heat Removal Service Water
Containment Pressure/ Temperature Control	Main Condenser	
	RHR	Residual Heat Removal Modes -Shutdown cooling -Torus cooling -Drywell spray
	Venting of DW or WW	
	RHRSW	Residual Heat Removal Service Water (injection to containment)

A detailed description of each of the above systems can be found in Section 3.1.2.1. These CSFs were used as headings for the event trees constructed for each initiating event category.

2.3.2.2 Success Criteria

Success criteria for each of the systems listed above are summarized in Section 3. The bases for the success criteria were a combination of realistic calculations using MAAP, USAR and operations manual descriptions, and the IPEM. Important system success criteria and responses were evaluated using the Monticello plant simulator.

2.3.2.3 Fault Tree Modeling

The IPE/PRA attempts to represent realistic failure potential for each system in the PRA through development of fault trees. System notebooks were prepared to provide the basis for the system fault trees. Each notebook contains the following information about the respective systems:

Table of Contents
Introduction
System Description
Fault Tree Structure
Support Systems
Instrumentation and Controls
Technical Specifications
Component Location
System Performance During Accident Conditions
Success Criteria
Initiator Impacts on Systems
Impact of Failure of Systems on Other Systems
System Performance
Operating Experience
Assumptions
Initiating Event Review
Summary of Key Results and Insights
Summary of Review Comments
References

One notebook may contain several related systems; for example, RHR contains LPCI, torus cooling and drywell sprays. Both frontline and support system notebooks were assembled for the purpose of developing the fault trees.

Multiple top events were defined for fault trees that served multiple functions. Again, torus cooling, LPCI, and drywell sprays provide an example of such a multiple purpose fault tree. Transfers to other systems were included to account for dependencies on support systems. Support systems were modeled up to the interface with the frontline system or another support system. For example, the service water system model contains only one general model for loss of flow to the common discharge and return headers. This model would be the same for each of the specific loads that the service water system cools; therefore defining the boundary at this point limits duplication of logic between fault trees.

The level of detail is a prime consideration in failure model development. Two criteria were used in developing the Monticello fault trees: the availability of data to support quantification of system components; and the relative importance of failure modes for a given system or component. It is not necessary to model a pump down to the bearings or control circuits if the available data included these types of subcomponent failures and further insights would not result from more detailed fault trees. Faults associated with passive components, such as pipes and manual valves with failure rates that are orders of magnitude lower than the system failure rate, were excluded from the model. The major components that were included in the Monticello fault trees are listed below:

- All major active components - motors, pumps, diesel generators, air compressors.
- All components required to change position to fulfill function (including check valves).
- Instrumentation and controls (I&C) to contact/relay level when the I&C affected the success of an entire system or redundant components in more than one system.
- Removal of equipment from service for testing or maintenance.
- Restoration of equipment out of service for testing or maintenance.
- Human actions necessary to initiate non-automatic system recovery.

With rare exceptions, no passive component failures (e.g. pipe failure) were included.

2.3.2.4 Dependency Treatment

Dependency matrices were also developed as part of system fault tree modeling. These matrices are presented in Section 3.2.3 of this report. The dependency matrices were developed to document the following:

- Initiator effect on frontline and support systems.
- Support system effect on frontline and other support systems.
- Frontline system effect on other frontline systems.

The dependency matrices were used to assist in developing and understanding the results of fault trees. With the use of fault tree linking, the dependencies between systems were explicitly accounted for by the cutset generator during sequence quantification.

2.3.2.5. Quantification Process

The computer program CAFTA (EPRI) was used for managing fault trees. The computer program PCSETS (LAI) was used for sequence quantification. Both were run on 80386-based personal computers.

NSP used the fault tree linking approach as opposed to developing support states or special fault tree models depending on previous success or failure of supporting systems. The failure equations of support systems were linked or "plugged in" to the frontline system fault trees as a part of the sequence quantification. Therefore each frontline system fault tree contains explicit modeling of support system failures that could disable the frontline system. Dependencies of several frontline systems on a given support system are therefore modelled explicitly in the Boolean logic used to combine frontline system failures.

The event tree functional headings labelled critical safety functions (CSFs) were defined by using the Boolean "AND" operator to combine the failure equations of multiple systems which must all fail for the CSF to be unsuccessful. For instance, the CSF equation for low pressure coolant makeup is the combined failure of LPCI, core spray, condensate, and RHRSW crosstied to LPCI.

Core damage sequence cutsets were calculated by "AND"ing together an appropriate initiating event with the failure equations of the CSFs that must fail to reach a particular endstate. Credit for successful CSFs was taken using the DLTRM feature of PCSETS. This eliminated cutsets which would indicate a failure which was already determined to be successful by the event tree. This produced minimal cutset equations for core damaging accidents, often referred to as "Level 1 Analysis", and a core damage probability for Monticello.

The probability and characterization of radioactive release was the subject of the level 2 sequence quantification. The level 1 results acted as the input to the level 2 analysis. Sequence quantification proceeded as described above, by "AND"ing the failure CSFs and using DLTRM for the successful CSFs.

Throughout these analyses, a truncation limit of 1E-9/yr or less was used. This truncation limit is well below the reporting criterion of 1E-6/yr.

2.4 Information Assembly

2.4.1 Design Features

This section provides an overview of the design features, positive (+) or negative (-), significant to the results of the level 1 and 2 PRA. A more complete description of the Monticello plant design features and operating characteristics, and their effects on the results, can be found in Section 6. The first area to be discussed is inventory make-up, which is considered very reliable due to the following:

- motor driven feedpumps which are independent of main steam availability (+)
- condensate pumps are independent of support systems except for AC power (+)
- reliable switchyard configuration (+)
- CRD pump capable of offsetting decay heat within 20 minutes of a reactor trip (+)
- RHRSW and Fire System capable of injection through RHR (+)

The second area was grouped under pressure control. The important features are listed below:

- SRV solenoid power supplies are very reliable (+).
- All SRVs are piped to the suppression pool, none discharging directly to the drywell. (+)
- Long term SRV activation for depressurization is dependant on the availability of AC power. (-)

- The automatic depressurization system is presently inhibited for most scenarios, making it a manually operated system; however, this provides time for recovery of high pressure systems. (-,+)

The third area covers reactivity control, and the important features are:

- To meet the NRC ATWS rule the concentration of the boron has been increased to 55% enriched boron-10 at a concentration of 10.7%. This means that with a 24 gpm pumping rate we can meet the requirement of 86 gpm of sodium pentaborate at 13 wt% concentration of natural boron-10. Either SBL pump has the capability to meet the 24 gpm flow rate. (+)

- The 95% capacity of the SRVs is greater than reactor power even with only one recirculation pump tripped. (+)

- A reactor trip signal does not cause a turbine trip. (+)

- The bypass capacity of the turbine bypass valves is only 15%, limiting the capacity of the main condenser to accommodate reactor power during ATWS. (-)

The last level 1 area to be discussed is station blackout.

- The emergency diesel generators have good reliability, which limits on-line maintenance unavailability. (+)

- The emergency batteries have four hours of capacity. (+,-)

- Two trains of AC independent high pressure makeup are available in the form of HPCI and RCIC. (+)

- No AC independent low pressure injection is available. (-)

The only added feature concerning the level 2 analysis is the fact that the drywell sumps would contain most of the debris coming out of the vessel early in a core damage event. These sumps (6' x 6' x 3' with 2 pumps) are considered to have a positive impact, in that the potential for debris flowing to contact the containment wall is small. Refer to figure 4.4-6 for the layout of sumps, drain lines, pump cavities, and overflows. A potentially negative implication with respect to debris cooling is that the sump depth is 3 feet.

2.4.2 PRA or IPEM Used for Comparison

The Industry Degraded Core Rulemaking Individual Plant Evaluation Methodology (IDCOR IPEM) was used in the initial development stage of the Monticello PRA. The IDCOR IPEM used studies from Shoreham, Limerick, and Peach Bottom as a basis for system modeling and sequence quantification. Appendix D of the IPEM was used to setup and check the fault trees. This was used as a starting point for a more detailed PRA analysis for Monticello. The PRA differs from the IPEM primarily because:

- More detailed component data analysis was done.
- Common cause was added.
- Detailed fault trees were developed and linked in the PRA.
- More detailed CET development with explicit quantification of sequences is used in the PRA.
- More involved internal flooding analysis was done.

As part of initial information gathering, NUREG-1150 (2/87) was reviewed for information specifically pertaining to Peach Bottom, since this plant most closely resembles Monticello. Some of the insights relating to Peach Bottom are listed below:

1. The diversity of high and low pressure injection systems made the probability loss of coolant injection very low. Monticello accident sequence results confirm this insight.

2. Failures of coolant injection systems principally involved loss of support systems, common phenomenological failures due to high containment pressure or temperature, and common cause miscalibration of instrumentation. Support systems were incorporated explicitly in the Monticello models. Transient analysis of each of the functional sequence types was performed to determine the effects of plant conditions on system response.

3. Common cause failures contributed significantly to risk. Common cause failure of the station batteries and the diesel generators were the most significant events. The model for the batteries was based on NUREG 0666. Common cause failure analysis was performed for Monticello plant systems using the multiple Greek letter approach, and common cause factors were included explicitly in the fault trees.

4. Containment venting for DHR was considered.

5. All ATWS events involved MSIV closure and mechanical failure of the rods to insert. The Monticello results suggest that mechanical failure of control rods dominates electrical RPS failures, principally due to ARI. ATWS risk is split between events with and without the main condenser because of turbine trip with bypass available predominates over MSIV closure events in the initiating event distribution.

6. Potential for core debris attack on the drywell wall was considered, but at Monticello the large sumps in the containment limit the potential for this failure mode.

7. Containment failure pressure was assumed to be about 103 psig. The drywell is assumed to be the most likely place the containment fails. A similar containment failure pressure and location was derived for Monticello although the distribution for failure location appears to be roughly evenly split between the drywell upper head and wetwell airspace from the vent bellows.

8. Early containment failure with vapor suppression does not necessarily lead to large releases because of the potential mitigating effects of the suppression pool scrubbing in removing radionuclides or their possible retention in the reactor building. Suppression pool scrubbing as a means of release mitigation was modeled in the Monticello CETs.

During the performance of the PRA, a representative from Nuclenor-Santa Maria de Garona (Spain) visited Monticello. Since the Santa Maria de Garona plant is very similar to Monticello, a comparison of PRA results was performed. The Santa Maria de Garona CDF was quantified at 2.5E-4/yr, much higher than Monticello. If the same assumptions on ATWS and stuck open SRVs were made, the damage class distributions would have been similar. In their ATWS events they didn't take credit for SLC injection; in other words, they would increase their CDF with an ATWS. Nuclenor also had a higher initiating frequency for stuck open SRVs because they assumed the SRVs would not close as pressure decreased. NSP considered that the SRVs would reclose 85% of the time.

The level 1 results of both the IPEM and the PRA were very consistent. The IPEM was able to identify nearly all significant insights identified by the full level 1 PRA. With the exception of the internal flooding analysis, no new insights resulted from the additional level 1 PRA analysis. The IPEM CDF was 1.5E-5/yr, while the PRA study result was 1.92E-5/yr.

The specific insights from the Monticello PRA study are described in Section 6 of this report.

2.4.3 Reference Documents Used

The documents used for this study are listed below along with the general type of information taken from each area.

1. Updated Safety Analysis Report (USAR)
   - Initiating event
   - System success criteria

2. Plant Operations Manuals
   - System descriptions
   - Operating procedures

3. Emergency Operating Procedures: (EOPs Revision 3)
   - System operations during an emergency
   - Operator actions during an emergency

4. Monticello Drawings
   - System components
   - System layout
   - System interconnections

5. Scram Reports, Significant Operating Event Reports, License Event Reports
   - Failure data
   - Plant Response

6. Plant Surveillance Procedures
   - Demand data
   - Test frequencies
   - Run times

7. Work Requests
   - Failure data

8. NPRDS
   - Failure data
   - Pump data
   - Run time

9. Environmental Qualification Report (EQ)
   - Input to equipment survivability

10. SOER 85-05
   - Flooding analysis

A number of means were used to confirm the accuracy of the above documents. Since the system analysts were located at the site, they had ready access to the systems, the system engineers, the operators, and the plant simulator to verify the accuracy of the data. The system engineers were utilized to review and comment on system descriptions, success criteria, and major insights.

2.4.4 Walkdowns

Many types of walkdowns were performed throughout the IPE. First introductory or general walkdowns were completed for all areas outside containment including the reactor building, the torus room, the turbine building, the screenhouse and the simulator. This walkdown included all members of the NSP PRA group, and consultants. The human error analysis walkdown included an NSP analyst responsible for HEP derivation and the consultant responsible for HEP guidelines. The areas covered were the simulator, and areas outside the control room in which operator actions were required. The internal flooding walkdown was done by two members of the PRA group, one who was SRO certified, who looked at flood sources, components, supplies and drains in each area, and the interconnections to adjacent areas. The simulator walkdown included one instructor (SRO licensed), the NSP system analysts (one SRO certified), and a consultant. In the simulator walkdown, all functional accident classes and the important success criteria were reviewed. Periodic walkdowns were also completed by system analysts as required.

3. FRONT END ANALYSIS

This section contains the results of the Monticello Level 1 PRA, beginning with an introduction of initiating events and continuing through the quantification of accident sequences potentially leading to core damage. The contents are summarized as follows:

Section		Summary
3.1	Accident Sequence Description	- Initiating events - Level 1 event trees - Frontline system success criteria - Accident sequence classification
3.2	System Analysis	- Frontline and support system descriptions - Fault tree modeling methods - Dependency matrices
3.3	Sequence Quantification	- Generic and plant specific data - Human actions - Common cause analysis - Sequence quantification method - Internal flooding analysis method
3.4	Results and Screening	- Screening criteria - Sequence results by accident class - Vulnerability screening - Decay heat removal evaluation - Internal flooding evaluationn

3.1 Accident Sequence Description

3.1.1 Initiating Events

3.1.1.1 Plant Specific and Generic Initiating Events

Events which require a manual shutdown or a scram, either manually or automatically initiated, are called initiating events. There are many potential types of initiating events. They include internal events, such as a loss of feedwater, turbine trip, loss of service water and LOCA, as well as external events (e.g., earthquakes, fires, tornadoes, etc.). This report focuses on internal events in accordance with Generic Letter 88-20. Internal flooding was also included in this analysis because it was included in the generic letter. Evaluation of initiators caused by external events will be addressed as a part of NSP's response to the NRC's IPEEE requirement. Table 3.1-1 summarizes the initiating events evaluated in the Monticello IPE and provides the frequency for each initiating event. The plant specific initiating events used in the IPE were:

1. Turbine trip.
2. MSIV closure.
3. Loss of main condenser vacuum.
4. Loss of feedwater.
5. Loss of instrument air.
6. Inadvertent open relief valve.
7. Manual shutdown.
8. Loss of drywell cooling.
9. Loss of RBCCW.
10. Loss of service water.

Generic initiating event frequencies were used for those initiators where plant specific initiating frequencies could not be derived. The following list identifies the generic initiating events used in the Monticello IPE along with the source of the frequency:

1. Large LOCA (EPRI NP-438).
2. Medium LOCA (EPRI NP-438).
3. Small LOCA (EPRI NP-438).
4. LOCA outside containment (Plant specific estimate of pipe lengths, valve failure rates, and generic pipe break frequency from Wash 1400).
5. Loss of a single 125 volt DC bus (Industry experience and generic repair and recovery data).
6. Reactor water level reference line break (IDCOR IPE Methodology, IPEM).
7. Internal flooding (Plant specific estimate of pipe lengths, valve failure rates, and generic pipe break frequency from Wash 1400).

One initiator, Loss of Offsite Power (LOOP), uses a combination of plant experience and generic data to derive its frequency.

Plant-centered LOOP	(Plant specific data)
Weather-caused LOOP	(NUMARC-8700)
Grid-Related LOOP	(NUREG-1032)

3.1.1.2 Initiating Event Frequencies

Transient occurrence data from the period 1/1/72 through 12/31/87 were used to derive the plant specific initiating event frequency estimates. Descriptions of the occurrences from scram reports, LERs, significant operating event reports, and monthly operating data reports were used to classify the events according to transient initiator categories. Transient initiator frequency estimates were derived by dividing the number of events by the number of years of data. Generic initiating event frequencies were obtained from the published sources noted in Section 3.1.1.1.

3.1.1.3 Rationale For Grouping

Although the number of possible individual initiating events is large, the number of significantly different ways in which the plant responds is much smaller. Therefore, initiating events are grouped into categories based on similarities in plant response. The representative event is chosen so that the challenges to critical safety functions; as well as the plant responses to and operator actions following the event, encompass those for other events within the category. The grouping for plant initiating events is:

1. Loss of coolant accidents (LOCA)

   -Small
   -Medium
   -Large
   -Outside containment.

2. Anticipated transients and special initiators

   -Turbine trip
   -Loss of feedwater
   -Loss of condenser vacuum
   -Manual shutdown
   -MSIV closure

-Inadvertent open relief valve
-Loss of drywell cooling
-Loss of RBCCW
-Loss of air
-Loss of a DC bus
-Loss of service water
-Reference leg leak

3. Loss of offsite power.

4. Anticipated transients without scram (ATWS)

-MSIV closure
-Loss of condenser
-Loss of offsite power
-Loss of feedwater
-Turbine trip with bypass
-Turbine trip without bypass
-Inadvertent safety valve operation

5. Internal flooding.

A description of the various groups of initiating events with specific discussion of the rationale for grouping follows:

Loss of Coolant Accidents- A LOCA is defined as any reactor inventory loss which exceeds the plant technical specifications for primary coolant leakage, or that causes a high drywell pressure scram. LOCAs can be separated into break sizes for evaluating the plant response to this class of initiator. In many risk analyses the break sizes are classified according to the requirements for success of the ECCS. This distinction is not related to the licensing basis LOCA sizes but rather as an input into the definition of the success criteria of equipment required for mitigation of the postulated LOCA. LOCA events were grouped separately to reflect unique event tree modeling which included:

- different success criteria for high and low pressure injection systems
- the need for the depressurization function
- the need for the vapor suppression function
- environmental considerations

The Monticello IPE classifications for LOCAs are:

1. Large LOCA - Defined as any break in the reactor system piping which leads to a loss of coolant of sufficient size to:

    a) rapidly depressurize the primary system to the point where low pressure injection systems can operate, and

    b) result in rapid loss of injection capability by the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) systems due to low vessel pressure, and

    c) result in the inability of the condensate system to make up to the reactor, due to depletion of hotwell inventory, prior to establishing effective core cooling or inability to supply makeup to the core due to the break location in the downcomer region.

2. Medium LOCA and Stuck Open Relief Valve - Defined as any break in the reactor system piping which leads to a loss of coolant of sufficient size that:

    a) coolant injection with the RCIC system alone is insufficient, but

    b) the rapid depressurization described for large LOCAs does not occur and the HPCI system is required to maintain reactor coolant inventory until the reactor is depressurized to the point where low pressure systems can operate, and

    c) requires reactor depressurization through the SRVs should HPCI be unavailable in order to enable operation of low pressure systems.

3. Small LOCA - Defined as any break in the reactor system piping which leads to a loss of coolant of sufficient size that:

    a) inventory will gradually be lost from the vessel unless maintained with the aid of a coolant makeup system,

b) feedwater or HPCI operation are sufficient to prevent uncovering the core (but RCIC is not), and

c) the vessel does not depressurize sufficiently from the break for low pressure systems to operate, but requires SRVs for depressurization should feedwater and HPCI be unavailable.

4. Interfacing system LOCAs - These are LOCAs which occur outside of the containment boundary and for which the following conditions may exist:

a) isolation of the break may be possible in order to limit the release of fluid to the reactor or turbine building

b) in the event of an unisolated break, there may be a high environmental stress produced on equipment in the reactor or turbine building, and therefore the operation of ECCS equipment may be compromised

c) the consequences of a core melt in this situation could be significantly different than other situations because of the direct pathway from the primary system to the reactor or turbine building.

<u>Anticipated Transients and Special Initiators</u>- This category includes anticipated transient initiators and support system related initiators. These events include common event tree modeling such as:

- reactor decay heat removal through SRV operation at elevated reactor pressure after the initiating event.
- inventory makeup to accommodate losses due to decay heat
- depressurization should all high pressure injection systems fail
- containment decay heat removal

Transient and special initiators included in the IDCOR IPE methodology (IPEM) were reviewed to develop a preliminary list of initiating events appropriate for consideration in the Monticello PRA. A review of the initiating events from the IPEM indicated that all were potentially applicable to Monticello. However, a review of current plant design and operating experience, including insights gained during plant walkdowns and from application of the IPEM, indicated that a few of the events have little impact on the ability to maintain plant shutdown, or have very low frequencies of occurring. Initiating events which were found to have a low initiating frequency or minimal impact on plant shutdown include loss of instrument nitrogen (because of air system redundancy), loss of reactor building closed cooling water (because it affects only the CRD pumps and drywell coolers), and degradation of an onsite AC power bus (because of low failure probability and redundancy). Further, loss of drywell cooling was not considered to have a significant impact on the Monticello level 1 results because successful reactor and containment pressure control would result in accomplishing containment temperature control, and because loss of containment temperature control was expected to have only limited effect on the operation of core cooling systems.

Loss of Offsite Power- The loss of offsite power initiating event was modelled separately from the anticipated transients noted in the preceding paragraph. The primary factors which required special treatment were consideration of recovery of offsite power and repair of diesel generators. In addition, modeling of time phased event trees was required to account for the effects of station blackout events.

ATWS- This category included the most frequent anticipated transients (with the exception of manual shutdown), and coupled with an electrical or mechanical failure to scram, i.e., failure to insert the control rods following the need for a signal from the Reactor Protection System. The Monticello IPE utilizes a specific set of event trees to investigate ATWS sequences. Modeling unique to the ATWS event includes the ARI and SLC systems and modified success criteria for reactor inventory makeup and heat removal systems.

A relatively frequent anticipated transient for which an ATWS evaluation is not considered necessary is the manual shutdown event. During a controlled manual shutdown the operators will be inserting control rods in a prescribed pattern. If the rods insert as required and the reactor is shutdown, there is no ATWS by definition. If at some point a sufficient number of control rods fail to insert so that the reactor cannot be completely shutdown, the IPE assumes that the operator will be able to maintain the current condition of the plant, i.e., a state in which the plant is producing power at a reduced level. At this point there would still be no challenge to any other safety system and, although not a desirable state, the reactor could continue operation until the operators were able to correct the control rod problem, or some other event occurred which challenged safety systems (i.e., an event occurs which requires a rapid shutdown (scram) and operation of other systems). If such an event occurred, the plant response was modeled by using one of the existing ATWS or anticipated transient event trees, depending on whether scram is successful or not.

Internal Flooding- Internal flooding events used the same basic event tree structure as anticipated transient events. Flooding is a spatially dependent initiator, where the impact on core cooling and containment systems is dependent on the location of the flood. Internal flooding was modelled as a separate damage class in the Monticello internal events IPE.

3.1.2 Event Trees

Figures 3.1-1 through 3.1-12 are the event trees used to represent the Monticello plant response to the transient and accident initiators identified in Section 3.1.1. In this Section, the functional headings of the event trees are defined, as well as important assumptions made in the development of the event trees for each of the initiators.

3.1.2.1 Critical Safety Functions

As mentioned previously, the event trees used for the Monticello IPE analysis were developed around a framework of critical safety functions (CSF) that may be required following any given plant transient. Generally, a CSF can be defined as a condition that when satisfied, limits the potential for breaching (or mitigate challenges to) the barriers to fission product release; the fuel cladding, the reactor coolant system, and the containment. The CSFs can be fulfilled by automatic actuation of plant systems, by passive system performance, or by operator action taken as directed by the plant procedures. Together, the CSFs for the level 1 analysis address a complete set of conditions which must exist to ensure fuel integrity following an abnormal plant transient. In general, the CSFs can be broken down into two basic categories: reactor control, and containment control.

This section provides a general description of each CSF considered in the Monticello level 1 IPE. These CSFs very closely follow the reactor and containment functions contained within the BWR Owners Group Emergency Procedure Guidelines. The CSFs that provide the framework for safe operation of the Monticello plant include the following:

1. Reactivity Control
2. Reactor Pressure Vessel (RPV) Pressure Control
3. High Pressure Coolant Makeup
4. RPV Depressurization
5. Low Pressure Coolant Makeup
6. Containment Pressure Control (Including Vapor Suppression)

Each CSF is described below.

<u>Reactivity Control</u> -During a postulated accident sequence, an important safety function to be performed is the insertion of negative reactivity to bring the reactor subcritical. The primary method for inserting negative reactivity is to scram the reactor by rapid insertion of control rods into the reactor core. For event trees other than ATWS trees, there is no detailed breakdown of this CSF. Initiating events in which rod insertion is assumed to be unsuccessful are transferred to the ATWS event tree for further analysis.

The Reactor Protection System (RPS) and the Control Rod Drive (CRD) system are designed to perform this safety function. The RPS and CRD systems both have a significant level of redundancy which results in a highly reliable reactivity control function. There is no detailed fault tree development of the RPS and CRD system involving rapid rod insertion. Probabilities from WASH-0460 are used instead. For failure to scram events, the mechanical RPS system is backed up by the SLC system. The electrical portion of the RPS is backed up by both the Alternate Rod Insertion (ARI) system and the SLC system. The Recirculation Pump Trip (RPT) system assures that reactor power is quickly reduced to a level where the SRVs can handle the steam load without the main condenser available.

Other event tree headings under reactivity control include alternate boron injection and operator level control. Alternate boron injection with CRD or RWCU is principally considered when the main condenser is available to accept a large fraction of the power being generated in the reactor. Operator level control deals with preventing containment pressurization by lowering reactor level and power during events in which steam can be diverted to the main condenser; and with the ability of the operator to control reactor level and boron concentration after boron has been injected.

Reactor Pressure Vessel (RPV) Pressure Control -RPV pressure control is required to limit nuclear system pressure increases that could result in loss of integrity of the reactor coolant pressure boundary. A number of the transient events considered in the IPE analysis result in the power conversion system being unavailable because of closure of the main steam isolation valves; thus the safety/relief valves (SRVs) are required for pressure control (see RPT discussion as well for failure to scram incidents). Failure of a sufficient number of SRVs to open when required is assumed to lead to excessive reactor vessel pressure and the possibility of a LOCA condition.

In addition, it is desirable that all SRVs involved in these pressure limiting actions reclose after the pressure falls below the SRV setpoints to prevent further release of reactor coolant inventory to the suppression pool after the event has been terminated. A stuck open SRV has the characteristics of a medium LOCA.

High Pressure Coolant Makeup -The high pressure coolant make-up function provides reactor vessel coolant inventory makeup without the need for rapid reactor vessel depressurization. Transients such as turbine trips or small loss of coolant accidents will result in high pressure conditions in the reactor vessel for relatively long periods as inventory losses occur at only decay heat rates.

In general, successful initiation of coolant make-up is required within a relatively short time following a reactor scram from high power, approximately 25 to 35 minutes. Successful operation of a high pressure coolant make-up system during this time frame will often preclude the need for demands on other critical safety functions such as low pressure injection. Otherwise, it is necessary to depressurize the reactor vessel so that low pressure make-up systems can be utilized to recover vessel water level and control coolant inventory. For this study the high pressure coolant make-up systems considered are feedwater, HPCI and RCIC. CRD was also considered and credited for a limited number of sequences.

RPV Depressurization -Depressurization with SRVs in conjunction with the low pressure core cooling systems, serves as a backup to the high pressure coolant makeup systems. Emergency procedures direct the operator to inhibit ADS as inventory falls below low-low reactor level and then manually depressurize the reactor when level is just above the top of the core to reduce reactor system pressure so that low pressure injection can inject water. Successful depressurization is provided by opening relief valves to relieve nuclear system steam to the suppression pool. The relief valves are located on the main steam lines within the drywell. For rapidly evolving events such as large breaks, the reactor vessel depressurizes rapidly through the break and opening SRVs to depressurize the reactor is not required.

Because the emergency operating procedures direct manual control of reactor depressurization, inhibiting ADS and manual initiation of SRVs is assumed for any event in which loss of high pressure injection leading to low reactor level occurs. The ability of the SRVs to open on reactor pressure above the SRV setpoints is not affected by operator actions to control individual valves manually.

Low Pressure Coolant Makeup -The low pressure coolant make-up function is required following depressurization of the reactor vessel through normal cooldown, automatic or manual actuation of the automatic depressurization system, or due to breaks in the primary system that depressurize the reactor vessel below the operating range of the high pressure injection systems. For accident scenarios in which no break in the primary system exists and coincidental failure of high pressure injection occurs, the low pressure coolant injection systems provide adequate coolant inventory makeup once the reactor vessel is depressurized. For LOCAs which cause a rapid depressurization of the reactor vessel, low pressure injection systems are the principal means relied upon to maintain adequate core cooling.

There are a number of systems at Monticello which can perform the low pressure coolant makeup function. These systems can generally be grouped into three categories: systems which are provided as part of the emergency core cooling systems which are safety-related, those alternative systems which are not safety-related but do not require any special actions outside the control room to align, and those systems that require alignment from outside the control room. For this study the low pressure coolant makeup systems considered are:

1. Emergency Core Cooling Systems (ECCS)
   -Low Pressure Coolant Injection (LPCI)
   -Core Spray (CS)

2. Non-safety Related Systems Available from Control Room
   -Condensate System

3. Systems Available from Outside Control Room
   -RHR Service Water System cross-tied to LPCI

Containment Pressure Control -Containment pressure control is an additional functional requirement for safe shutdown. Successful maintenance of containment pressure at or below primary containment pressure limits assures the integrity of the containment structure and permits continued operation of equipment in the vicinity of the containment. Successful containment pressure control during transients is accomplished through operation of decay heat removal systems. LOCAs require an additional means of pressure control through vapor suppression in addition to decay heat removal.

The normal method of removing decay heat from the reactor vessel is through the main condenser. Heat removal from the main condenser includes the entire circuit from the reactor vessel to the main condenser, via the main steam lines and the turbine bypass lines, and back to the reactor vessel via the condensate and feedwater lines.

If the main condenser is unavailable for decay heat removal or is isolated from the reactor vessel, the Monticello containment is designed to accommodate substantial heat addition due to decay heat while alternate means of containment heat removal are aligned. Following operating events which isolate the main condenser, the decay heat is transferred to the suppression pool. The suppression pool is a large pool of water located in the torus, containing approximately 70,000 cubic feet of water. The water in the suppression pool is intended to condense steam in containment following a LOCA, an operation of ADS, an opening of SRVs, or exhaust steam from operation of the HPCI or RCIC steam turbines. The size of the pool is such that it can accommodate decay heat for one to two days prior to pressurization to the containment design limit.

The residual heat removal (RHR) system is used for containment heat removal whenever the main condenser is unavailable or suppression pool cooling is required. There are four operating modes of the RHR system which may be used for containment heat removal: suppression pool cooling, shutdown cooling, wetwell sprays and drywell sprays. The suppression pool cooling mode of RHR removes decay heat from the torus. The shutdown cooling mode of RHR permits heat removal directly from the reactor vessel. Suppression pool cooling can be placed in service without regard to reactor vessel pressure whereas shutdown cooling is permitted only after reactor depressurization has been accomplished. Wetwell and drywell sprays may be manually initiated on containment pressurization in accordance with the Emergency Operating Procedures. The pump suction is from the torus for wetwell and drywell spray modes.

In the event the main condenser and the various modes of the RHR system are unavailable to remove decay heat, the containment pressure will rise and other means of maintaining containment pressure must be relied upon. The other method available for containment pressure control to supplement the main condenser and RHR systems is containment venting. EOPs instruct the use of the Containment Atmospheric Control System for venting, beginning with small 2" lines and using 18" vent and purge lines, if necessary. Should it be required, venting is initiated preferentially from the suppression pool, backed up by the drywell vent lines.

In summary, successful containment pressure control can be is assumed for the Monticello IPE with any of the following systems:

1. The main condenser.

2. The Suppression Pool Cooling Mode of the RHR System.

3. The Shutdown Cooling Mode of the RHR System (only if RPV pressure is low).

4. Wetwell or drywell spray mode (on high containment pressure)

5. Containment Venting.

Other potential methods of containment heat removal which are available at Monticello but are not included in the level 1 analysis are the reactor water clean-up system, the spent fuel pool heat exchanger, drywell coolers, containment sprays using RHRSW, and HPCI/RCIC recirculation to the condensate storage tank. These systems either have relatively low heat removal capacities as compared to decay heat loads immediately after reactor shutdown, were considered too dependent on other modes used in the analysis, or only act to add more water to the containment which increases the heatsink capacity in the torus but does not actually remove decay heat.

<u>Vapor Suppression</u> -The vapor suppression function is accomplished through the use of the suppression pool and the vacuum breakers. The primary purpose of the vapor suppression system is to provide primary containment overpressure protection following a (LOCA) or stuck open safety/relief valve (SORV). Additionally, the vapor suppression system is the heat sink for automatic or manual reactor vessel depressurization, or steam exhausted from the HPCI or RCIC steam turbines. The vapor suppression system performs this function by condensation of the steam which is released from the reactor vessel due to any of the mechanisms mentioned above. The vapor suppression system consists of the suppression pool, a drywell downcomer system which directs steam and non-condensible gases from the drywell into the wetwell, SRV discharge line T-Quenchers and vacuum breakers, and a vacuum breaker system which equalizes pressure between the wetwell and the drywell. Steam from a LOCA is directed to the suppression pool through the downcomers and the vacuum breakers prevent excessive differential pressure between the drywell and wetwell during drywell steam condensation.

For successful vapor suppression to occur following a LOCA, the steam flow exiting the drywell to the suppression pool discharges into the suppression pool through eight vent pipes. The vacuum breakers are designed to prevent flow from the drywell to the wetwell airspace, such flow bypasses the vapor suppression function. If vacuum breakers were to fail open for any reason, the steam flow would discharge into the wetwell airspace and may not be condensed effectively. Condensation, at the pool surface and on heat sinks through the containment, would occur. Successful operation of the vapor suppression system therefore requires closure of seven of the eight drywell to torus vacuum breakers during a LOCA. Failure of vapor suppression is not considered to be likely during non-LOCA sequences because of the extremely low probability of a break of the SORV discharge line in the wetwell airspace coincident with blowdown through the SRVs.

Operator action to initiate reactor depressurization through the SRVs to the pool can be effective in maintaining the vapor suppression function should vacuum breakers be open during a LOCA. It also may be possible to control the pressure in containment if the drywell or wetwell sprays are used for containment pressure reduction following a LOCA.

3.1.2.2 **Front-Line Event Trees**

An event tree established for the IPE study is a model used to determine possible combinations of system or operational failures which may result in core damage. Event trees were constructed for each initiating event category. The event trees developed for the Monticello IPE are similar in form to those included in the BWR IPEM. Brief discussions of the CSFs and each event tree type are provided in this section.

Four general types of event trees are used to analyze the plant response to various initiating events:

- Anticipated transients and special initiators.
- Loss of offsite power.
- Loss of coolant accidents (LOCAs).
- Anticipated transients coupled with a failure to scram (ATWS).

A review of the Monticello plant design and operating experience indicates that the above general types of event trees accurately reflect the plant response for any plausible initiating event. It was concluded that there were no other anticipated transients or other initiating events which occurred or might occur at Monticello which exhibit significantly different characteristics of plant response. The level 1 event trees used in the Monticello IPE are described below.

Anticipated Transients and Special Initiators -The event tree used for the evaluation of anticipated transients and special initiators is similar to that included in the IPEM and is shown in Figure 3.1-1. The form of the event tree is the same for each of the following events:

- Turbine trip
- MSIV closure
- Loss of feedwater
- Loss of condenser vacuum
- Manual shutdown
- Reference leg break
- Loss of DC
- Loss of instrument air
- Loss of service water
- Internal flooding

The headings on this event tree correspond to the critical safety functions (CSFs) which were described in Section 3.1.2.1.

Loss of Offsite Power (LOOP) -Because of the pervasive effect of offsite and onsite AC power on frontline and support systems, the LOOP event tree models are significantly different than other event tree models. The LOOP event tree models in the IPEM were modified to accurately reflect the Monticello offsite and onsite AC power design. The resulting LOOP event trees for the Monticello IPE are shown in Figures 3.1-2 through 3.1-5.

Once the status of the power sources is determined (e.g., offsite power available, 2 DGs or 1 DG available), the evaluation of the LOOP events continues to determine the status of the various vessel water level inventory control systems including HPCI, RCIC, ADS, and low pressure injection systems. The sequences modeled range from successful operation of the emergency power system in the short-term, to extended station blackout sequences in which neither onsite nor offsite AC power is available for several hours.

To assure completeness of the modeling, any sequence in which offsite power is recovered is transferred to an event tree developed in a manner similar to the MSIV closure initiating event frontline system tree. This is because plant response once offsite power is recovered is similar to situations modeled for a closure of the MSIVs. The main condenser will be lost early in a LOOP event due to the loss of power to main condenser support systems. With offsite power recovered the actions required of the operator and the various plant systems are the same whether the initiating event is as a result of a MSIV closure or a LOOP event in which the main condenser is temporarily lost. However, although the sequence of events is the same, the probabilities of system unavailability may differ (e.g., the feedwater pumps must be restarted after recovery of offsite AC power if they are to be used).

Sequences for which offsite power is not recovered but onsite power is available also transfer from the LOOP event trees into one of two other event trees that are similar to the MSIV closure frontline system event tree. These sequences include any in which at least one train of emergency AC power is available and vessel water level is successfully maintained by either the HPCI/RCIC systems or by low pressure systems following vessel depressurization. System availabilities applied to these event trees differ depending on whether one or two diesel generators are available.

Loss of Coolant Accidents (LOCAs) -The LOCA event trees used in the Monticello IPE are similar in form and content to those used in other published BWR PRA's, and are shown in Figures 3.1-6 through 3.1-10. They are different from the transient and special initiator event trees in several ways. Vapor suppression is considered because coolant inventory would be entering the containment resulting in a potentially rapid increase in containment pressure. For medium LOCAs and SORV, the reactor will eventually depressurize on its own and low pressure injection systems must be available even if high pressure injection is initially available.

Anticipated Transients Coupled with Failure to Scram (ATWS) -The following initiators were considered as input to the ATWS event trees.

- Turbine trip
- MSIV closure

- Loss of main condenser
- Loss of feedwater
- Loss of offsite power
- IORV/SORV

The frequency of these initiators, when coupled with the probability of failure to insert control rods, resulted in ATWS initiators which had very low likelihoods of occurrence, but with a potential challenge to containment as a result of the failure to scram in addition to the demand for core cooling systems.

The Monticello ATWS event trees are shown in Figures 3.1-11 and 3.1-12. Two event trees were used to model the sequence of events following each transient initiator considered, one tree for reactor trip with turbine trip, and one for reactor trip without turbine trip.

3.1.2.3 Assumptions

Assumptions about plant behavior for event tree development follow:

1. The event trees were based on current plant design, operational practices, and procedures. Monticello plant specific emergency operating procedures based on Revision 3 of the BWR Owners Group Emergency Procedure Guidelines were used to evaluate operator actions expected during transient and accident events. The plant will be switching to Revision 4, in March, 1992, after the present requalification cycle. Revision 4 will be included in the next PRA update.

2. With few exceptions, the plant evaluation and model quantification did not take credit for nonproceduralized operator actions.

3. The Monticello plant is conservatively assumed to be operating at 100% power at the beginning of all transients considered in this evaluation, including those which develop into transients without scram.

4. A mission time of 24 hours was used for time dependent component failure rates, time frames for recovery were considered assuming that system failures occurred at T=0.

5. The end state of any sequence in the level 1 event trees was either a safe stable condition with the core cooled and the containment intact - designated "OK"; a damaged core; or a failed containment with an intact reactor core. The last category occurs only for class 2 loss of decay heat removal accidents. For this accident class, further development was carried out with a supplemental event tree to determine if core damage occurs after containment failure. The level 1 results reported in this document are the class 2 core damage sequences carried through both the primary and supplemental event trees; and the remaining accident class sequences which resulted in core damage on the level 1 event trees. The supplemental event tree for class 2 is included in Section 4 of this report, "Back End Analysis", because it includes structure to model containment response as well as structure to model core cooling following containment failure.

6. The effects of spatially dependent external events such as fires, seismic events, tornados, etc. are not included in the Monticello IPE models. Internal flooding was evaluated.

7. Repair and recovery actions were considered on a sequence by sequence basis depending upon the sequence timing, operator interviews, operating experience, procedures, and judgment of the analysts.

8. Reactor Level Control Function:
   Core uncovery and core damage were assumed to occur at approximately 30 minutes following failure of high pressure injection systems after a scram based on MAAP analysis.

9. Depressurization Function:
   If high pressure injection is available, depressurization will occur successfully as a result of a SORV or medium LOCA allowing low pressure injection without the need to manually initiate SRVs.

10. Containment Pressure Control Function:

Loss of containment decay heat removal and ATWS events will eventually overpressurize the containment resulting in containment failure. Core damage may occur resulting from the adverse reactor building environment following containment failure causing loss of injection. If containment pressure control has failed and high pressure injection is available, repair and recovery actions are terminated at 56 psig (one to two days for loss of decay heat removal events) subsequent to containment heat removal failure. If high pressure injection is not available, containment pressure is assumed to continue to increase until the SRVs close (approximately 70 psig). The vessel will then repressurize and any injection from low pressure systems will stop. Core damage resulting from either containment failure or lose of injection was analyzed with MAAP. Core damage from containment failure was ultimately assumed.

11. Loss of Offsite Power Initiators:

Recovery of offsite power at 24 hours is considered for use of the main condenser as a heat sink for loss of offsite power events.

12. Station Blackout:

During station blackout, battery depletion occurs after 4 hours. If the reactor is at high pressure when batteries deplete, core damage occurs at 6 hours. If the reactor is at low pressure when the batteries deplete, core damage occurs at 8 hours, the difference resulting from the additional time required to reheat the primary system.

13. ATWS Initiators:

For the ATWS initiating event, the probability of the turbine tripping when a scram signal is received was evaluated. The RPS logic does not include an automatic turbine trip when a scram signal is generated. The significance of this is that for a large fraction of reactor trips, power operation will continue, giving the operator an indefinite amount of time to take corrective actions. For turbine trip ATWS events, a significant number of scrams would not have caused a turbine trip based on a review of plant scrams. Out of 29 turbine trip initiators considered for the IPE, 13 scrams would not have caused a turbine trip. There were 16 turbine trips during the data sample period and 13 miscellaneous reactor scrams that behaved like turbine trips. Therefore, the IPE model was developed assuming 29 turbine trips had occurred. The availability of the turbine as an energy sink is important for ATWS, so the ATWS turbine trip initiator included only the 16 actual turbine trips. Another ATWS initiator called "reactor trip without turbine trip" covers the 13 miscellaneous scrams that were classified as turbine trips for non-ATWS events.

14. The ATWS/Failure to Scram sequences listed below were not explicitly included in the quantification because their probability when combined with RPS initiation event was less than 5E-7/yr.

- Loss of DC
- Reference Leg Leak
- Loss of Instrument Air
- Small LOCA
- Medium LOCA
- Large LOCA

15. Interfacing LOCA Initiators:
The interfacing LOCA evaluation used the following assumptions:

- The break probabilities are based on WASH 1400 pipe failure rates. They also include the effect of the isolation valves failing, when appropriate.
- HPCI, CRD, and RCIC are all assumed to be unavailable subsequent to an interfacing LOCA. The availability of other injection systems is dependent on whether they are the source of the LOCA and the environment in which they are required to operate. The environment could also affect the RHR and core spray injection valves and pumps. The RHRSW pump, which can also be used for injection, is located in the intake structure and is assumed to not be affected by this LOCA.

3.1.3 Success Criteria for Frontline Systems

Table 3.1-2 summarizes key frontline system success criteria for a representative group of accident initiators. In order to simplify the table, the critical safety functions considered have been condensed into two very general categories Coolant Injection and Containment Heat Removal.

The frontline system success criteria shown in Table 3.1-2 were derived from plant specific Monticello analysis of system response to transient and LOCA initiating events. The basis for the success criteria was a combination of realistic calculations using MAAP, simulator verification of the dominant sequences, USAR and operations manual descriptions, and the IPEM. A summary of transient analyses performed for the level 1 portion of the Monticello PRA is provided in Section 7.1.

Each of the critical safety functions were successfully accomplished when any of their corresponding frontline systems successfully operated. Successful operation of these systems was defined in terms of the physical processes which were being performed. For example, successful operation of coolant injection systems involved providing enough water to the reactor core to prevent core uncovery for an extended period of time. Because risk assessments usually examine the response of specific plant systems, it is convenient to translate these physical process parameters into equivalent system performance characteristics.

Thus, successful coolant injection can be defined as providing a minimum flow rate of water to the core. For coolant injection systems, the minimum flow rate requirements can be expressed as the flow provided by a certain pump or a number of pumps from a given system. The criteria for operational success of each frontline system may vary with the type of initiator that results in the need for frontline system operation. For example, the inventory makeup requirements to prevent core uncovery for a small break in the RPV pressure boundary are less than for larger breaks. Also, if a system serves more than one function, it's success criteria may be different for each function.

The frontline system success criteria for ATWS sequences are summarized in Table 3.1-3.

3.1.4 Support System Modeling

Fault tree linking was used to account explicitly for support system interdependencies in the IPE. Fault trees for all support systems were developed. The support system fault trees were then linked to the frontline and other support systems where required. The frontline systems were then combined with other frontline systems as dictated by the event trees using PCSETS. No specific support states were developed for the Monticello IPE. However, the fault trees were quantified in steps of increasing complexity and support system requirements. They produced a hierarchy support systems which provided insights in the ways support systems interact.

3.1.5 Accident Sequence Classification

This section discusses the method used to group core damage sequences into categories based upon characteristics of the accident sequences. These core damage sequence categories are called "Accident Classes" and serve as input to the level 2 evaluation.

The potential types and frequencies of accident sequences at a nuclear power plant cover a broad spectrum. In order to limit these sequences to a manageable number, sequences with similar characteristics (e.g., similar initiating events, primary system conditions, and containment conditions) were grouped together. Table 3.1-4 illustrates this grouping process as practiced in many PRAs and as performed in the IPE Methodology. This process was employed for the Monticello IPE as well.

The accident sequences leading to core damage are categorized into classes and sub-classes. Grouping of similar core damage sequences into classes is performed based upon the following criteria:

- Integrity of the containment
- Integrity of the primary system
- Relative timing of the core melt at the time of core melt
- Primary system pressure
- Critical functions which failed thereby leading to core damage

Based on the above, an accident sequence classification into five classes has been performed. These five accident sequence classes are described in Table 3.1-4.

The five classes are further divided into sub-classes based upon the unavailability of key functions. Table 3.1-5 provides a description of those sub-classes.

In summary, the event tree sequence end states are either a safe shutdown condition or one in which core damage has occurred. As noted, a wide spectrum of possible core damage states exists. The core damage sequences are categorized into five classes plus associated subclasses to provide a discrete representation of this spectrum. The core damage classes provide the entry conditions to the containment event trees and source term evaluation. They also establish the boundary conditions for quantifying the radionuclide releases.

Table 3.1-1
INITIATING EVENTS

CATEGORY	INITIATING EVENT	DESIGNATOR	FREQUENCY (PER RX. YEAR)	SOURCE[9]
GENERAL TRANSIENTS	TURBINE TRIP	T_T	1.8	
	MSIV CLOSURE	T_M	0.72[1]	
	LOSS OF FEEDWATER	T_F	0.56	
	LOSS OF CONDENSER VACUUM	T_C	0.19	
	MANUAL SHUTDOWN	T_{MS}	3.3	
LOSS OF OFFSITE POWER	LOOP	T_E	7.9E-2	Plant Data, NUMARC-8700, NUREG-1032
LOCA's	SMALL LOCA	S_2	8.0E-3	EPRI NP-438
	MEDIUM LOCA	S_1	3.0E-3	EPRI NP-438
	LARGE LOCA	A	7.0E-4	EPRI NP-438
	LOCA OUTSIDE CONTAINMENT (INCLUDES INTERFACING SYSTEMS LOCA)	A_{OUT}	2.3E-7	WASH-1400 (PIPING FAILURE RATE)
SPECIAL TRANSIENTS	LOSS OF SERVICE WATER	T_{SW}	9.0E-3[2]	
	LOSS OF REACTOR BUILDING COOLING WATER	T_{RB}	[3]	
	LOSS OF 1 BUS 125VDC POWER	T_D	1.2E-3[4]	INDUSTRY EXP.
	REFERENCE LEG LEAK	RLL	4.0E-2	IPEM
	LOSS OF INSTRUMENT AIR	T_{IA}	6.0E-2	
	INADVERTENTLY OPENED SAFETY/RELIEF VALVE (IORV)	T_I	4.7E-3[5]	
	LOSS OF NITROOEN	T_{N2}	[3]	
	LOSS OF AC BUS	T_{AC}	[3]	
	LOSS OF DRYWELL COOLERS	T_{DW}	[3]	
	INTERNAL FLOOD	T_{FL}	[8]	WASH-1400 (PIPING FAILURE RATE)
FAILURE TO SCRAN (ATWS)	TURBINE TRIP	-	0.44[6][10]	
	LOSS OF FEEDWATER	-	0.56[10]	
	MSIV CLOSURE	-	1.17[6][10]	
	LOSS OF CONDENSER VACUUM	-	0.19[10]	
	LOSS OF OFFSITE POWER	-	7.9E-2[10]	
	IORV	-	4.7E-3[10]	
	Rx TRIP W/O TURBINE TRIP	-	0.9[7][10]	

Table 3.1-1 (continued)
Initiating Events Notes:

(1) The frequency for MSIV closure is obtained by adding the MSIV initiating event frequency to that fraction of IORV/SORV initiators for which relief valve reclosure is expected (see also Note 5).

(2) Derived from boolean expression for service water system.

(3) These initiators were not evaluated further because either the initiating event is a low frequency event or its impact is insignificant.

(4) No automatic trips are initiated as a result of loss of DC. This event requires a manual shutdown in 10 hours if DC is not restored. The initiator frequency includes the probability of failing either a bus or a battery with the additional failure to recover the DC bus within 10 hours.

(5) This value includes the probability of the valve reclosing after the primary system pressure drops: calculated by multiplying the IORV frequency by the probability of the valve not reclosing (0.031 X 0.15 = 4.7E-3).

(6) Transient initiator frequencies after redistribution of turbine trip events.

(7) Based on past experience, approximately 50% of reactor trip events did not result in turbine trip.

(8) Location dependent, see flooding analysis.

(9) Plant specific data unless otherwise noted.

(10) All frequencies are multiplied by the failure to scram per demand rate referenced in WASH 0460 (3E-5/d) to determine ATWS frequency.

TABLE 3.1-2
FRONTLINE SYSTEM SUCCESS CRITERIA

	SUCCESS CRITERIA	
Accident Initiator	Coolant Injection	Containment Heat Removal
Large LOCA	1 LPCI Pump OR 1 Core Spray (CS) Pump	1 RHR Pump or one 18" Vent Line
Medium LOCA	1 Feedwater Pump [2] OR HPCI [2] and 1 LPCI Pump OR HPCI and 1 Core Spray (CS) Pump OR HPCI and 1 Condensate Pump OR HPCI and 1 CRD Pump OR ADS [1] and 1 LPCI Pump OR ADS and 1 Core Spray (CS) Pump OR ADS and 1 Condensate Pump	1 RHR Pump or one 18" Vent Line
Small LOCA	HPCI OR 1 Feedwater Pump OR ADS and 1 Core Spray (CS) Pump OR ADS and 1 LPCI Pump OR ADS and 1 Condensate Pump	PCS or 1 RHR Pump or one 18" Vent Line
Transient	Same as Small LOCA OR RCIC	PCS or 1 RHR Pump or one 18" Vent Line
IORV/SORV	Same as Medium LOCA	PCS or 1 RHR Pump or one 18" Vent Line
Reclosed IORV/SORV	Same as Transient	PCS or 1 RHR Pump or one 18" Vent Line

Notes: [1] ADS requires operation of two of three SRVs for adequate depressurization.

[2] For a medium LOCA, the reactor will eventually depressurize due to the break. Therefore, successful operation of a low pressure pump will eventually be required.

Table 3.1-3

SUCCESS CRITERIA FOR TRANSIENT INITIATORS WITH FAILURE TO SCRAM
(86 GPM EQUIVALENT SLCS, MANUAL INITIATION)

	Failed Systems(s) or Functions(s)									
Transient	RPT 1 Pump	RPT 2 Pumps	RCIC	HPCI	SORV	1 R/V Not Open	1 RHR Train	2 RHR Trains	1 SLCS Pump	2 SLCS Pumps
Turbine Trip with bypass	A	N	A	A	A	A	A	N	A	A[3]
MSIV Closure [1]	A	N	A	A	A	A	A	N	A	N
Loss of Normal AC Power	[2]	[2]	A	A	A	A	A	N	A	N
Inadvertent Open Relief Valve (IORV)	A	N	A	A	-	A	A	N	A	N

A - Acceptable (i.e., Failure of given system/function will not result in core damage.)
N - Not Acceptable (i.e., Failure of given system/function will result in core damage.)

[1] Applicable for any isolation event.

[2] Pumps trip as a result of initiating event.

[3] Performing Level/Power control directing all steam to the main condenser allows for alternate boron injection.

Table 3.1-4
ACCIDENT SEQUENCE CLASSES

ACCIDENT CLASS DESIGNATOR	DESCRIPTION	PHYSICAL BASIS FOR CLASSIFICATION	REPRESENTATIVE ACCIDENT SEQUENCES
Class I	Transients Involving Loss of Coolant Makeup	Fuel will melt rapidly if cooling systems are not recovered; containment is intact initially at low pressure; at core melt and release pathway early in the event is from the vessel to the suppression pool through SRVs	Transients involving loss of high pressure inventory makeup and failure to depressurize RPV; transients involving loss of both high and low pressure injection.
Class II	Transients Involving Loss of Containment Heat Removal	Fuel will melt relatively slowly due to lower decay heat level if cooling systems are not recovered; containment is breached prior to core melt; release pathway is from the vessel to the suppression pool through SRVs during initial stages of core damage	Transients involving loss of containment heat removal; inadvertent SRV opening accidents with inadequate heat removal capability
Class III	LOCAs	Fuel will melt rapidly if cooling systems are not recovered; containment intact at core melt, but initially at high internal pressure; involves a release from the vessel to the drywell	Large and medium LOCAs with insufficient high or low pressure coolant makeup; small and medium LOCAs with failure of the SRVs to actuate and loss of high pressure inventory makeup; RPV failures with insufficient coolant makeup
Class IV	ATWS	Fuel will melt rapidly if cooling systems are not recovered; containment fails prior to core melt due to overpressure; initial release pathway is from the vessel to the suppression pool through SRVs	Transients involving loss of scram function and backup reactivity control

Table 3.1-4 (continued)
ACCIDENT SEQUENCE CLASSES

ACCIDENT CLASS DESIGNATOR	DESCRIPTION	PHYSICAL BASIS FOR CLASSIFICATION	REPRESENTATIVE ACCIDENT SEQUENCES
Class V	Unisolated LOCAs Outside Containment	Fuel will melt rapidly if cooling systems are not recovered; containment failed from initiation of accident due to containment bypass; involves a release pathway from the vessel which bypasses the containment	LOCAs outside containment with insufficient coolant makeup; interfacing system LOCAs with insufficient coolant makeup

Table 3.1-5
ACCIDENT SEQUENCE SUBCLASSES

ACCIDENT SEQUENCE CLASS	ACCIDENT SEQUENCE SUBCLASS	DEFINITION	EXAMPLE*
CLASS I	A	Accident Sequences Involving Loss of Inventory Makeup in which the Reactor Pressure Remains high	TQUX
	B	Accident Sequences Involving a Loss of AC Power and Loss of Coolant Inventory Makeup	T_EQUX
	C	Accident Sequences Involving a Failure to Scram (ATWS) with a Coincident Loss of All Inventory Makeup	TCQUV
	D	Accident Sequences Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been Successfully Reduced to Low pressure.	TQUV
CLASS II	-	Accident Sequences Involving a Loss of Containment Heat Removal	TW
CLASS III	A	Accident Sequences Initiated by Reactor Vessel Rupture where the Containment Integrity is not Breached in the Initial Time Phase of the Accident	RV'V"
	B	Accident Sequences Initiated or Resulting in Small or Medium LOCAs for Which the Reactor is not Depressurized	S_2QUX
	C	Accident Sequences Initiated or Resulting in Medium or Large LOCAs for which the Reactor is Depressurized and All Low Pressure Injection Fails	S_1V
	D	Accident Sequences which are Initiated by a LOCA or RPV Failure and for which the Vapor Suppression System has failed, Challenging the Containment Integrity	AD
CLASS IV	-	Accident Sequences Involving Failure to Scram and Failure to Inject Boron Leading to a High Pressure challenge to the containment resulting from Power Generation into the containment	TCC_2
CLASS V	-	Unisolated LOCA Outside Containment	$A_{OUT}V$

*Nomenclature refers to core damage sequence designations employed on event trees.

Figure 3.1-1
ANTICIPATED TRANSIENTS EVENT TREE

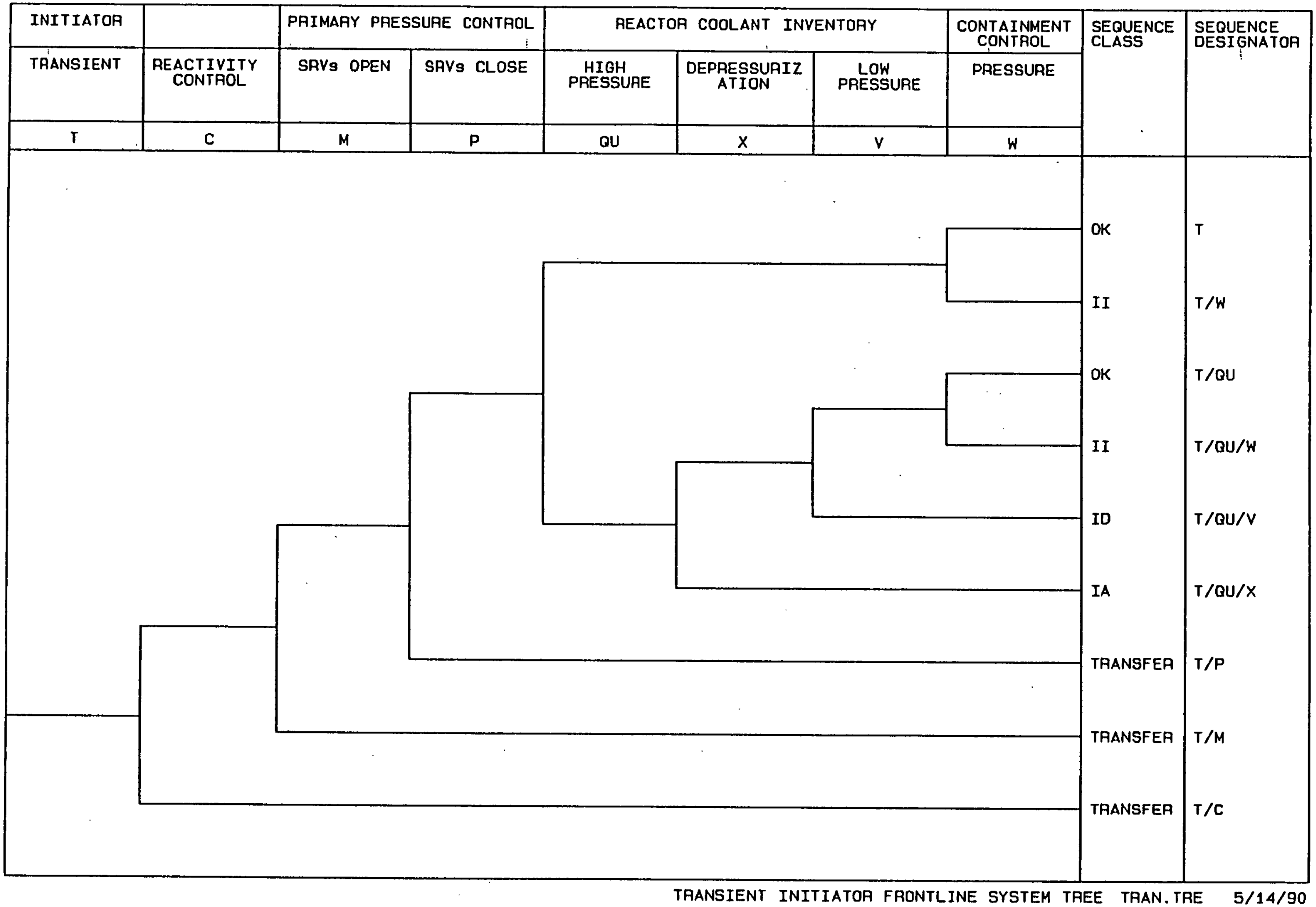
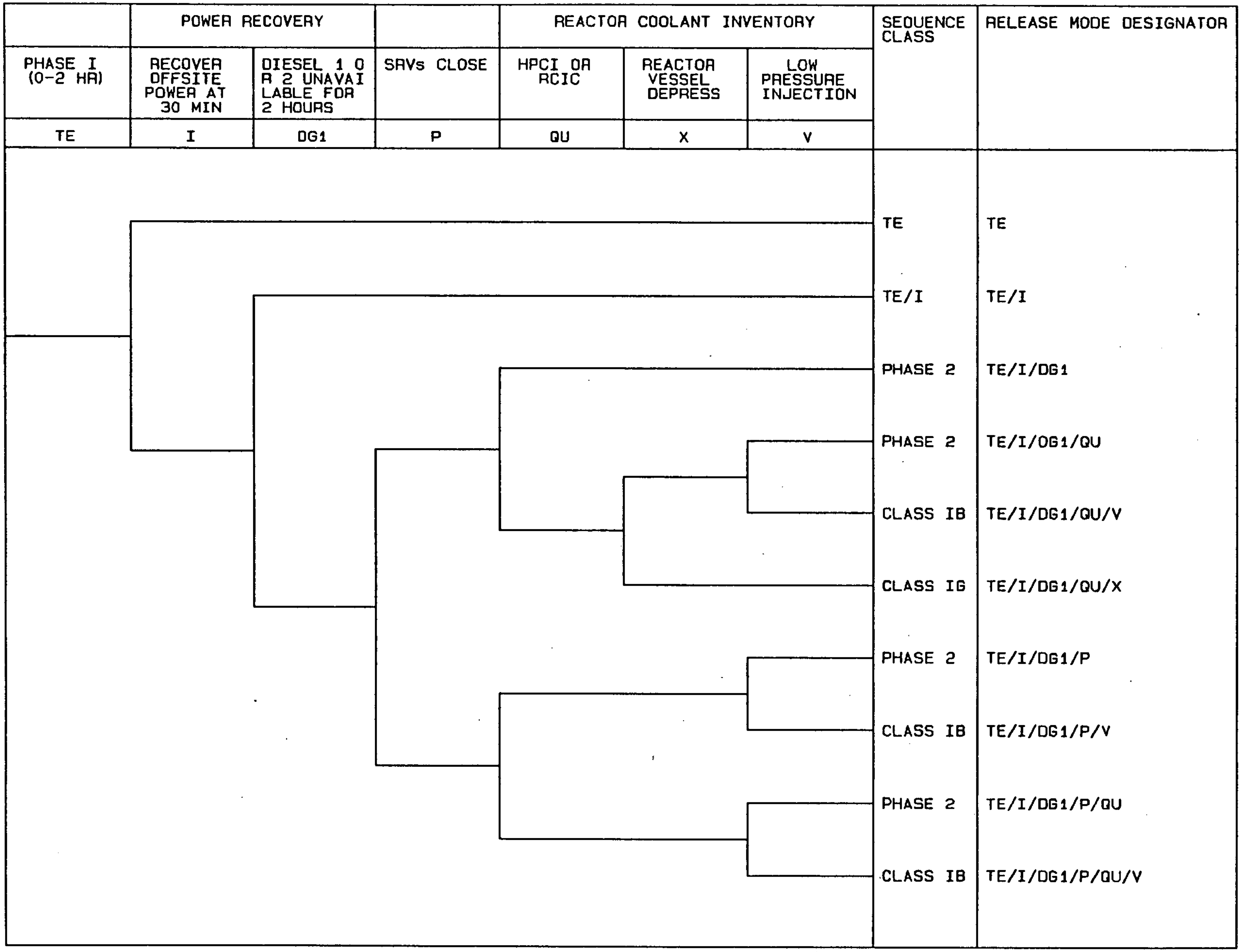
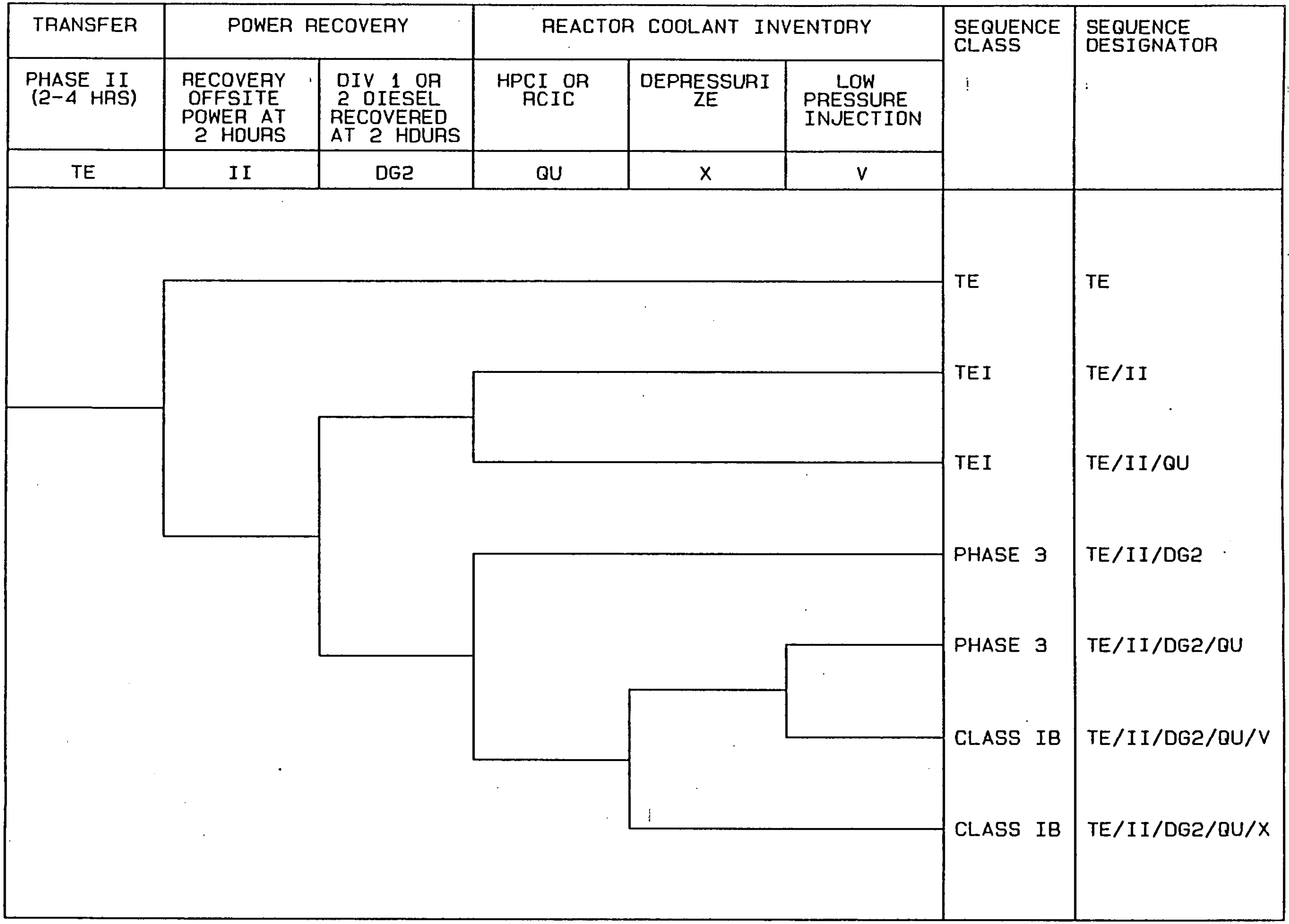


Figure 3.1-2
LOOP 0-2 HOURS EVENT TREE



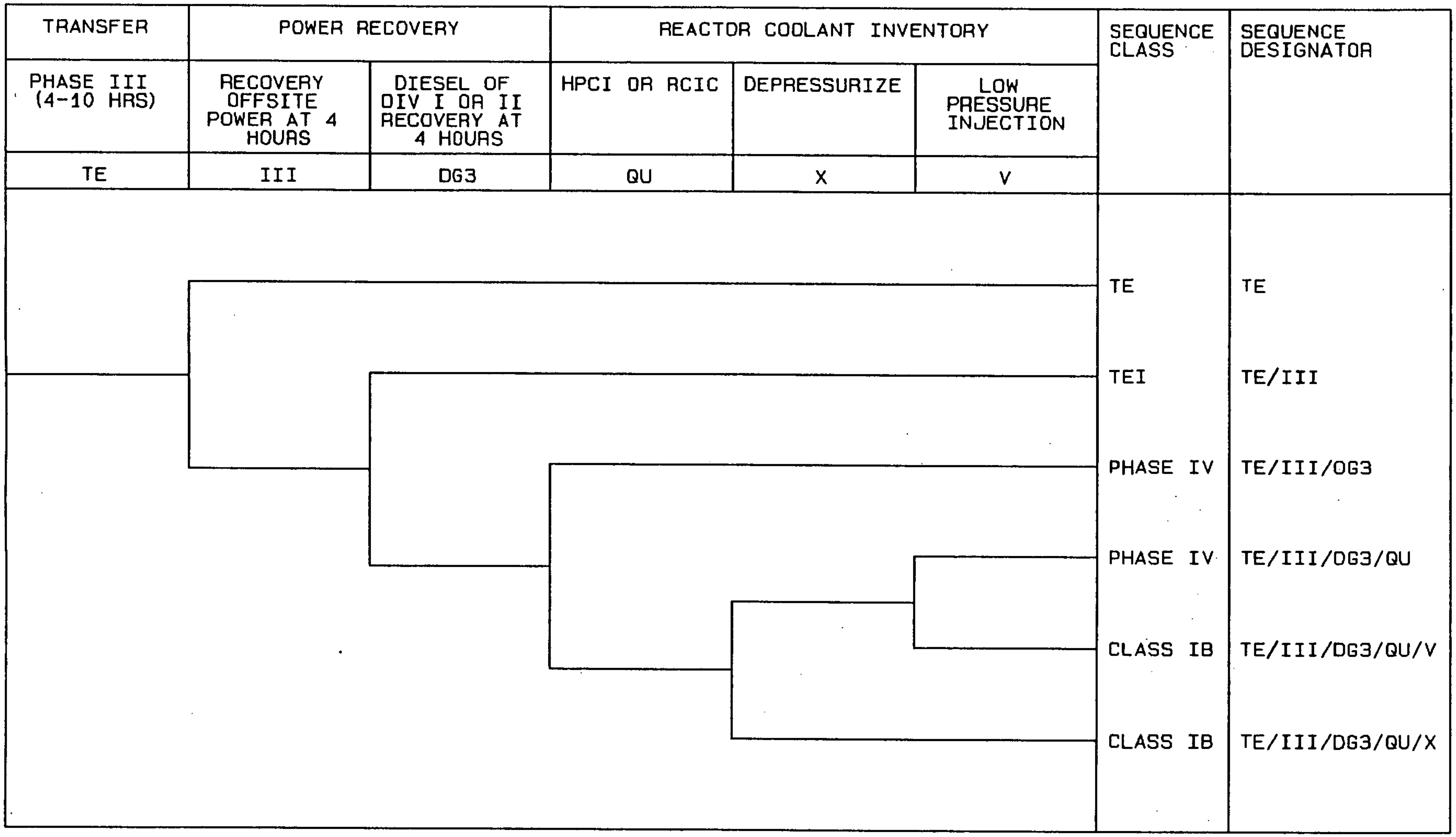
STATION BLACKOUT - PHASE I (0-2 HOURS) LOOPI.TRE 10/04/90

Figure 3.1-3
LOOP 2-4 HOURS EVENT TREE



STATION BLACKOUT - PHASE II (2-4 HOURS) LOOPII.TRE 5/18/90

Figure 3.1-4
LOOP 4-10 HOURS EVENT TREE



STATION BLACKOUT - PHASE III (4-10 HOURS) LOOPIII.TRE 5/14/90

Figure 3.1-5
LOOP WITHOUT OSP RECOVERY EVENT TREE

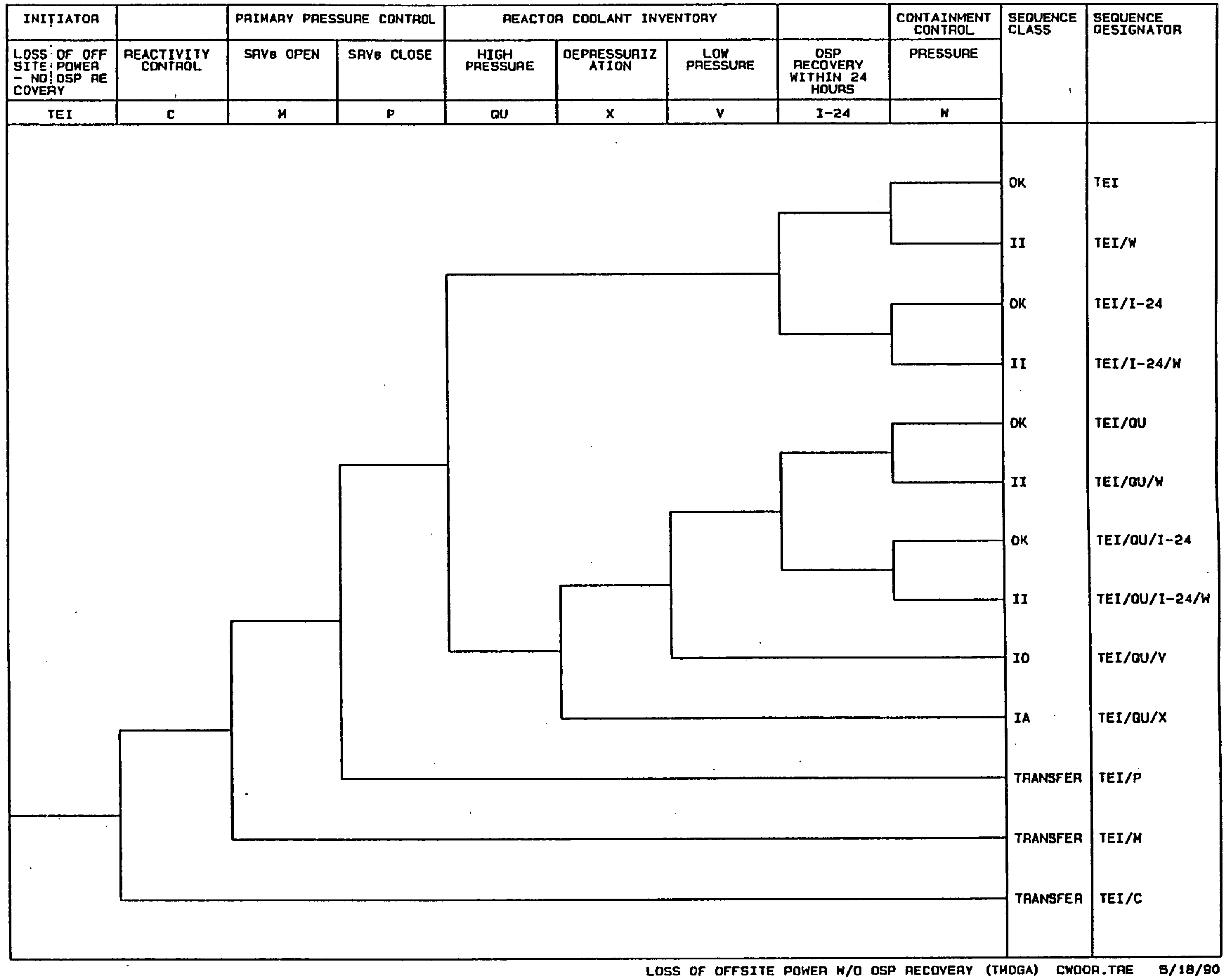
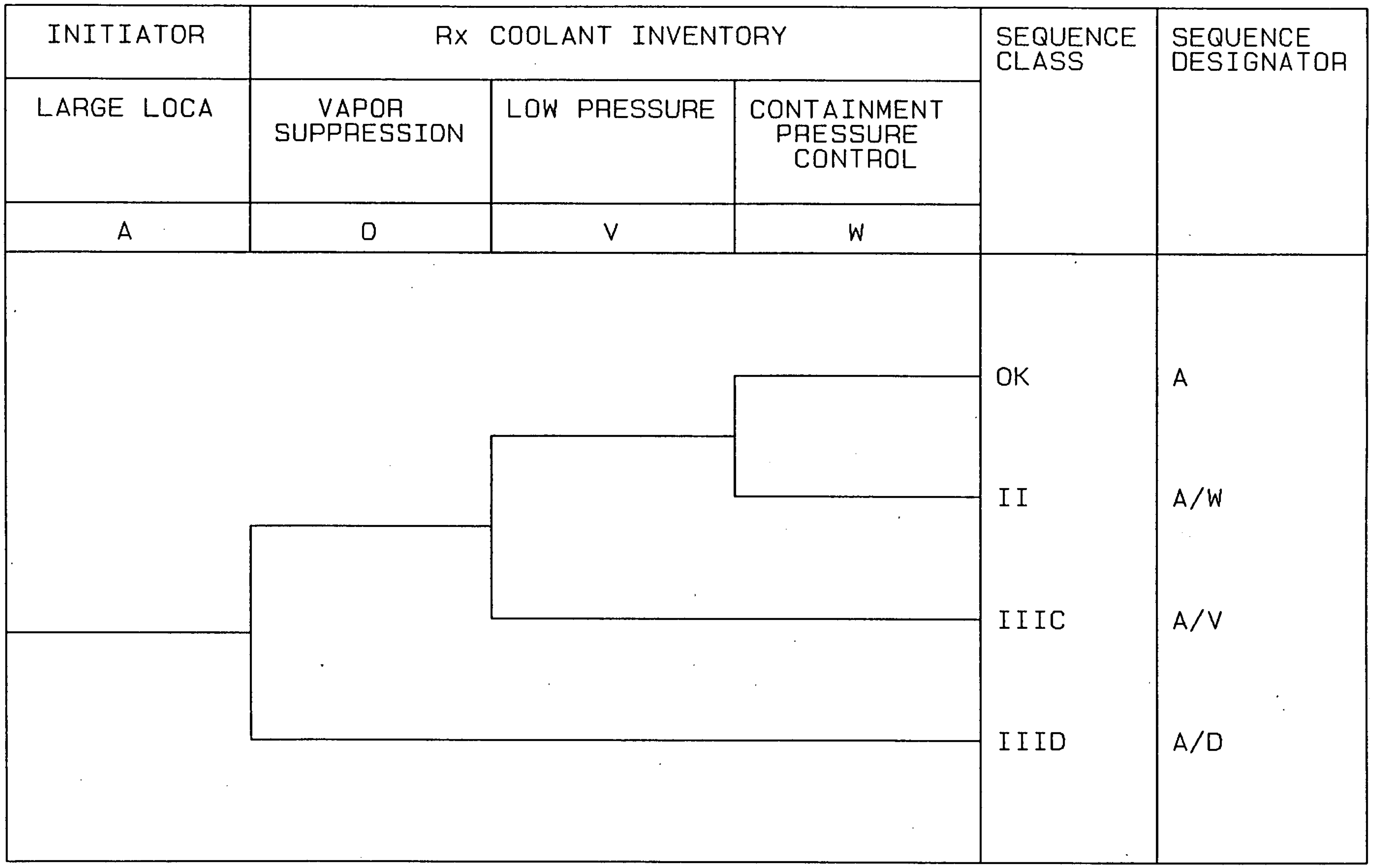


Figure 3.1-6
LARGE LOCA EVENT TREE



LARGE LOCA FRONTLINE TREE LLOCA.TRE 10/12/90

Figure 3.1-7
LARGE LOCA OUTSIDE CONTAINMENT EVENT TREE

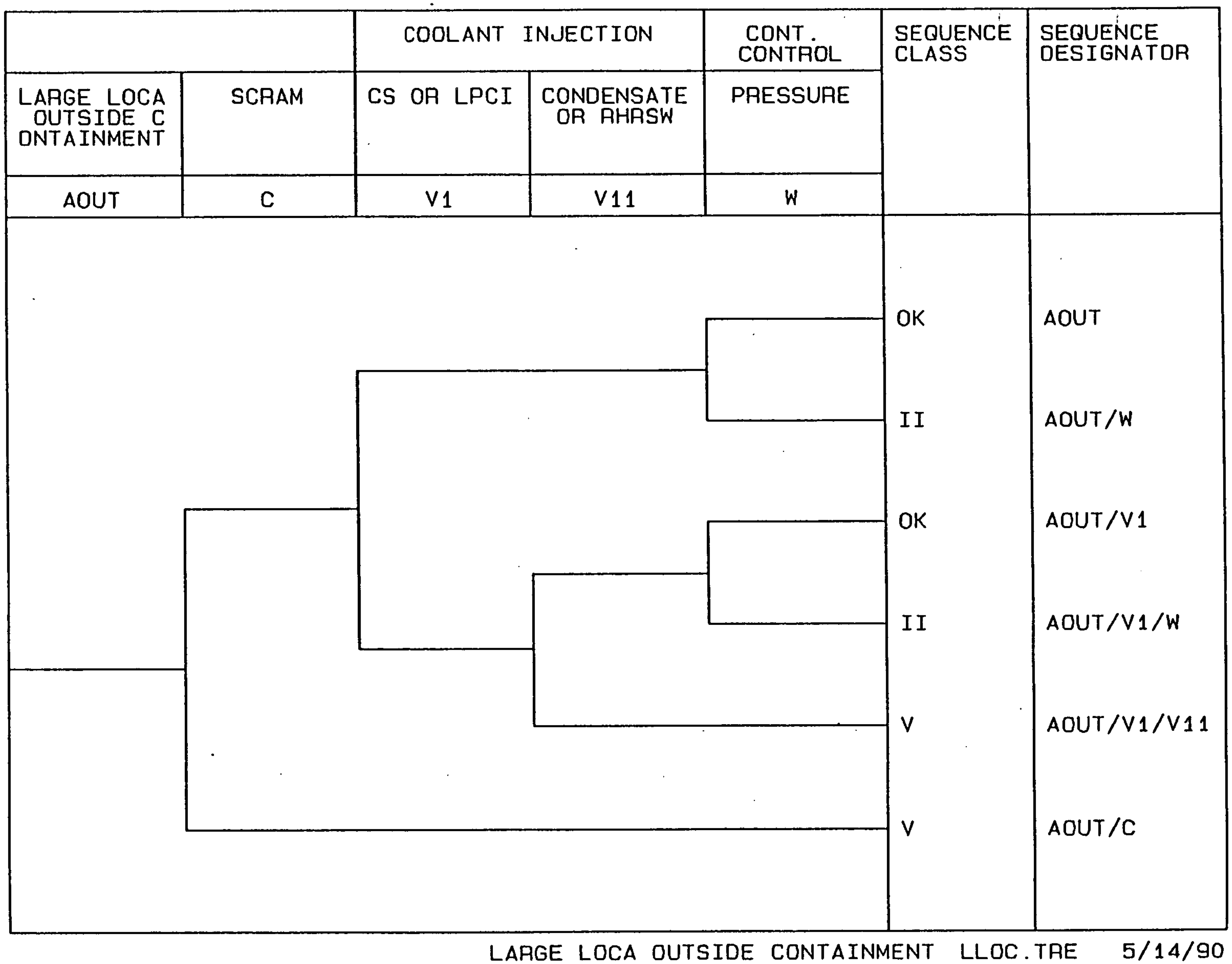
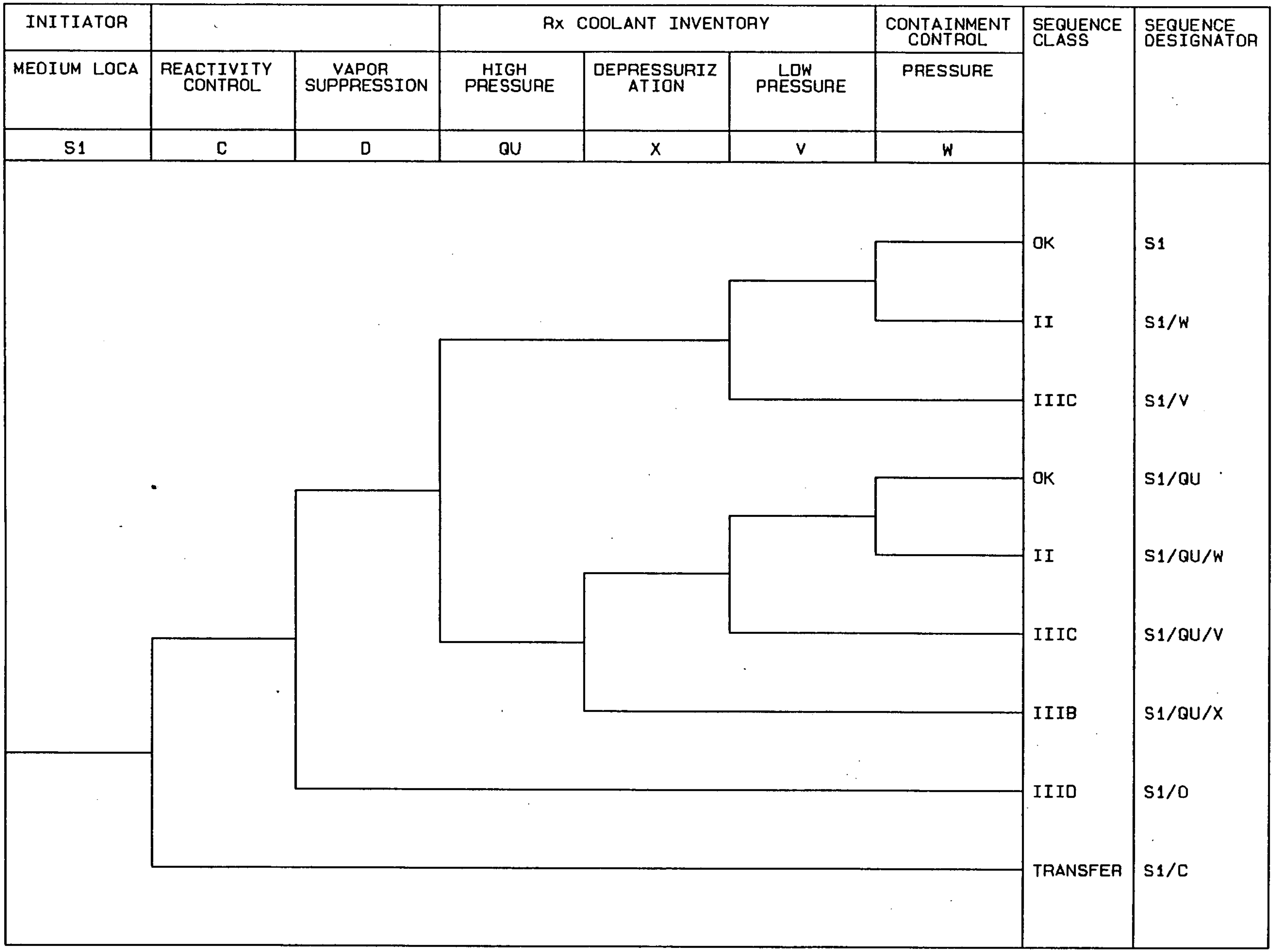
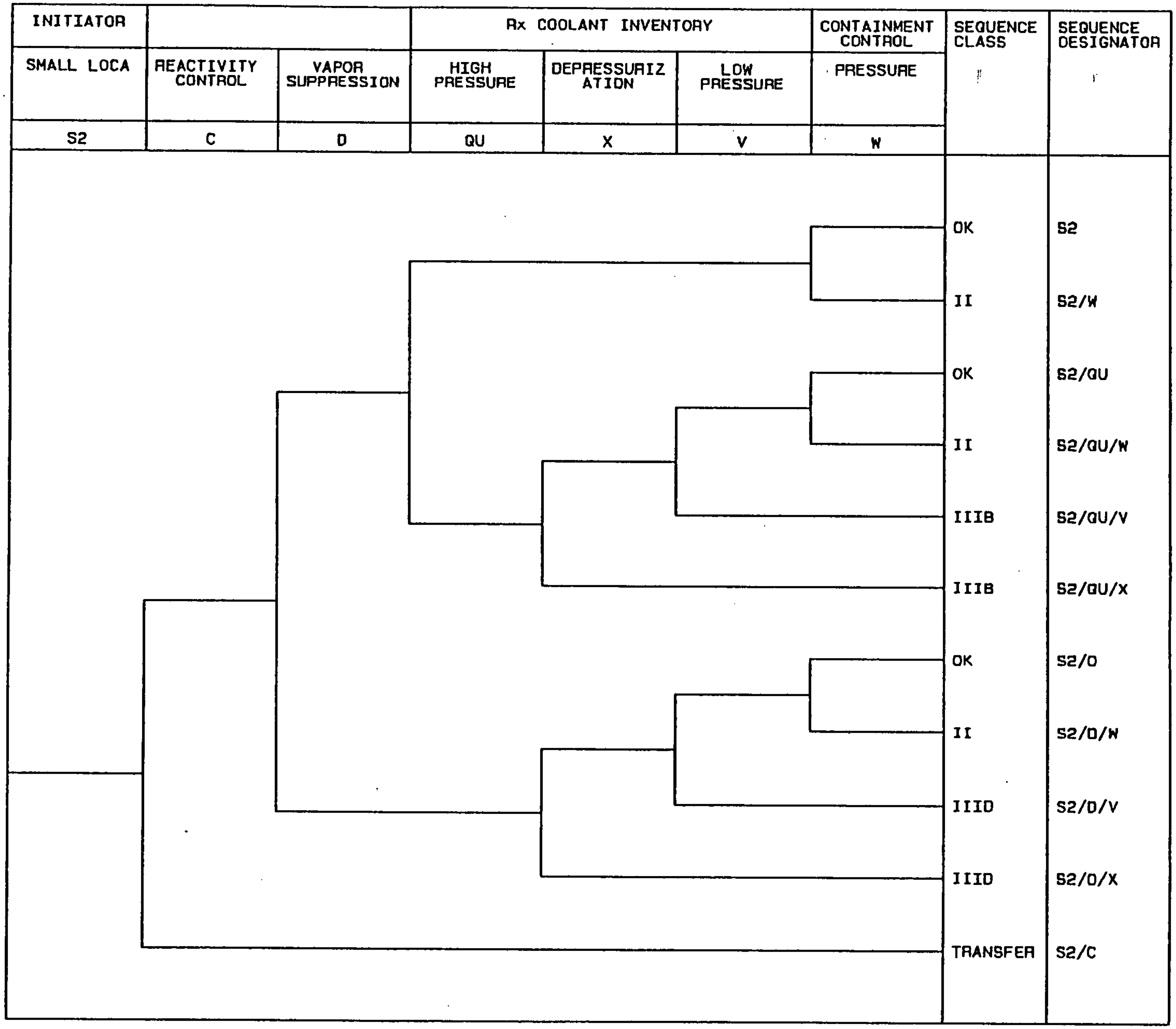


Figure 3.1-8
MEDIUM LOCA EVENT TREE



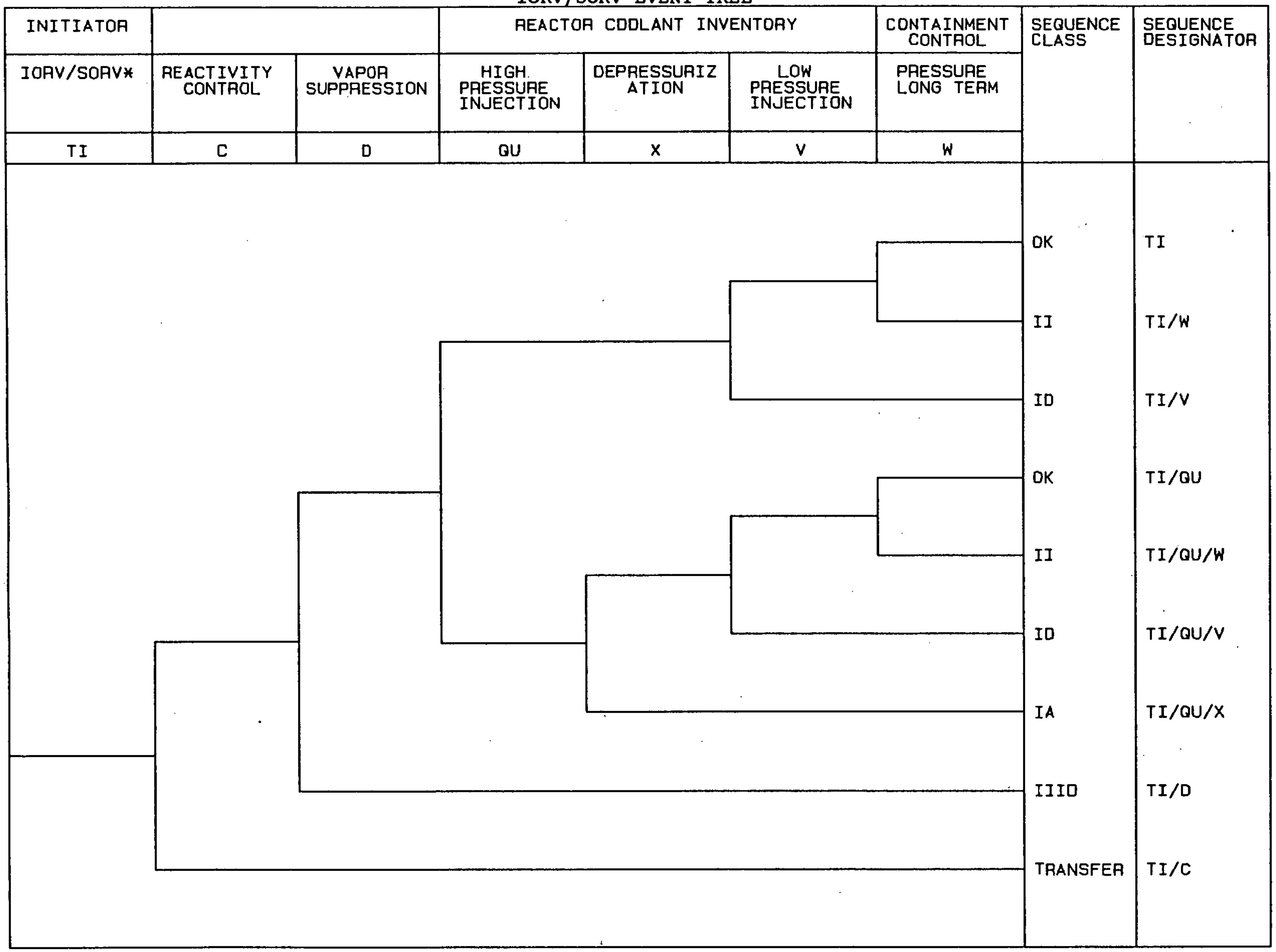
MEDIUM LOCA - FRONTLINE SYSTEM TREE MLOCA.TRE 5/14/90

Figure 3.1-9
SMALL LOCA EVENT TREE



SMALL LOCA - FRONTLINE SYSTEM TREE SLOCA.TRE 10/10/90

Figure 3.1-10
IORV/SORV EVENT TREE



IORV/SORV EVENT TREE FOR SUPPORT STATE A STA.TRE 5/18/90

* INCLUDES RECLOSURE PROBABILITY

Figure 3.1-11
ATWS WITHOUT TURBINE TRIP EVENT TREE

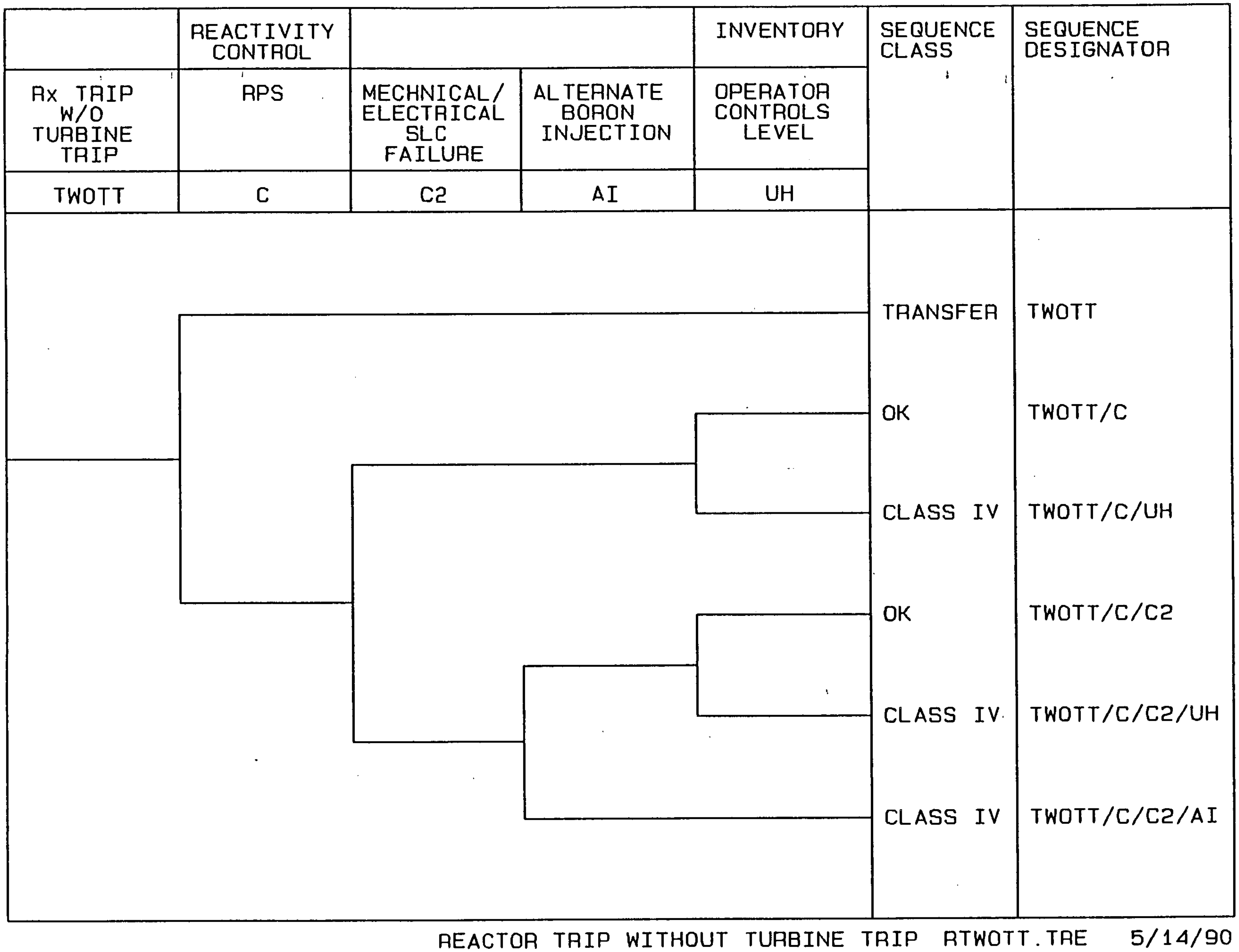


Figure 3.1-12
ATWS WITH TURBINE TRIP EVENT TREE

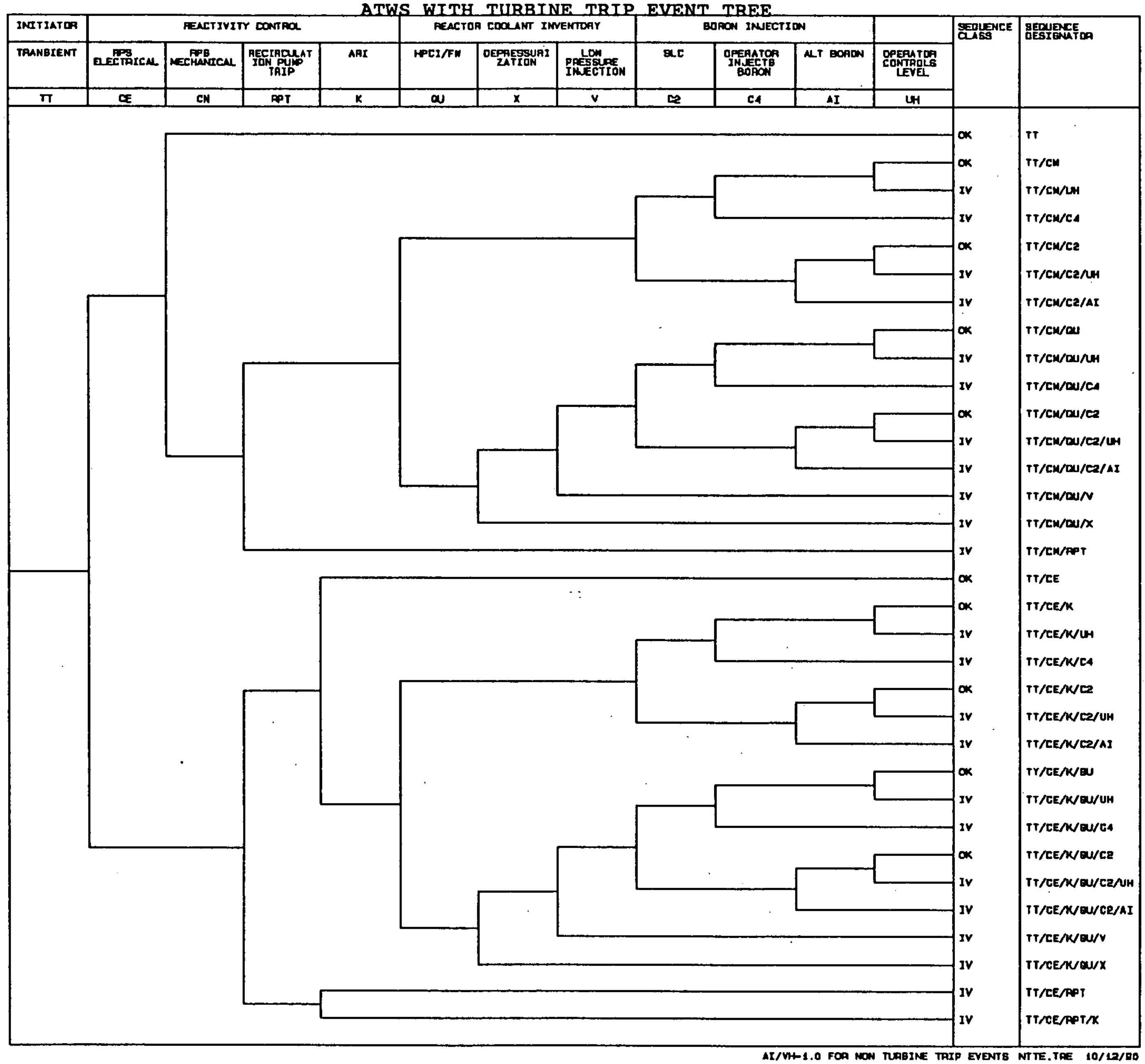


Table 3.2-5 (continued)
SUPPORT SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(1) - SYSTEM IS ACTUALLY ONE TRAIN WITH TWO PUMPS IN PARALLEL (A,B).

(2) - THE DEPENDENCIES SHOWN IN THIS TABLE FOR THE TORUS COOLING MODE OF RHR SYSTEM OPERATION ARE ALSO APPLICABLE TO THE OTHER MODES OF THE RHR SYSTEM NOT LISTED HERE: SHUTDOWN COOLING, ALTERNATE SHUTDOWN COOLING, AND TORUS LEVEL CONTROL. THE SHUTDOWN COOLING AND TORUS LEVEL CONTROL MODES ALSO REQUIRE 250 VOLT DC POWER FOR VALVE OPERATION.

(3) - LOSS OF MCC 33A RESULTS IN LOSS OF HPCI ROOM COOLER V-AC-8A. THE OTHER HPCI ROOM COOLER (V-AC-8B) IS POWERED FROM MCC 43A AND THEREFORE WOULD STILL BE AVAILABLE. THE HPCI ROOM COOLERS ARE ASSUMED TO BE UNNEEDED TO CONTROL THE HPCI ROOM TEMPERATURE FOR ALL SCENARIOS OF CONCERN IN THIS STUDY.

(4) - LOSS OF MCC 43A RESULTS IN LOSS OF HPCI ROOM COOLER V-AC-8B. THE OTHER HPCI ROOM COOLER (V-AC-8A) IS POWERED FROM MCC 43A AND THEREFORE WOULD STILL BE AVAILABLE. THE HPCI ROOM COOLERS ARE ASSUMED TO BE UNNEEDED TO CONTROL THE HPCI ROOM TEMPERATURE FOR ALL SCENARIOS OF CONCERN IN THIS STUDY.

(5) - Y70 PROVIDES POWER TO RCIC FLOW CONTROL INSTRUMENTS AND CONTROL ROOM PRESSURE INDICATION. Y80 PROVIDES POWER TO HPCI FLOW CONTROL INSTRUMENTS AND CONTROL ROOM PRESSURE INDICATION. LOSS OF EITHER PANEL RENDERS THE AFFECTED FLOW CONTROL SYSTEM INOPERABLE, AND THE AFFECTED SYSTEM INOPERABLE.

(6) - A SUMMARY OF THE ESSENTIAL LOAD SUPPLIED BY THE DC PANELS IS PROVIDED BELOW:

D31 (FROM 250V BATTERY A) -	SUPPLIES 250 VOLT POWER TO MCC 0311 WHICH POWERS ALL OF THE RCIC MOTOR OPERATED VALVES AND AUXILIARY PUMPS AND PROVIDES 125V DC POWER TO PANEL D33.
D100 (FROM 250V BATTERY B) -	SUPPLIES 250 VOLT POWER TO MCC 0312 WHICH POWERS ALL OF THE HPCI MOTOR OPERATED VALVES AND AUXILIARY PUMPS, AND PROVIDES 125VDC CONTROL POWER TO PANEL D-312 WHICH SUPPLIES 125V DC CONTROL POWER TO MCC D312.
D11 (FROM 125V BATTERY A) -	PROVIDES 125 VOLT DC POWER TO THE RCIC RELAY PANEL. LOSS OF THIS PANEL RESULTS IN LOSS OF POWER TO THE RCIC INITIATION AND TRIP LOGIC. THIS PANEL ALSO SUPPLIES POWER TO 1/2 OF THE HPCI STEAMLINE ISOLATION LOGIC.
D111 (FROM 125V BATTERY A) -	SUPPLIED DIRECTLY FROM PANEL D11.
D21 (FROM 125V BATTERY B) -	PROVIDES 125 VOLT DC POWER TO THE HPCI RELAY PANEL. LOSS OF THIS PANEL RESULTS IN LOSS OF THE HPCI INITIATION AND TRIP LOGIC. THIS PANEL ALSO SUPPLIES POWER TO 1/2 OF THE RCIC STEAMLINE TEMPERATURE SWITCHES AND ALL OF THE RCIC HIGH STEAM FLOW DIFFERENTIAL PRESSURE SWITCHES.
D211 (FROM 125V BATTERY B) -	SUPPLIED DIRECTLY FROM PANEL D21.
D33 (FROM 250V BATTERY A) -	PROVIDES 125 VOLT CONTROL POWER TO MCC D311 WHICH SUPPLIES POWER TO ALL OF THE RCIC MOTOR OPERATED VALVES AND AUXILIARY PUMPS.

(7) - SERVICE WATER IS THE PRIMARY COOLING MEDIUM FOR THE HPCI ROOM COOLERS, AND THE ONLY COOLING MEDIUM FOR THE RCIC AND CRD ROOM COOLERS.
ESW TRAIN A IS A BACK-UP COOLING MEDIUM FOR HPCI ROOM COOLER V-AC-8A.
ESW TRAIN B IS A BACK-UP COOLING MEDIUM FOR HPCI ROOM COOLER V-AC-8B.
HPCI, RCIC, AND CRD HAVE BEEN DEMONSTRATED TO BE UNAFFECTED BY LOSS OF ROOM COOLING DURING THE 24 HOUR MISSION TIME.

(8) - V-AC-8A IS THE PRIMARY HPCI ROOM COOLER AND WILL START FIRST ON HIGH TEMPERATURE IN THE HPCI ROOM. V-AC-8B IS THE BACK-UP UNIT AND WILL START AT A HIGHER HPCI ROOM TEMPERATURE. EITHER ROOM COOLER IS CAPABLE OF CONTROLLING THE ROOM TEMPERATURE. LOSS OF BOTH ROOM COOLERS HAS NO SIGNIFICANT IMPACT ON HPCI OPERATION.

(9) - ALL EIGHT SAFETY/RELIEF VALVES (SRVs) MAY BE USED TO PROVIDE REACTOR VESSEL OVERPRESSURE PROTECTION, OR TO DEPRESSURIZE THE REACTOR VESSEL. THE OVERPRESSURE PROTECTION FUNCTION OF THE SRVs IS INDEPENDENT OF ALL OTHER PLANT SYSTEMS. THREE OF THE SRVs (A,C,D) ARE AUTOMATICALLY CONTROLLED BY THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS). ALL EIGHT OF THE SRVs MAY BE MANUALLY OPERATED FROM THE CONTROL ROOM TO DEPRESSURIZE THE REACTOR VESSEL. FOUR OF THE SRVs (E-H) MAY BE MANUALLY CONTROLLED FROM THE ALTERNATE SHUTDOWN PANEL LOCATED IN THE EFT BUILDING. ANY TWO OF THE SRVs MAY BE USED TO SUCCESSFULLY DEPRESSURIZE THE REACTOR VESSEL.

(10) - THE PRIMARY SOURCE OF DC POWER FOR SRV A,B,C,D SOLENOIDS IS D11. THE PRIMARY SOURCE OF DC POWER FOR SRV E,F,G,H SOLENOIDS IS D33. THE SOLENOID VALVES FOR SRV A,C,D ARE BACKED UP BY D21. SRVs E,F,G,H CAN BE OPERATED FROM ASDS POWERED FROM D100.

Table 3.2-5 (continued)
SUPPORT SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(11) - THE PRIMARY SOURCE OF PNEUMATIC SUPPLY FOR THE SRVs IS INSTRUMENT NITROGEN. IF THE NITROGEN HEADER PRESSURE IS TOO HIGH OR LOW, INSTRUMENT AIR IS AUTOMATICALLY ALIGNED TO PROVIDE CONTROL OF THE SRVs. EACH SRV ALSO HAS A PNEUMATIC ACCUMULATOR THAT IS RATED FOR APPROXIMATELY 30 MINUTES OF VALVE OPERATION. IN ADDITION, SRVs B,F HAVE BACK-UP NITROGEN BOTTLES.

(12) - THE RHR/CS ROOM COOLERS AND PUMP BEARING OIL IS COOLED PRIMARILY BY THE SERVICE WATER SYSTEM. THE ESW SYSTEM IS A BACK-UP COOLING SOURCE IF SERVICE WATER IS UNAVAILABLE. LOSS OF RHR/CS ROOM COOLING HAS LITTLE IMPACT ON THE OPERATION OF THE RHR/CS PUMPS FOR 24 HOURS.

(13) - THE RHRSW PUMP BEARING OIL MAY BE COOLED BY EITHER OF THE FOLLOWING SOURCES: (1) SERVICE WATER, OR (2) WATER TAPPED FROM THE OUTLET OF EACH PUMP BACK THROUGH ITS COOLER.

(14) - THE NORMAL AIR SUPPLY TO THE RHRSW FLOW CONTROL VALVES IS INSTRUMENT AIR. AIR COMPRESSOR K-10A IS A BACK-UP SUPPLY VALVE CV-1728; AIR COMPRESSOR K-10B IS A BACK-UP SUPPLY FOR CV-1729.

(15) - CLOSURE OF THE INBOARD MSIVs WILL RESULT IF PANELS Y70 AND D11 ARE DEENERGIZED. CLOSURE OF THE OUTBOARD MSIVs WILL RESULT IF PANELS Y80 AND D21 ARE DEENERGIZED.

(16) - LOSS OF INSTRUMENT AIR WILL RESULT IN CLOSURE OF THE OUTBOARD MSIVs WHEN THE MSIV ACCUMULATORS BLEED DOWN (ESTIMATED TO TAKE APPROXIMATELY 30 MINUTES). LOSS OF INSTRUMENT NITROGEN AND INSTRUMENT AIR WILL RESULT IN CLOSURE OF THE INBOARD MSIVs WHEN THE MSIV ACCUMULATORS BLEED DOWN.

(17) - 125 VOLT DC POWER IS REQUIRED FOR RHR, CORE SPRAY, AND RHRSW PUMP BREAKER CLOSURE. DC PANELS D111 AND D211 MAY BE MANUALLY CROSSTIED TO PROVIDE BREAKER CONTROL POWER IF LOSS OF EITHER DC BUS OCCURS. THE CIRCUIT BREAKERS FOR THESE PUMPS MAY BE MANUALLY OPERATED AT THE BREAKER CABINETS IN THE SWITCHGEAR ROOMS.

(18) - SERVICE WATER PROVIDES COOLING FOR THE INSTRUMENT AIR COMPRESSORS AND AFTERCOOLERS. LOSS OF SERVICE WATER WILL CAUSE LOSS OF AIR.

(19) - THE CONTAINMENT VENTING SYSTEM CANNOT BE OPERATED UNLESS INSTRUMENT AIR IS AVAILABLE.

(20) - LOSS OF HEATING FOR THE BORON TANK AND SLC PIPING.

(21) - THE BACK-UP SCRAM VALVES REQUIRE POWER FROM PANEL D11 OR D21 FOR OPERATION.

(22) - LOSS OF CRD ROOM COOLING WILL NOT IMPACT CRD PUMP OPERATION.

(23) - CONDENSATE PUMPS A & B HAVE DEDICATED COOLERS THAT ARE COOLED BY SERVICE WATER. THE IMPACT OF LOSS OF THESE COOLERS ON PUMP OPERATION IS SMALL.

(24) - ON LOSS OF AIR, THE CONDENSATE DEMINERALIZER VALVES FAIL CLOSED AND THE DEMINERALIZER BYPASS VALVE IS DESIGNED TO OPEN BUT HAS FAILED TO DO SO ON SOME OCCASIONS. THE FEEDWATER FLOW REGULATING VALVES FAIL AS IS. THIS WILL RESULT IN FEEDWATER PUMP TRIP DUE TO HIGH REACTOR LEVEL BECAUSE OF THE HIGH FW FLOW RATE FOLLOWING SCRAM AND THE FLOW REGULATING VALVES CANNOT BE CLOSED TO THROTTLE DOWN THE FEEDWATER FLOW.

(25) - THE CRD SYSTEM FLOW CONTROL VALVES FAIL CLOSED ON LOSS OF AIR. THE TEST BYPASS VALVE MUST BE MANUALLY OPENED.

(26) - RBCCW IS REQUIRED FOR CRD PUMP COOLING. SERVICE WATER PROVIDES RBCCW HEAT EXCHANGER COOLING. THE CRD PUMPS ARE ASSUMED TO FAIL IF COOLING IS LOST.

(27) - SERVICE WATER PROVIDES FEEDWATER AND CONDENSATE PUMP BEARING OIL COOLING. IT IS ASSUMED THAT THE FEEDWATER PUMPS WILL NOT OPERATE WITHOUT COOLING, BUT THE CONDENSATE PUMPS WILL WORK FINE.

(28) - PANEL L-35 PROVIDES POWER TO THE CONDENSATE PUMP THRUST BEARING OIL PUMP. IT IS ASSUMED THAT THE CONDENSATE PUMPS WILL NOT OPERATE WITHOUT THIS PUMP.

(29) - PANEL Y30 PROVIDES CONTROL POWER FOR THE FEEDWATER FLOW REGULATING VALVES. LOSS OF POWER TO THIS PANEL IS ASSUMED TO CAUSE THE REG. VALVES TO FAIL AS IS. THIS IS ASSUMED TO RESULT IN FEEDWATER PUMP TRIP FOR THE REASON DISCUSSED IN NOTE 24.

Table 3.2-5 (continued)
SUPPORT SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(30) - LOSS OF CONTROL POWER FOR PUMP BREAKER OPERATION. LOSS OF THESE PANELS WILL RESULT IN FAILURE TO AUTOMATICALLY TRIP A RUNNING PUMP, AND FAILURE TO START A PUMP THAT HAS TRIPPED. THE CIRCUIT BREAKERS FOR THESE PUMPS MAY BE MANUALLY OPERATED AT THE BREAKER CABINETS IN THE SWITCHGEAR ROOMS.

(31) - ONE CRD PUMP IS NORMALLY RUNNING AND ONE IS IN STANDBY. LOSS OF DC POWER CAUSES LOSS OF BREAKER CONTROL FOR THE STANDBY PUMP, AND THE RUNNING PUMP IF IT HAS BEEN TRIPPED. THE CIRCUIT BREAKERS FOR THESE PUMPS MAY BE MANUALLY OPERATED AT THE BREAKER CABINETS IH THE SWITCHGEAR ROOMS.

(32) - RBCCW COOLS THE RHR AND CORE SPRAY PUMP SHAFT SEALS. LOSS OF SHAFT SEAL COOLING FOR THESE PUMPS IS ASSUMED TO HAVE NO IMPACT ON PUMP OPERATION DURING THE 24 HOUR MISSION TIME.

(33) - LOSS OF INSTRUMENT AIR WILL CAUSE THE INTAKE BASIN LEVEL SENSOR TO FAIL LOW WHICH WILL RESULT IN CIRCULATING WATER PUMP TRIP.

(34) - RBCCW IS THE HEAT REMOVAL MEDIUM FOR THE DRYWELL COOLERS. SERVICE WATER PROVIDES HEAT REMOVAL FROM THE RBCCW HEAT EXCHANGERS.

(35) - LOSS OF THE RHRSW SYSTEM MAY HAVE A LONG TERM IMPACT ON LPCI AND CORE SPRAY OPERATION DUE TO LOSS OF SUPPRESSION POOL NPSH FOR THE LPCI AND CS PUMPS.

(36) - RHRSW TRAIN A COOLS RHR HEAT EXCHANGER A; RHRSW TRAIN B COOLS RHR HEAT EXCHANGER B.

(37) - LOSS OF THE CONDENSATE SERVICE WATER ("KEEP-FILL") SYSTEM IS ASSUMED TO HAVE NO ADVERSE IMPACTS ON THE OPERATION OF LPCI, CORE SPRAY, OR ANY OF THE RHR HEAT REMOVAL MODES.

(38) - THE RHR, HPCI, AND RCIC MINIMUM FLOW VALVES FAIL OPEN ON AIR LOSS. THIS WILL RESULT IN A 10% FLOW DIVERSION TO THE SUPPRESSION POOL. A 10% FLOW DIVERSION IS ASSUMED TO HAVE NO ADVERSE IMPACTS ON THESE SYSTEMS.

(39) - MCC 43A PROVIDES THE MOTIVE POWER FOR RCIC ROOM COOLER V-AC-6. LOSS OF V-AC-6 RESULTS IN LOSS OF RCIC ROOM COOLING. HOWEVER, RCIC CAN OPERATE FOR AT LEAST 24 HOURS WITHOUT ROOM COOLING.

(40) - BUSES 13 AND 14 PROVIDE MOTIVE POWER FOR THE CIRCULATING WATER PUMPS. BOTH MUST FAIL TO DISABLE MAIN CONDENSER.

(41) - POWER REQUIRED FOR PUMP MOTIVE POWER.

(42) - DC POWER IS REQUIRED TO OPERATE THE ARI SOLENOIDS AND TRIP THE RECIRC PUMP BREAKERS.

(43) - POWER REQUIRED FOR VALVE OPERATION.

(44) - POWER REQUIRED FOR FAN OPERATION.

(45) - TORUS OR DRYWELL VENTING REQUIRES 120 VOLT AC POWER. BOTH Y70 AND Y80 MUST BE AVAILABLE IN ORDER FOR EITHER VENT TO WORK.

(46) - NO POWER TO OPEN AND CONTROL THE RHRSW FLOW CONTROL VALVES (CV-1728, CV-1729).

(47) - LOSS OF MOTIVE AND CONTROL POWER TO THE DAMPERS. HOWEVER, THEY ARE ALL NORMALLY OPEN EXCEPT FOR THE STANDBY FAN.

(48) - LOSS OF RHR/CS ROOM COOLING HAS BEEN DEMONSTRATED TO HAVE LITTLE IMPACT ON RHR/CS OPERATION.

(49) - THE RCIC SYSTEM WILL OPERATE AT LEAST 24 HOURS WITHOUT ROOM COOLING.

(50) - STEAM JET AIR EJECTORS ARE THE PRIMARY MEANS OF REMOVING NON-CONDENSIBLES FROM THE CONDENSER. THE VACUUM PUMPS ARE A BACKUP IF POWER IS LESS THAN 5%, OFFSITE POWER IS AVAILABLE, AND THE MSIVs AND TBVs ARE OPEN.

(51) - 480 VOLT AC POWER REQUIRED FOR CIRCULATING WATER PUMP EXCITATION AND MVP.

Table 3.2-5 (continued)
SUPPORT SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(52) - TEMPERATURE SENSORS ARE LOCATED IN THE STEAM CHASE THAT WILL CLOSE THE HPCI AND RCIC STEAM SUPPLY VALVES AND THE MSIVs, ON HIGH TEMPERATURE.

(53) - LOSS OF POWER TO Y10 AND Y20 WILL CAUSE THE FEEDWATER FLOW REGULATING VALVES TO FAIL AS IS.

(54) - LOSS OF POWER TO Y20 WILL CLOSE THE NITROGEN SUPPLY VALVES TO B,F SRVs, PREVENTING DEPRESSURIZATION OF THE REACTOR VESSEL DURING STATION BLACKOUT. (Y20 IS NOT BACKED UP BY DC POWER). LOSS OF Y20 ALSO ISOLATES NORMAL PNEUMATIC SUPPLY TO SRVs.

(55) - LOSS OF MCC-41 (SUPPLIES L-41) RESULTS IH LOSS OF LIQUID N2 SUPPLY TO <u>ALL</u> SRVs.

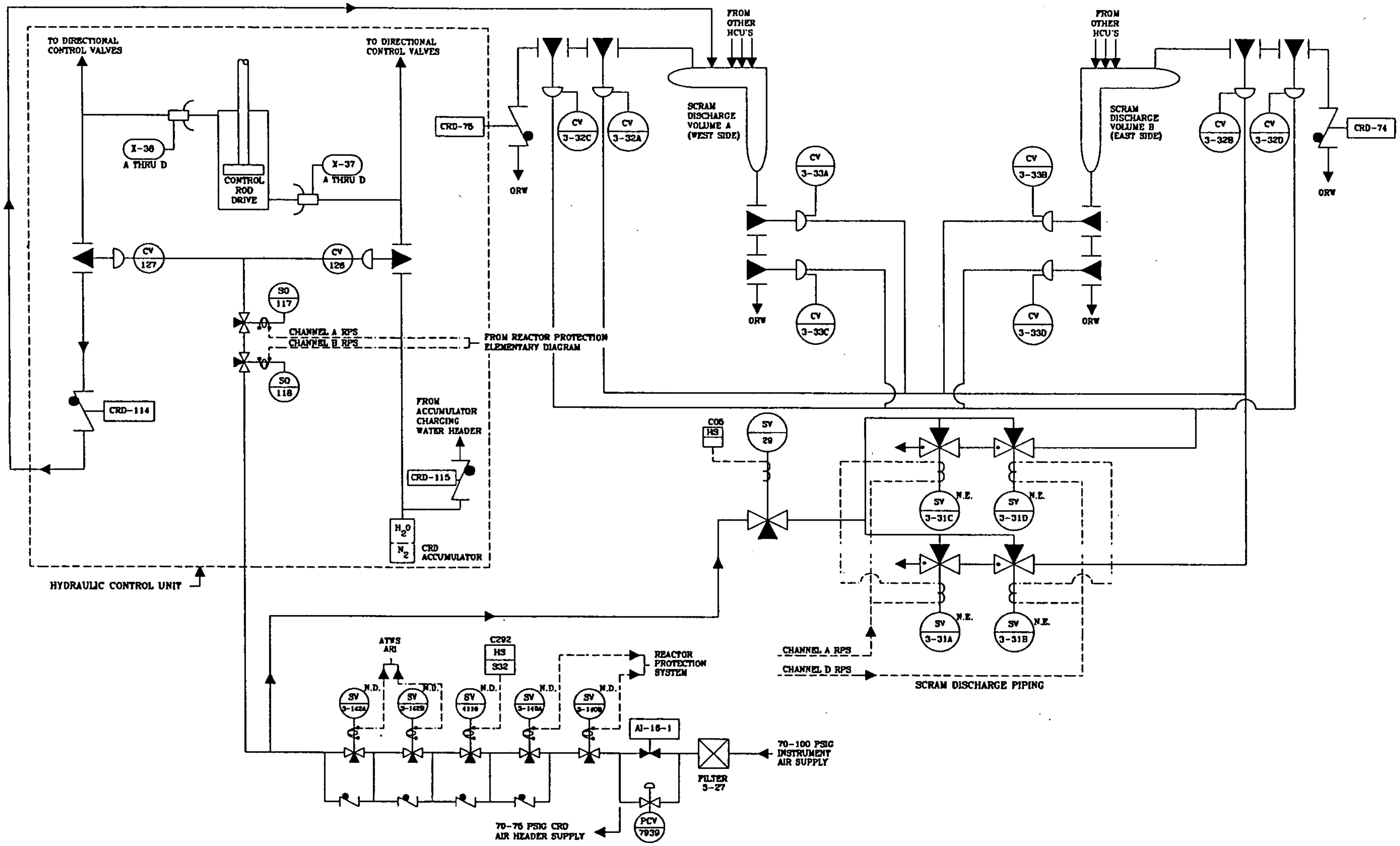


Figure No. 3.2-1
Reactor Protection System

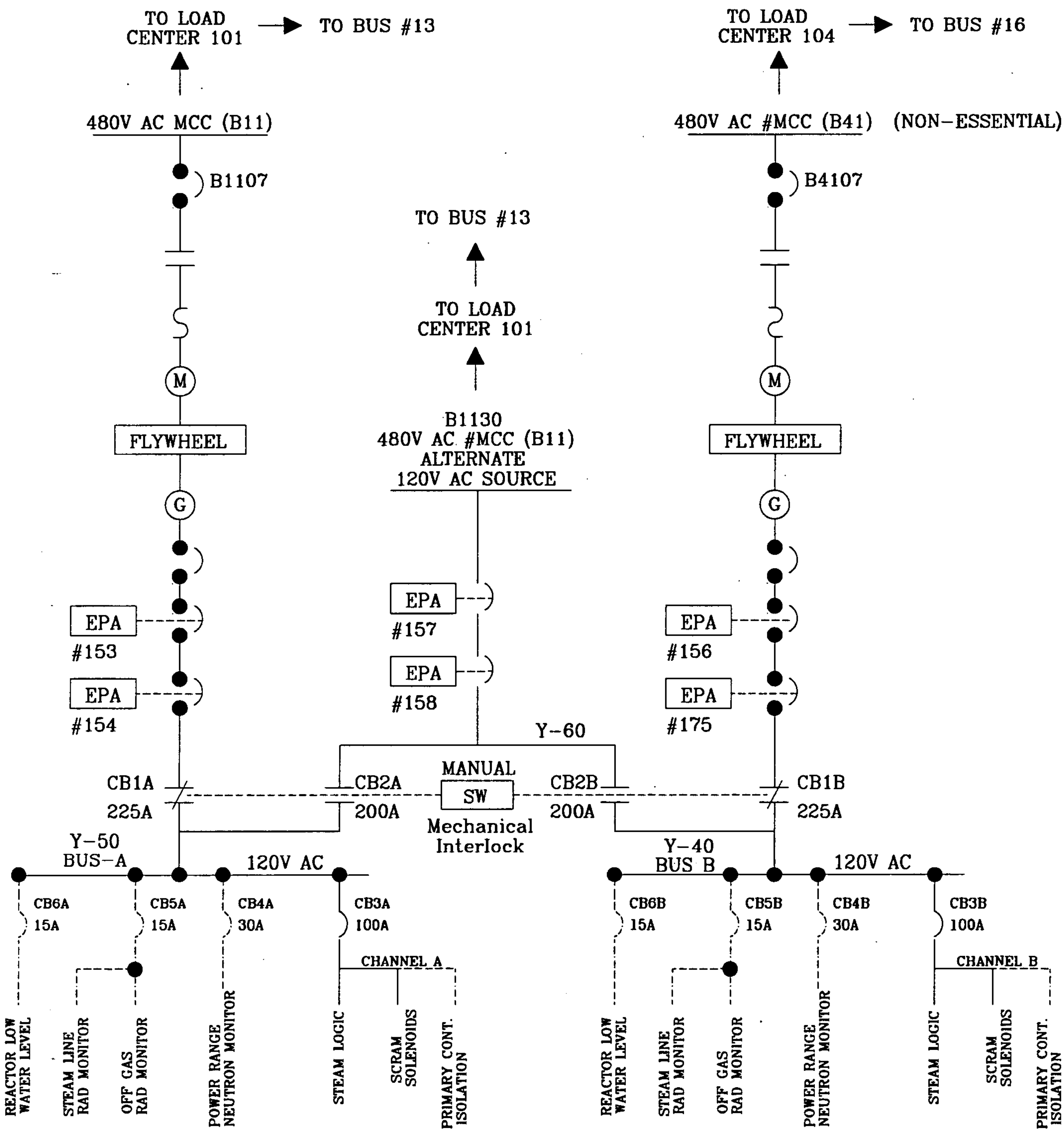


Figure No. 3.2-2
Reactor Protection System
Power Supply One Line Diagram

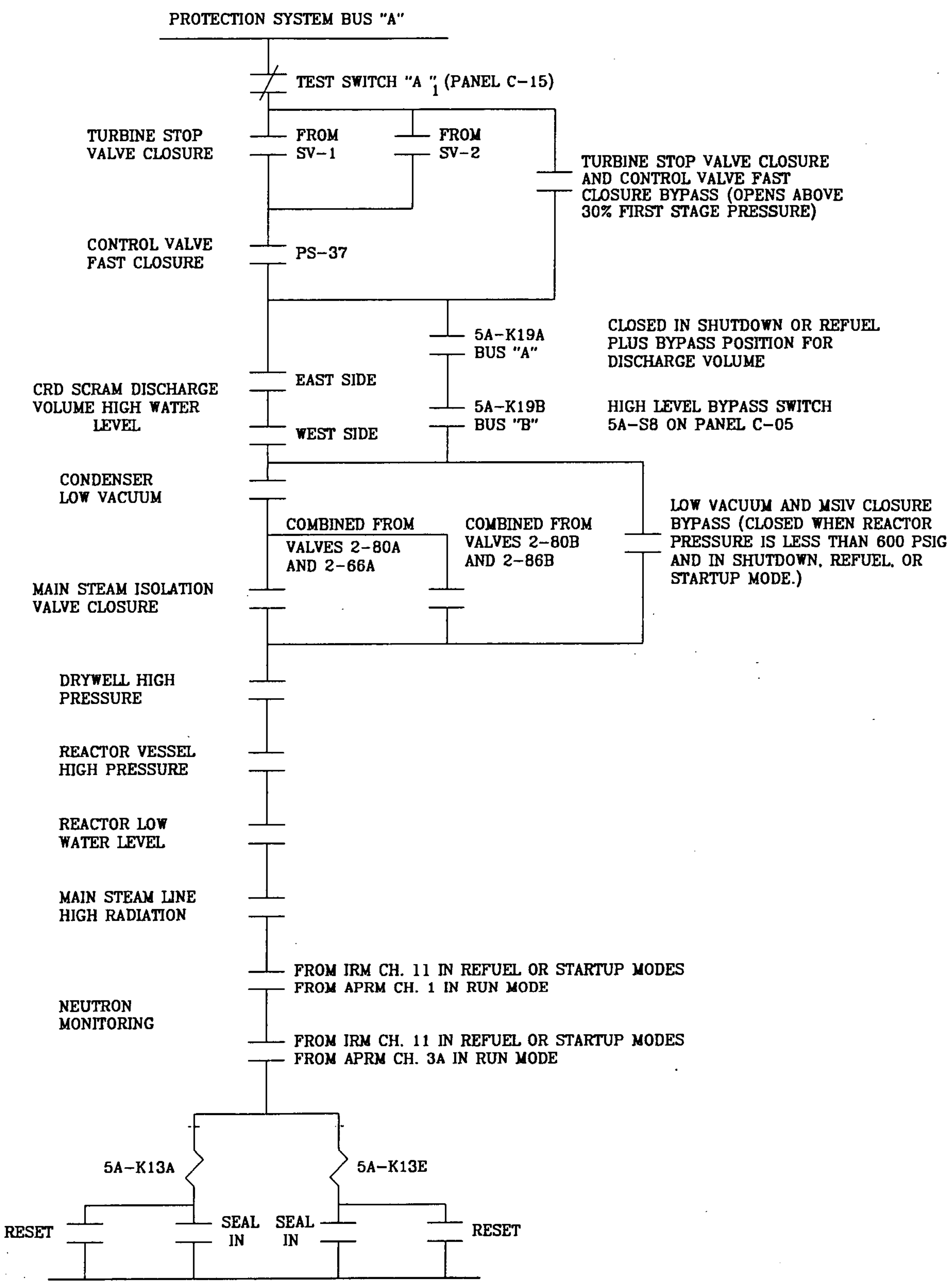


Figure No. 3.2-3
Scram Channel "A"

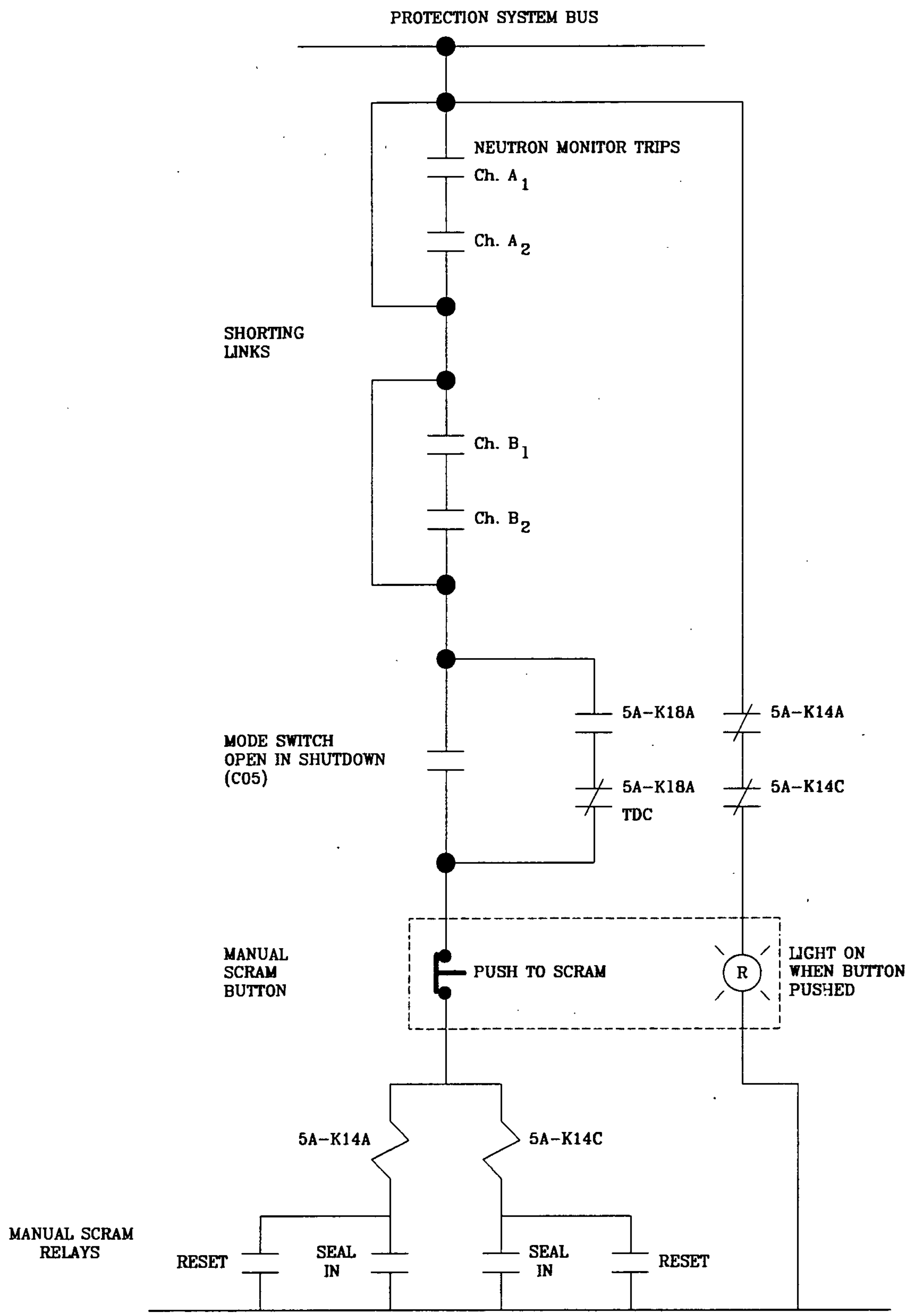


Figure No. 3.2-4
Manual Scram Channel "A"

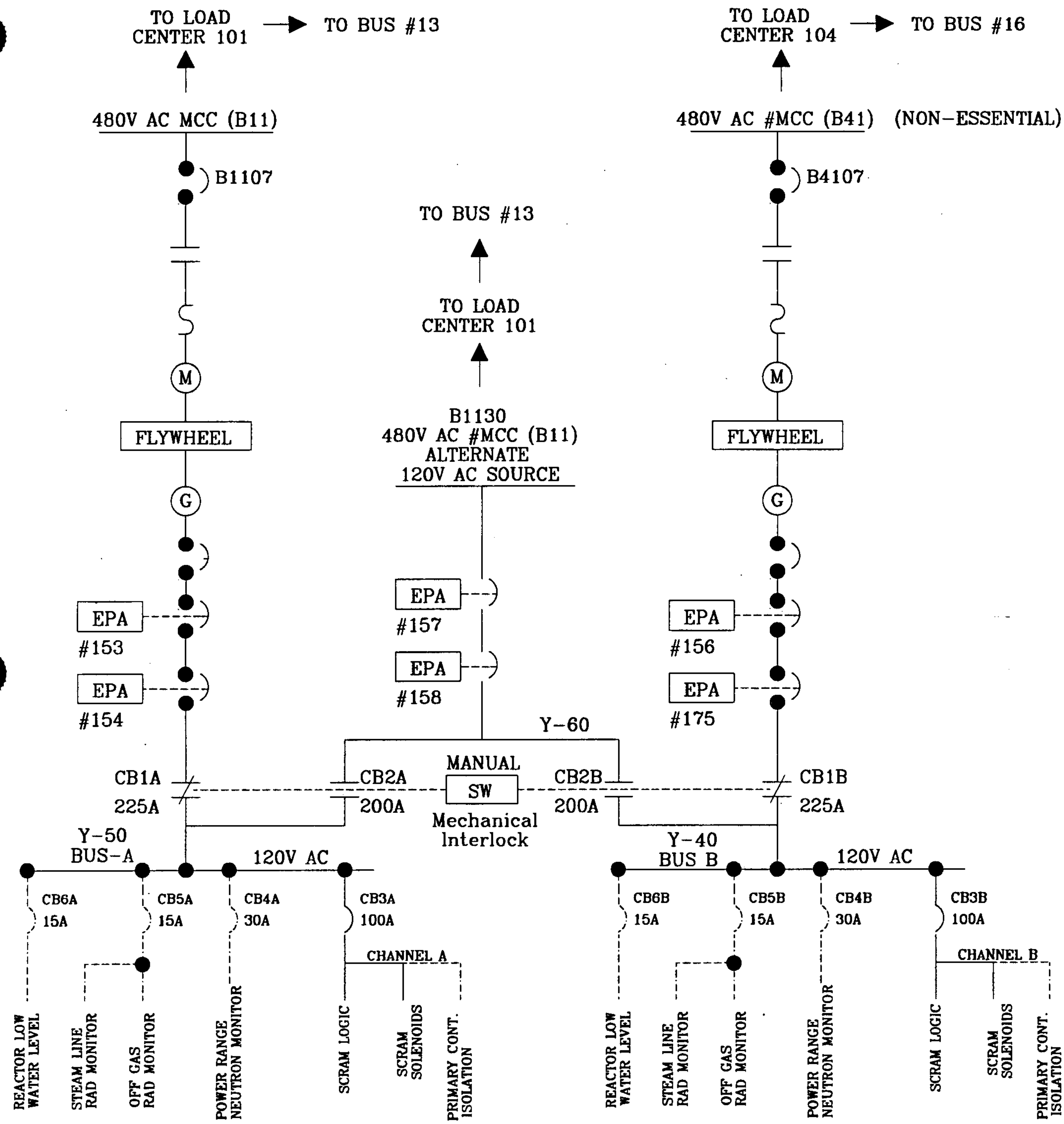


Figure No. 3.2-5
Reactor Protection System
Power Supply One Line Diagram

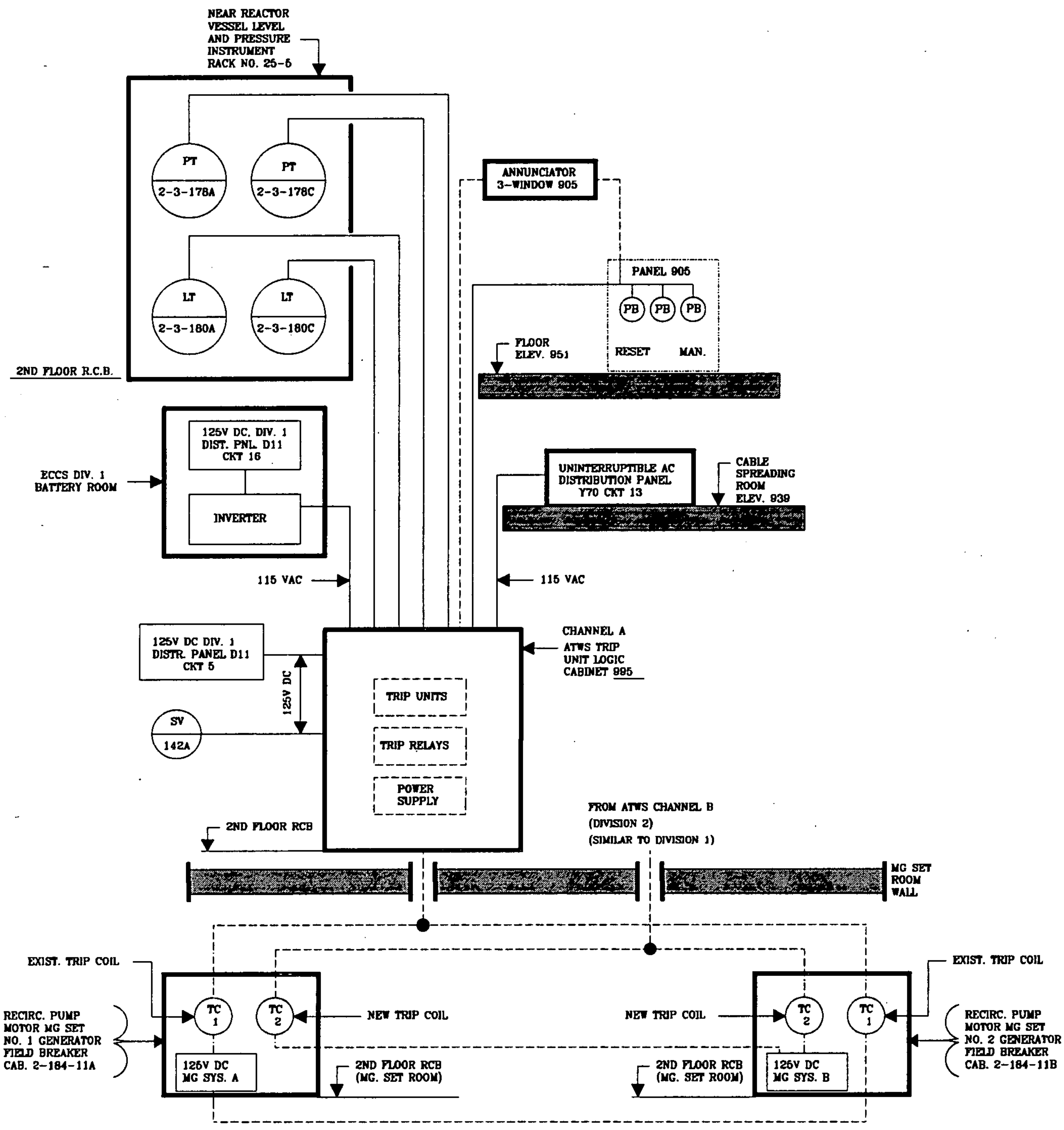


Figure No. 3.2-6
ATWS Interconnection Diagram

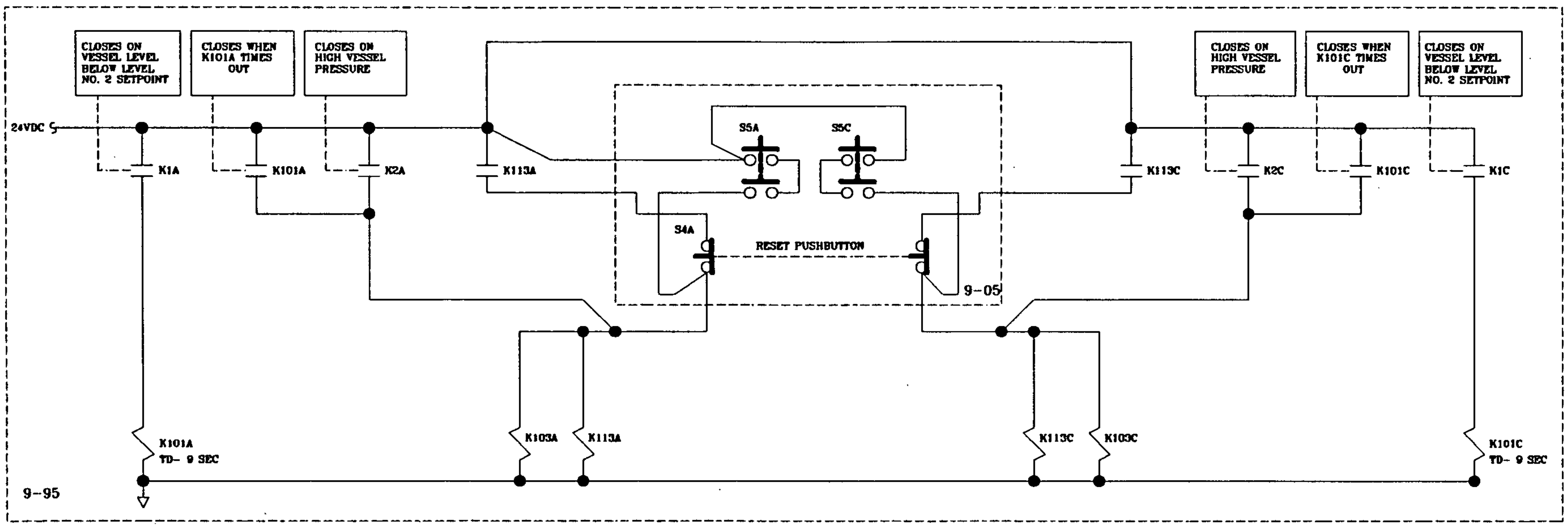


Figure No. 3.2-7
ATWS Division 1 Logic (Sheet 1)

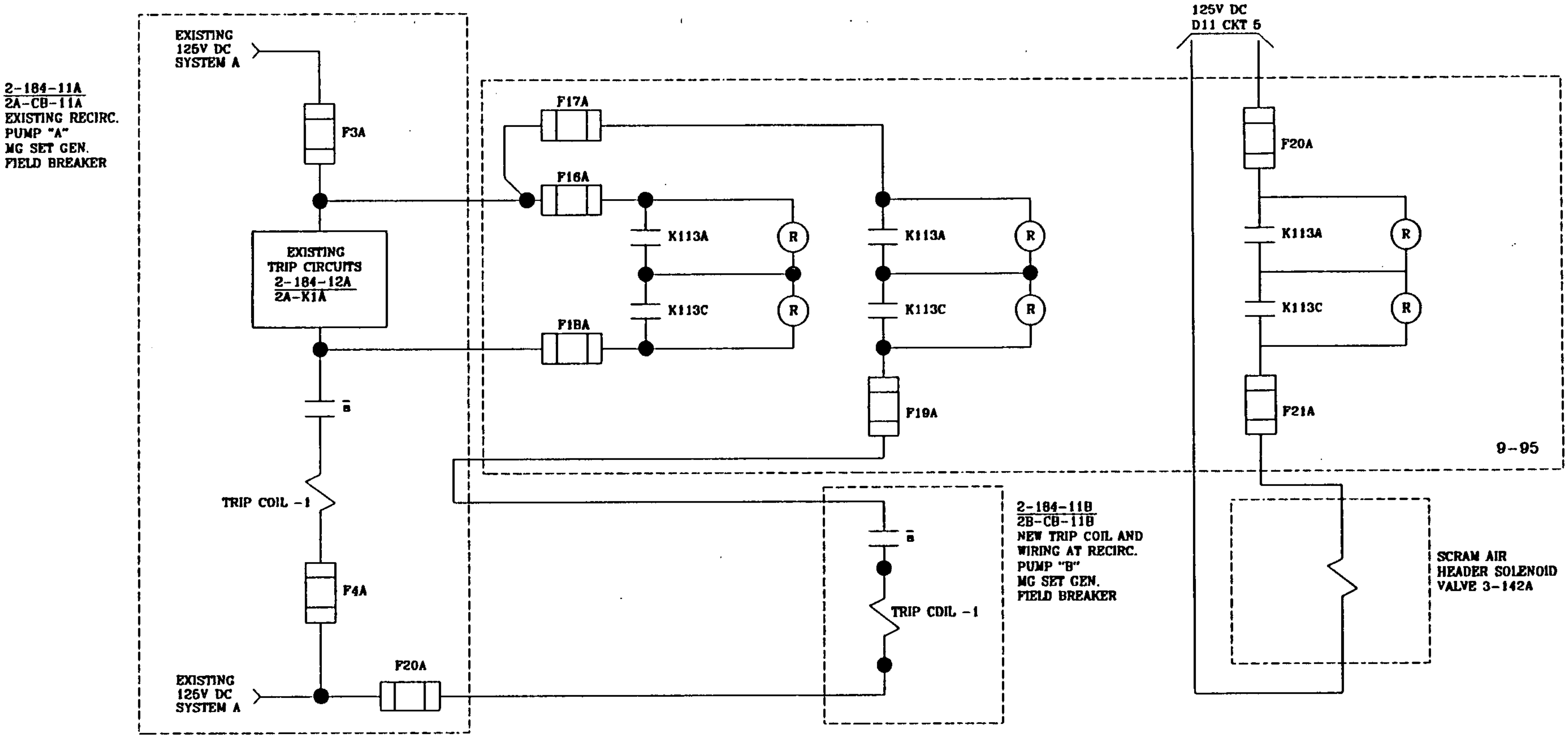
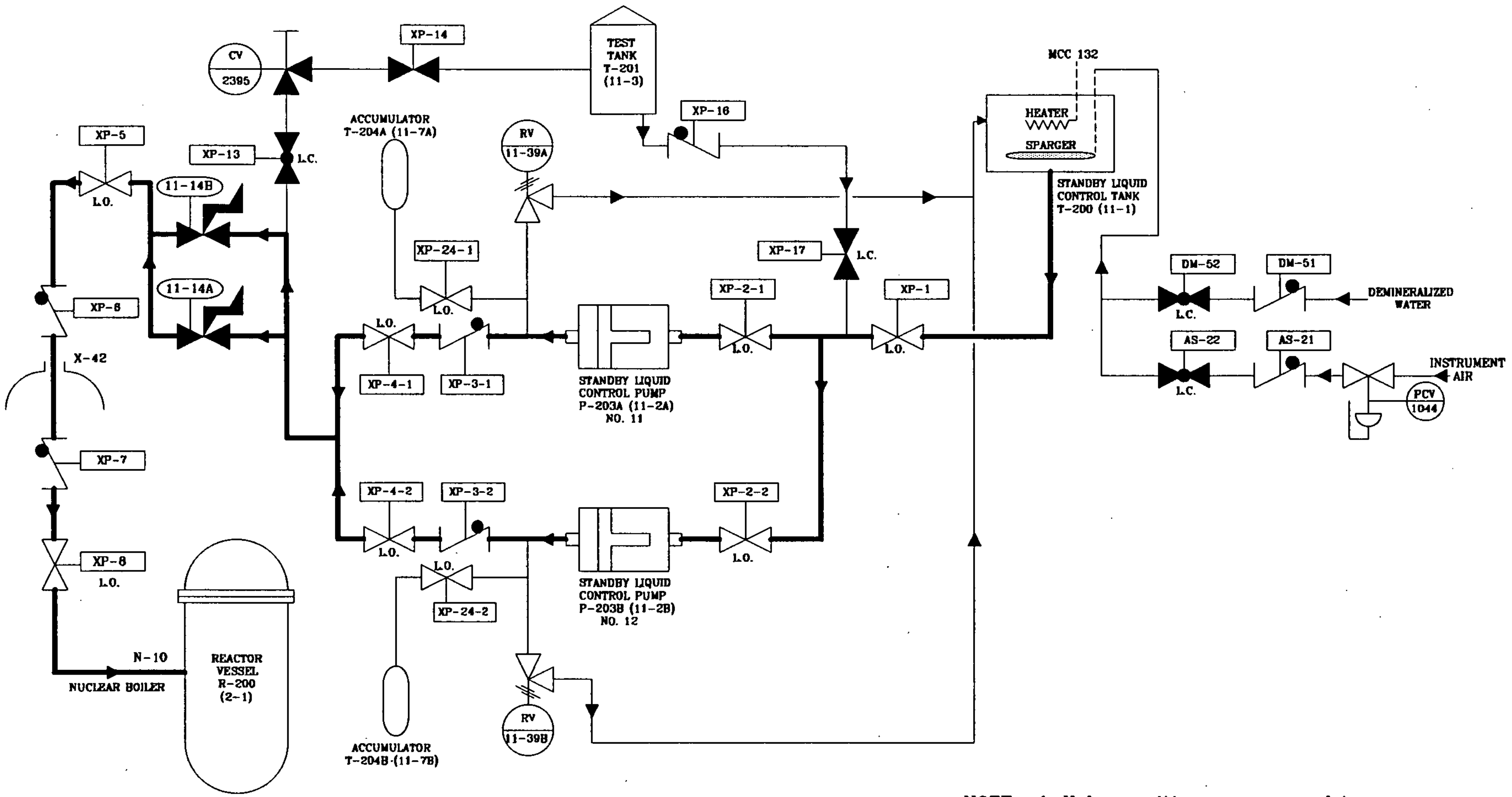


Figure No. 3.2-8
ATWS Division 1 Logic (Sheet 2)



NOTE: 1. Valve positions correspond to lineup for normal plant operation.

Figure No. 3.2-9
Standby Liquid Control

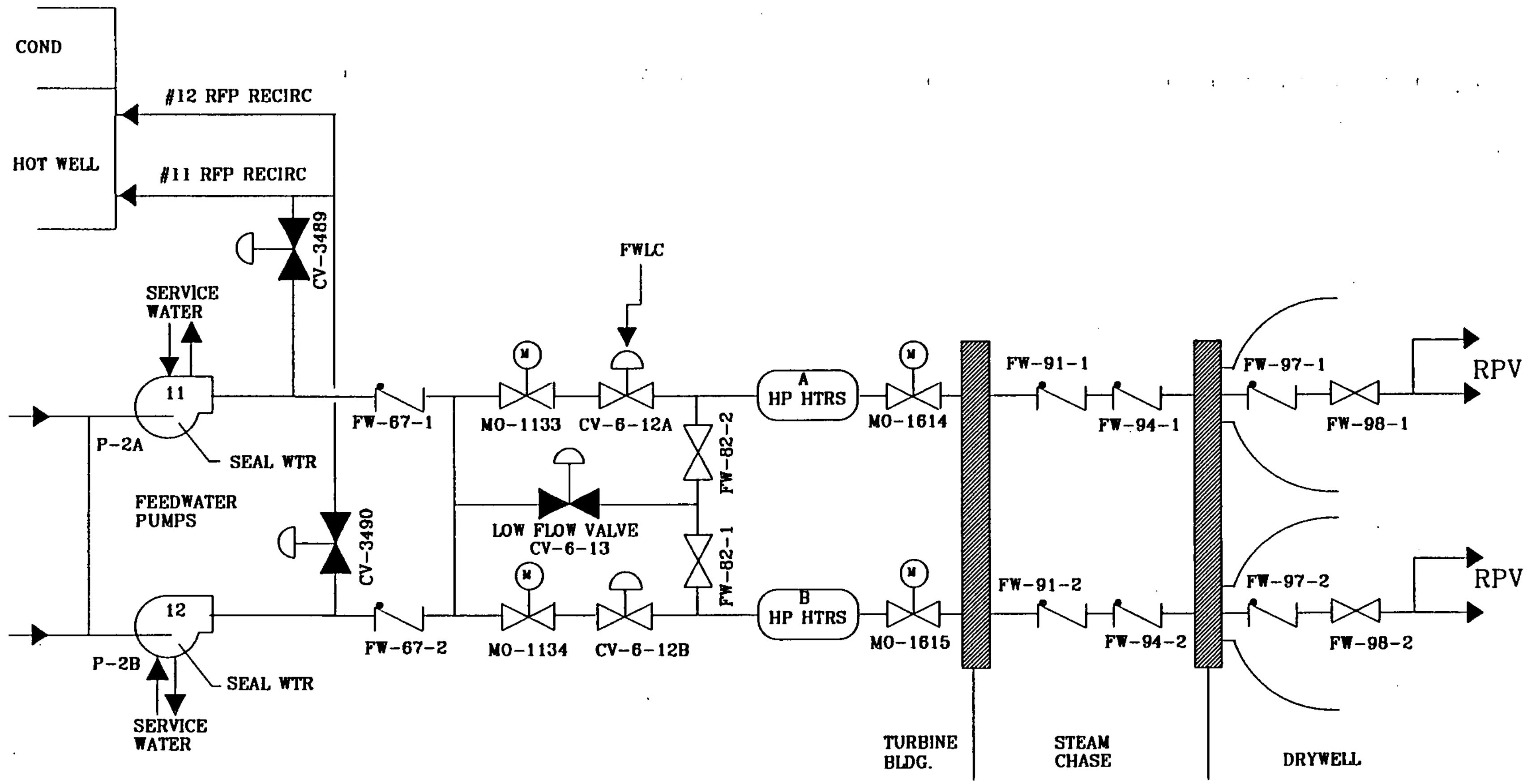
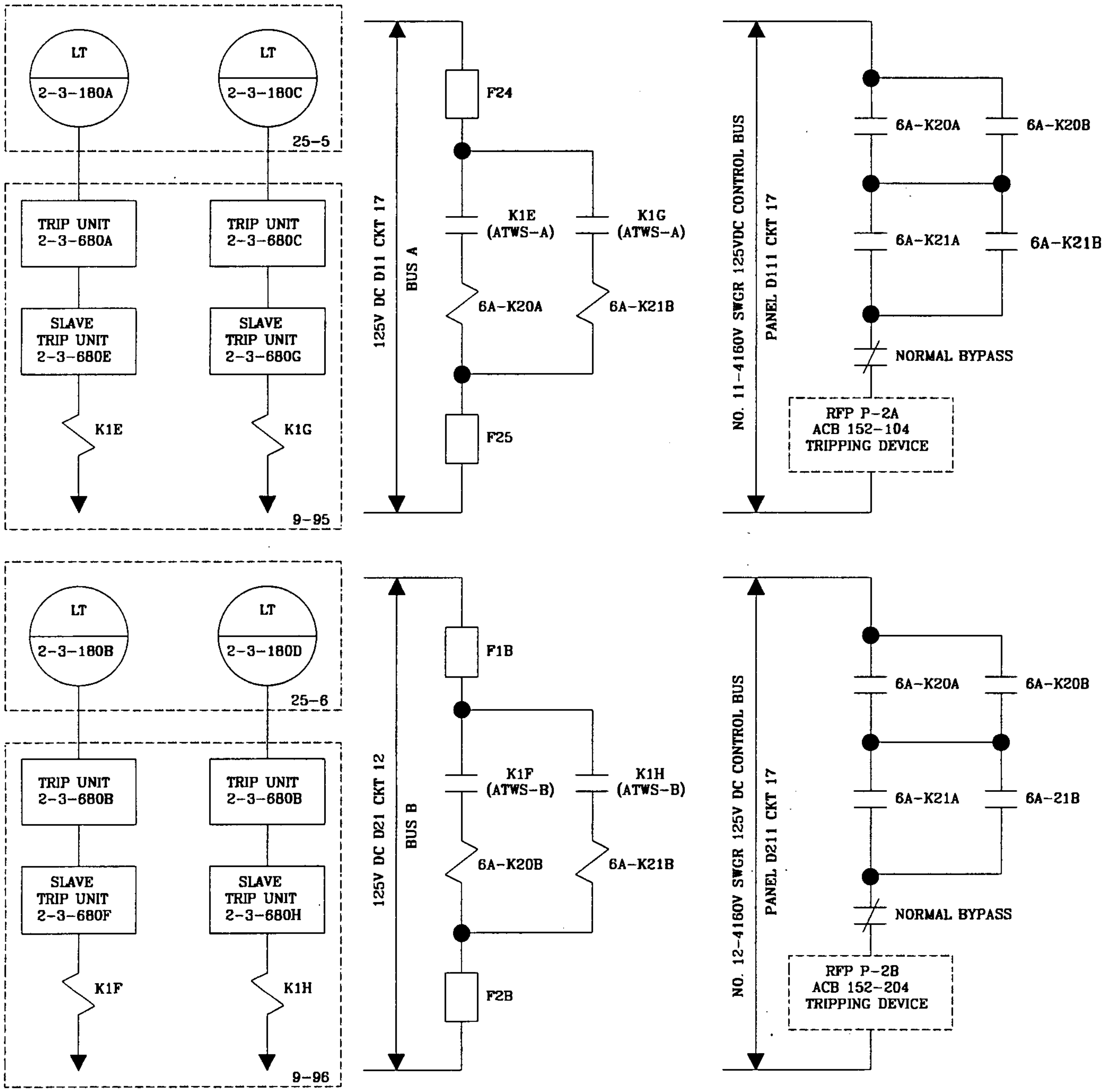


Figure No. 3.2-10
FEEDWATER



NOTE: 1. Similar logic for Turbine Lockout.

Figure No. 3.2-11
REACTOR FEEDWATER TRIP
ON HIGH WATER LEVEL

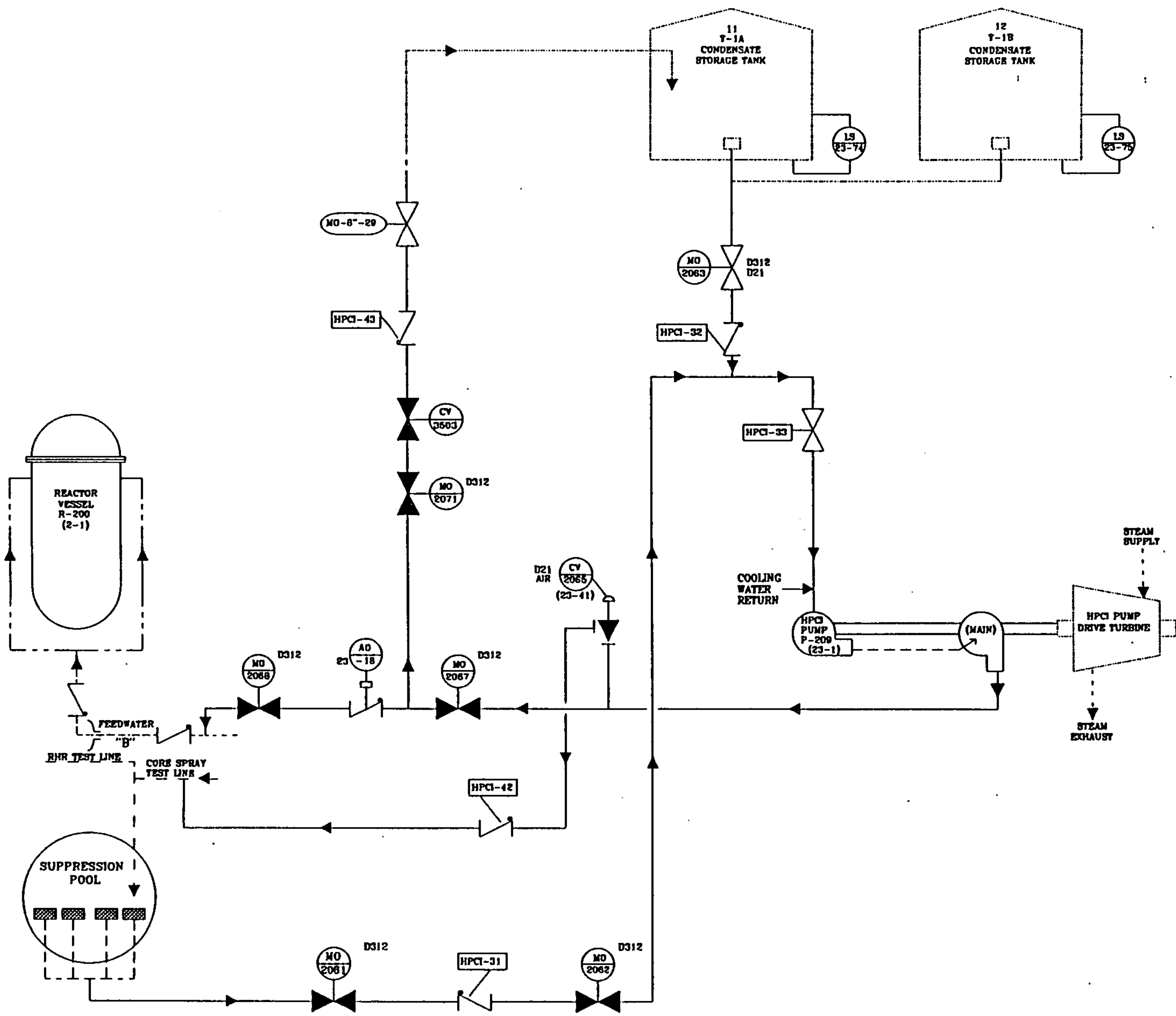


FIGURE NO. 3.2-12
HIGH PRESSURE COOLANT INJECTION
(HPCI)(WATER SIDE)

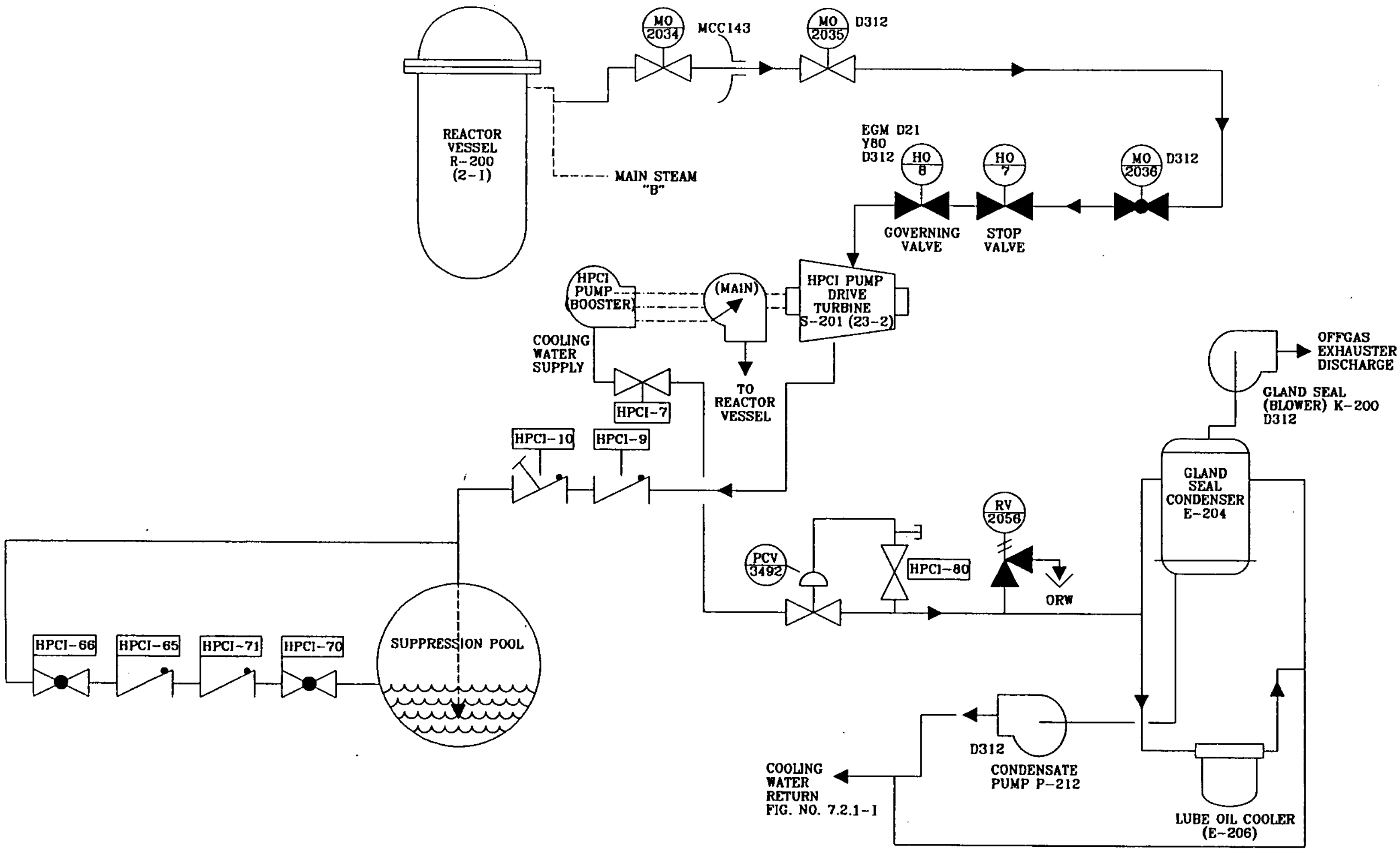


Figure No. 3.2-13
HIGH PRESSURE COOLANT INJECTION
(HPCI)(STEAM SIDE)

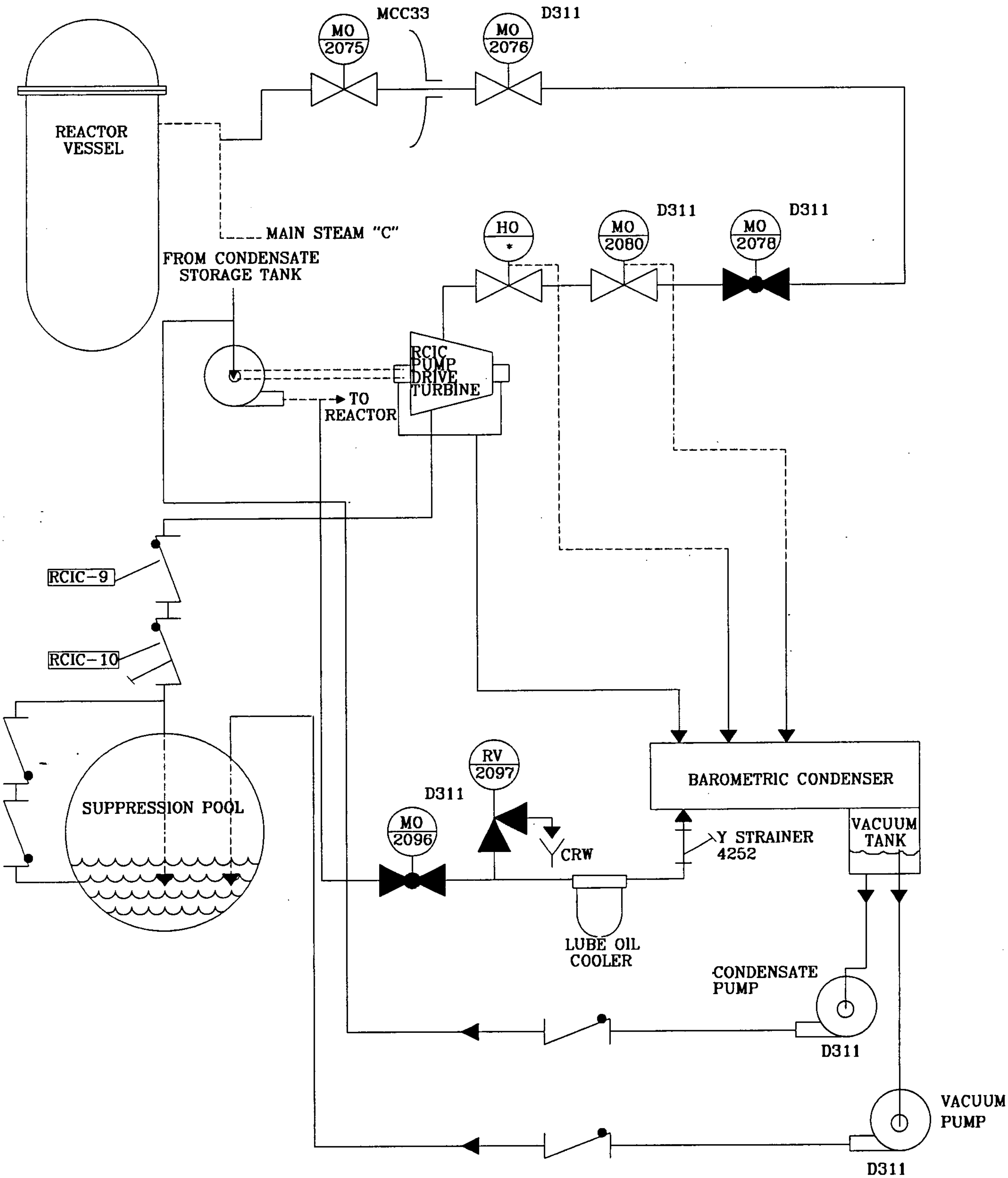


Figure No. 3.2-14
REACTOR CORE ISOLATION COOLING
(RCIC)(STEAM SIDE)

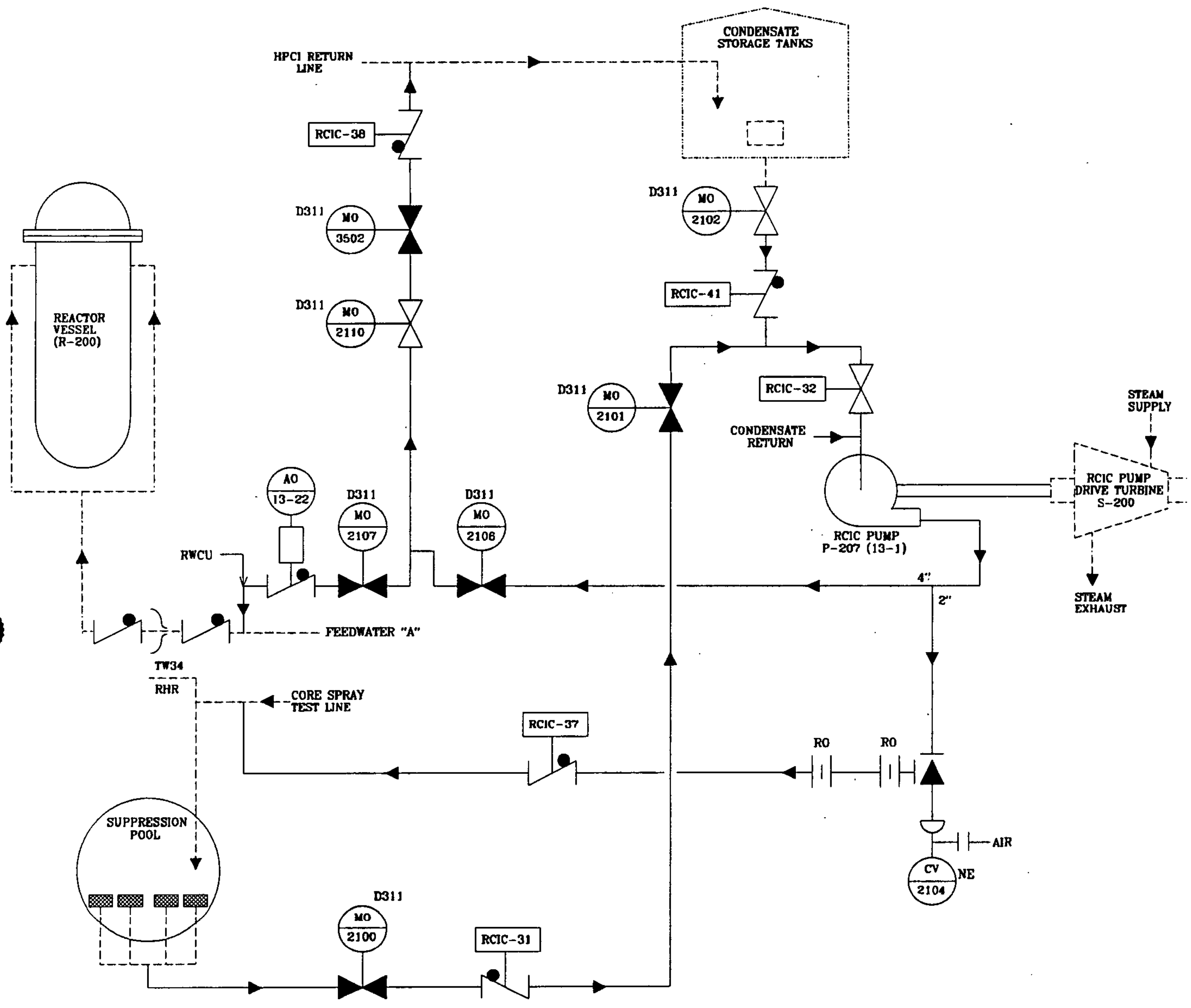
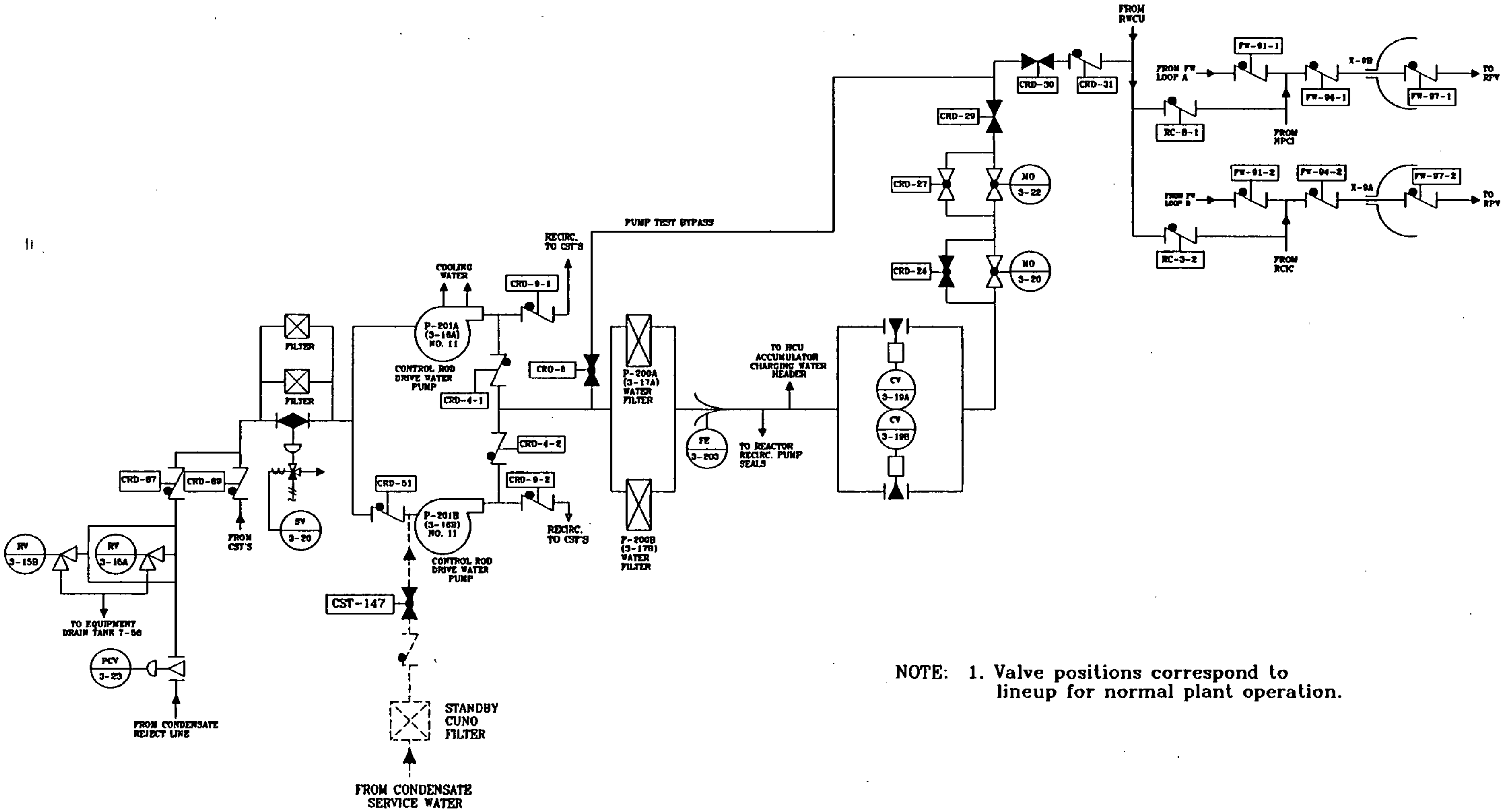
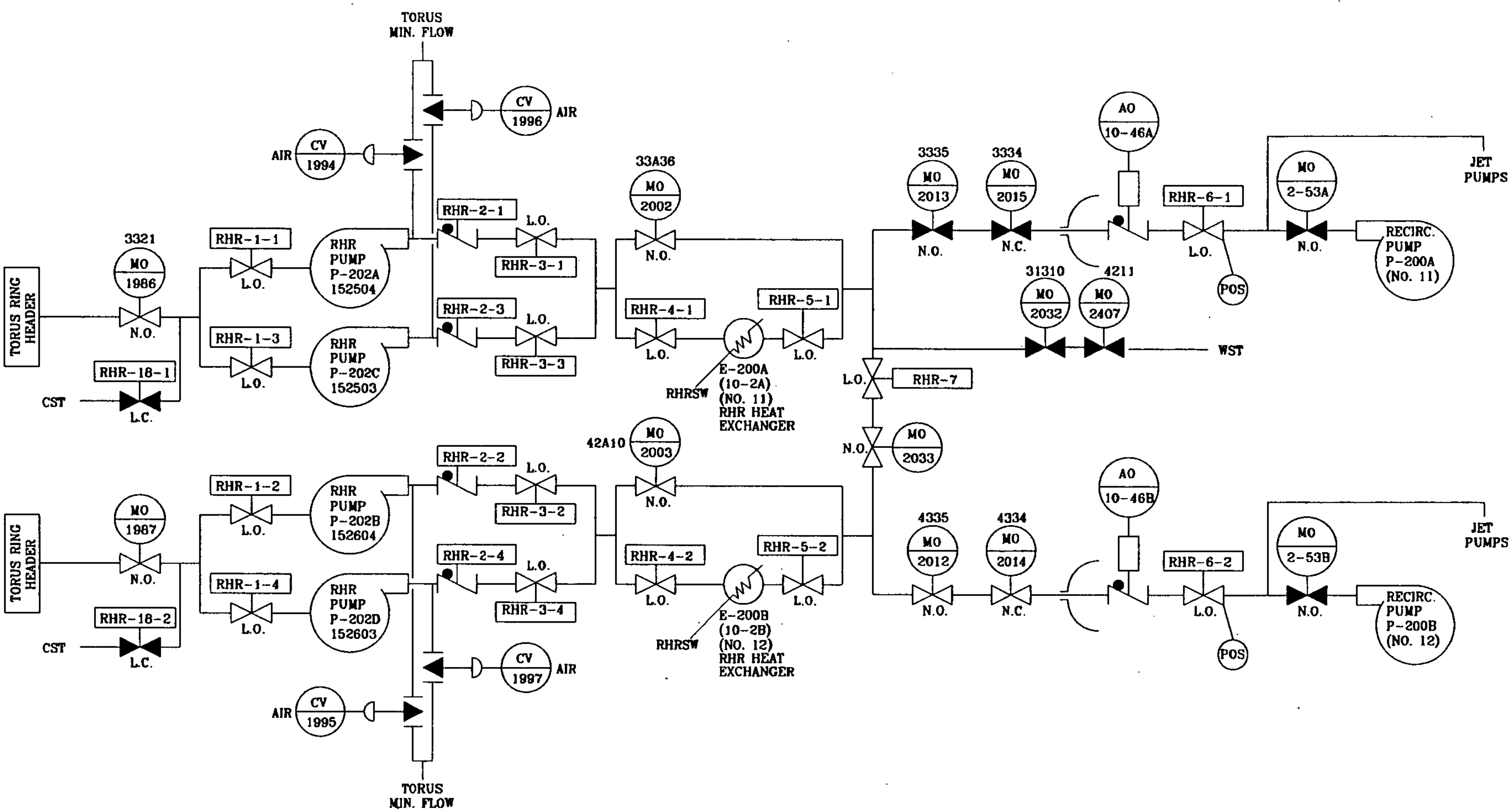


Figure No. 3.2-15
Reactor Core Isolation Cooling (RCIC)(Water Side)



NOTE: 1. Valve positions correspond to lineup for normal plant operation.

Figure No. 3.2-16
CRD HYDRAULIC SYSTEM EMERGENCY
FLOWPATH TO REACTOR PRESSURE VESSEL



NOTE: 1. Shown in configuration required for LPCI. Normally MO-2012 and MO-2013 are open and MO-2014 and MO-2015 are closed. LPCI Loop Select Logic selects the Recirc. Loop which does not have a break or the "B" loop if there is not a break. This closes the injection valves of the unselected loop and the Recirc. pump discharge valves. The HX bypass valves are closed as soon as possible after LPCI injection initiates.

Figure No. 3.2-17
Residual Heat Removal System
(LPCI Mode)

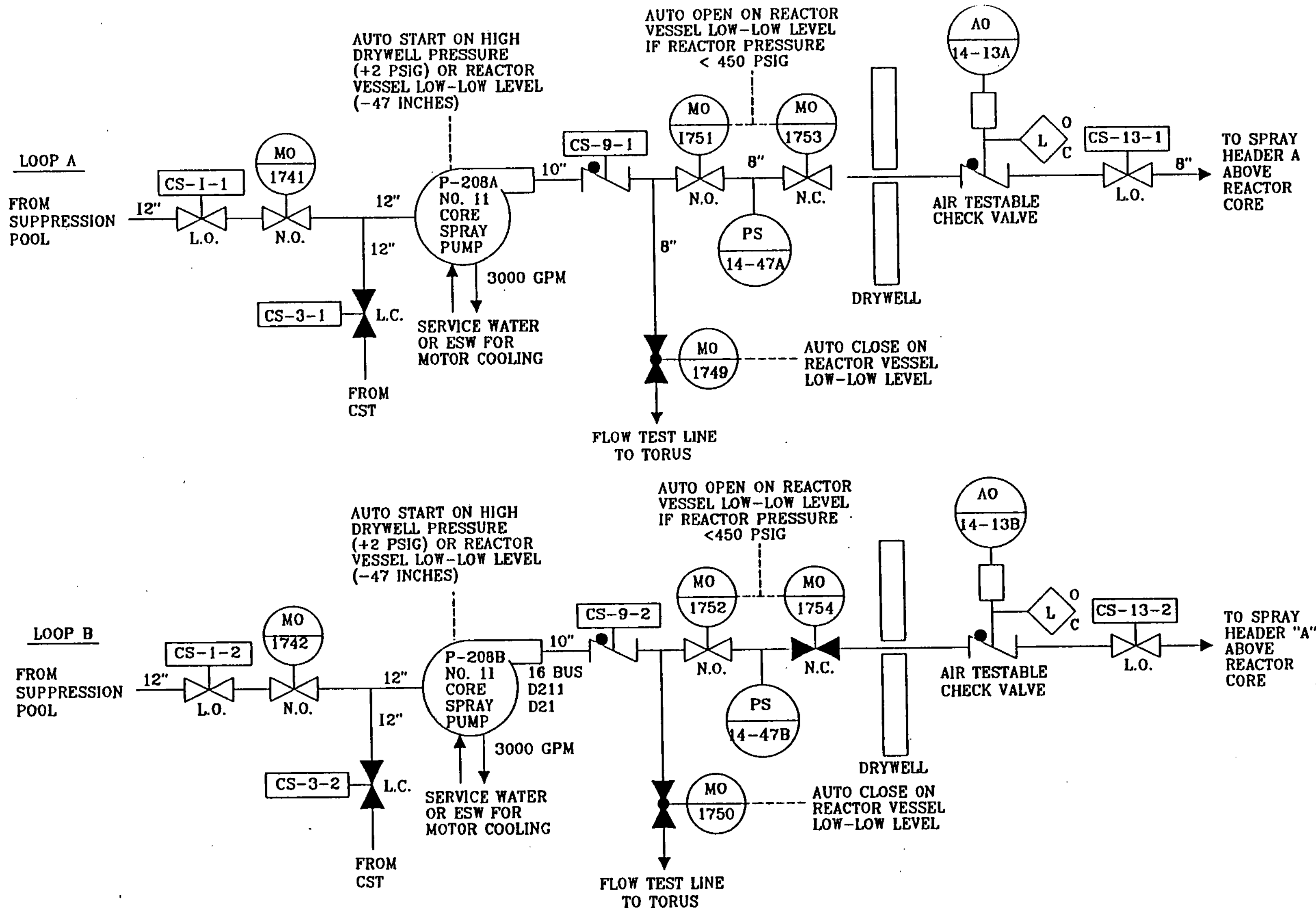


Figure No. 3.2-18
CORE SPRAY SYSTEM (SIMPLIFIED DRAWING)

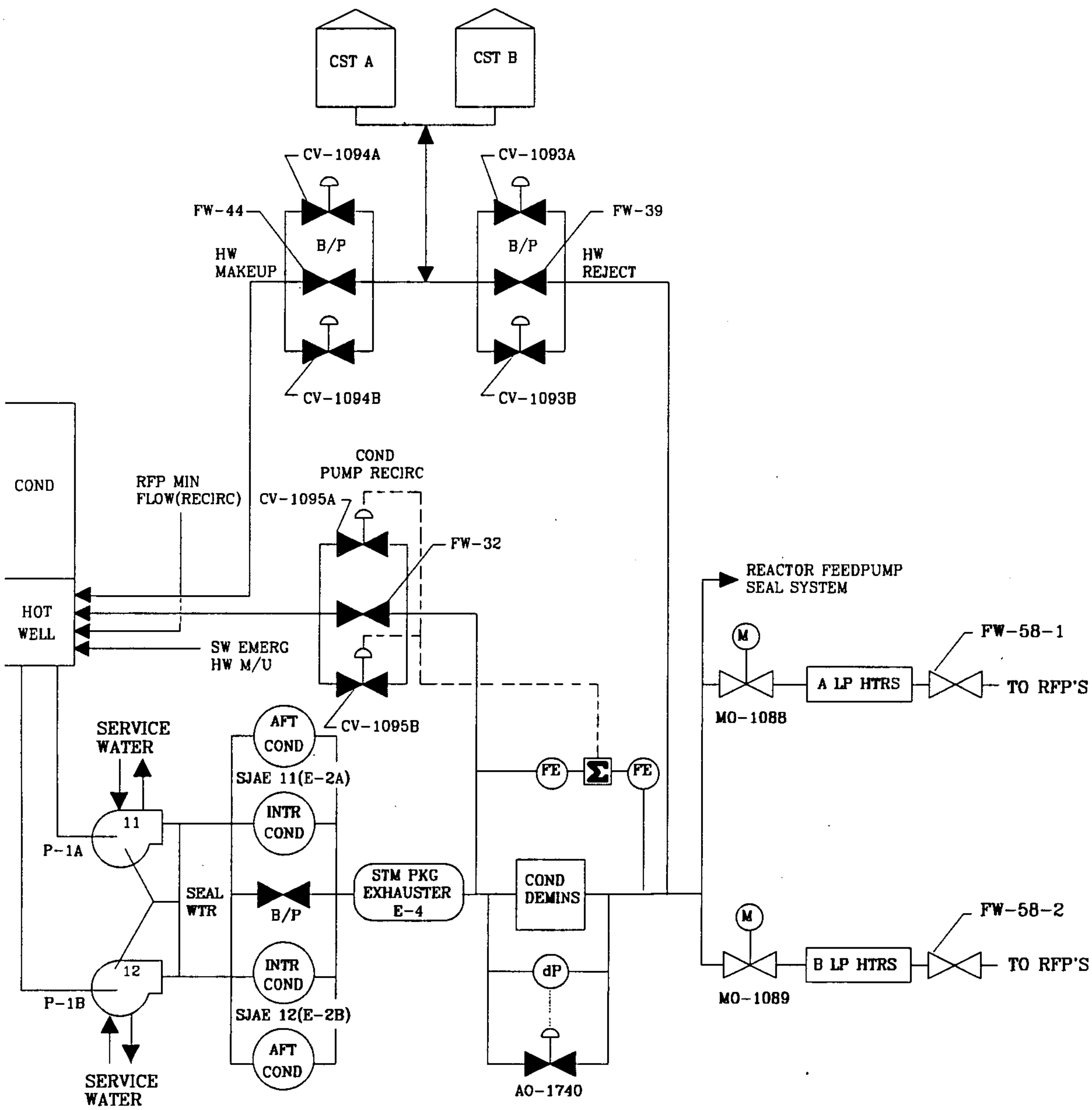


Figure No. 3.2-19
CONDENSATE

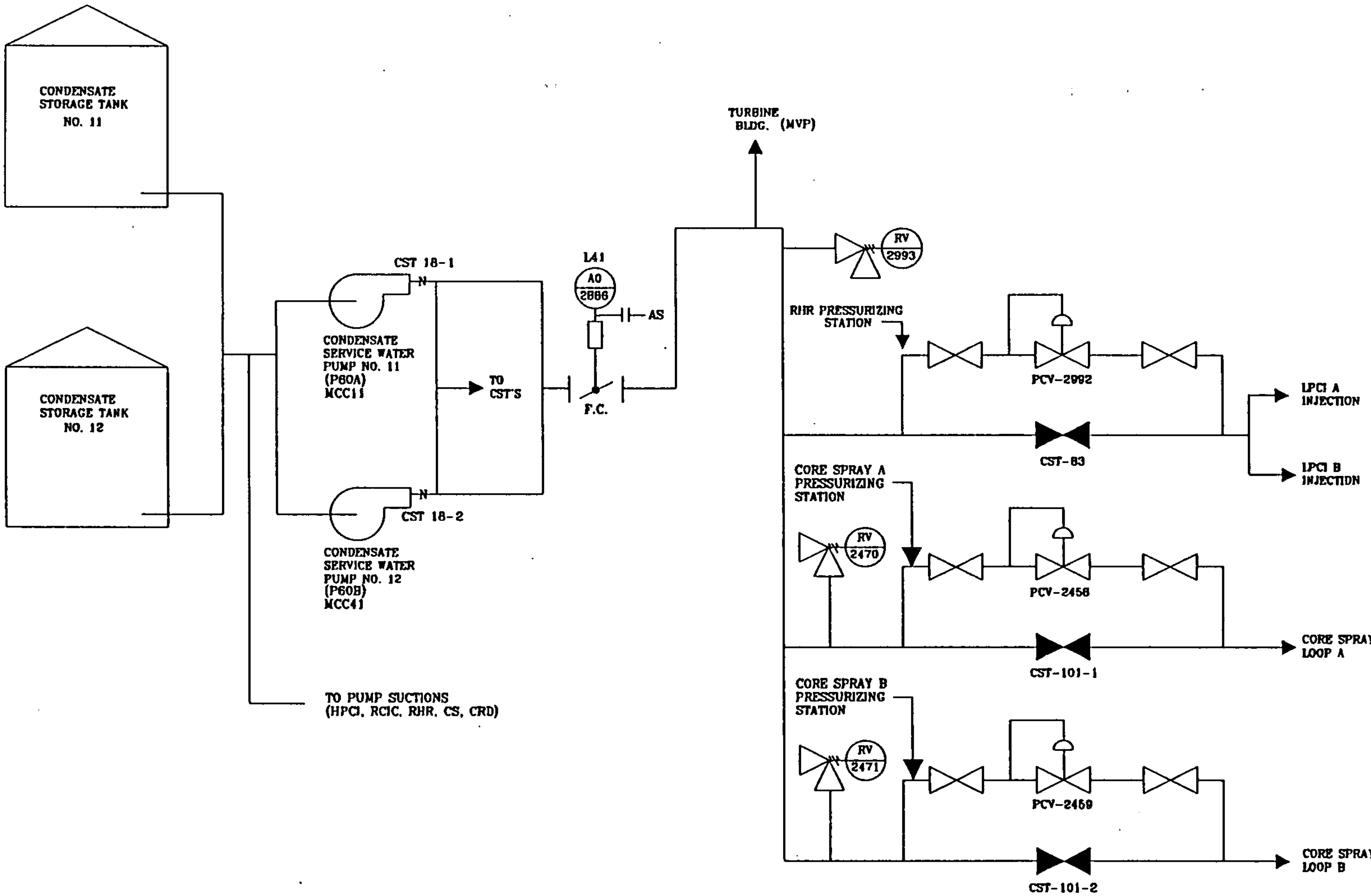


Figure No. 3.2-20
Condensate Service Water System

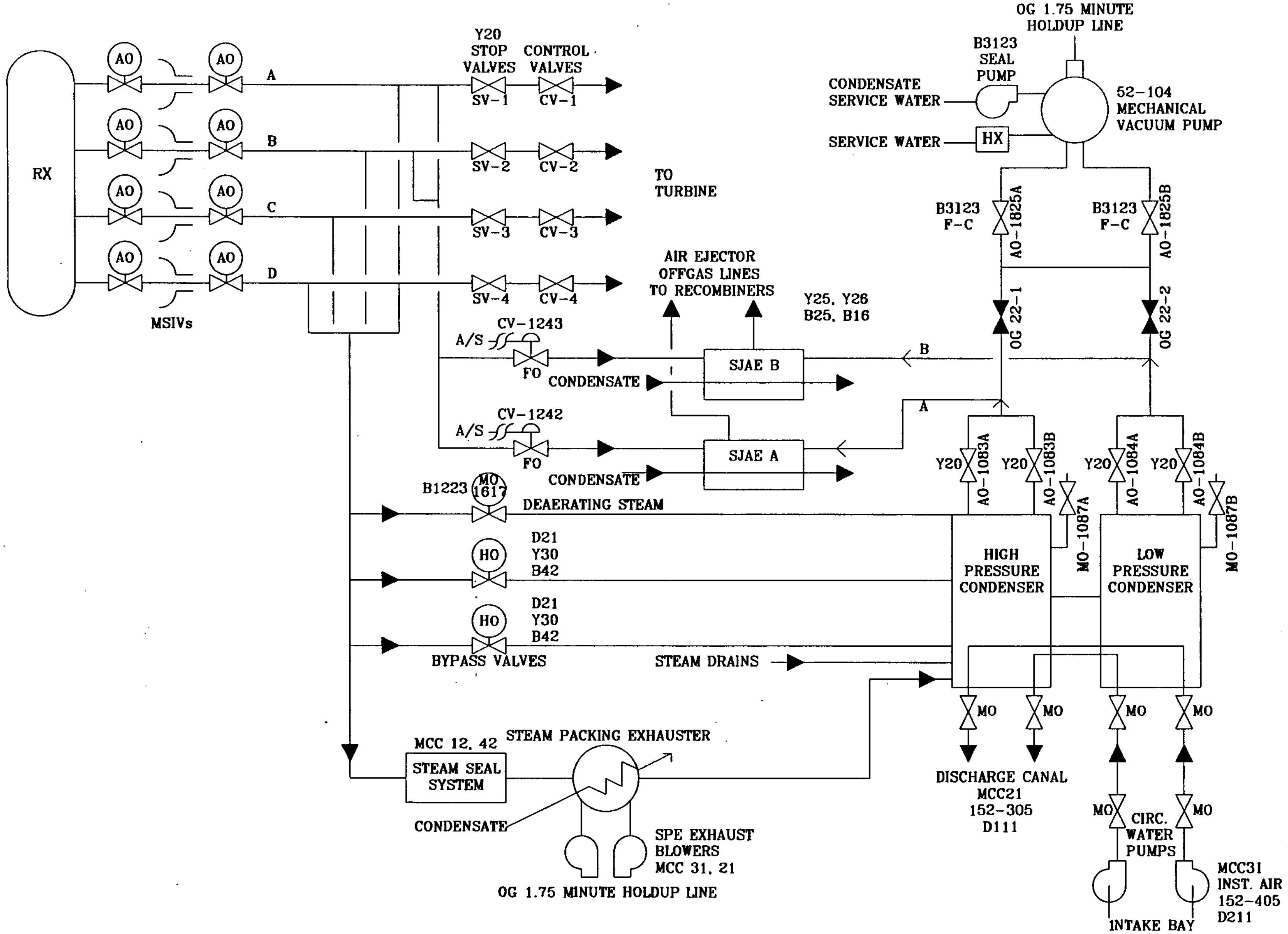
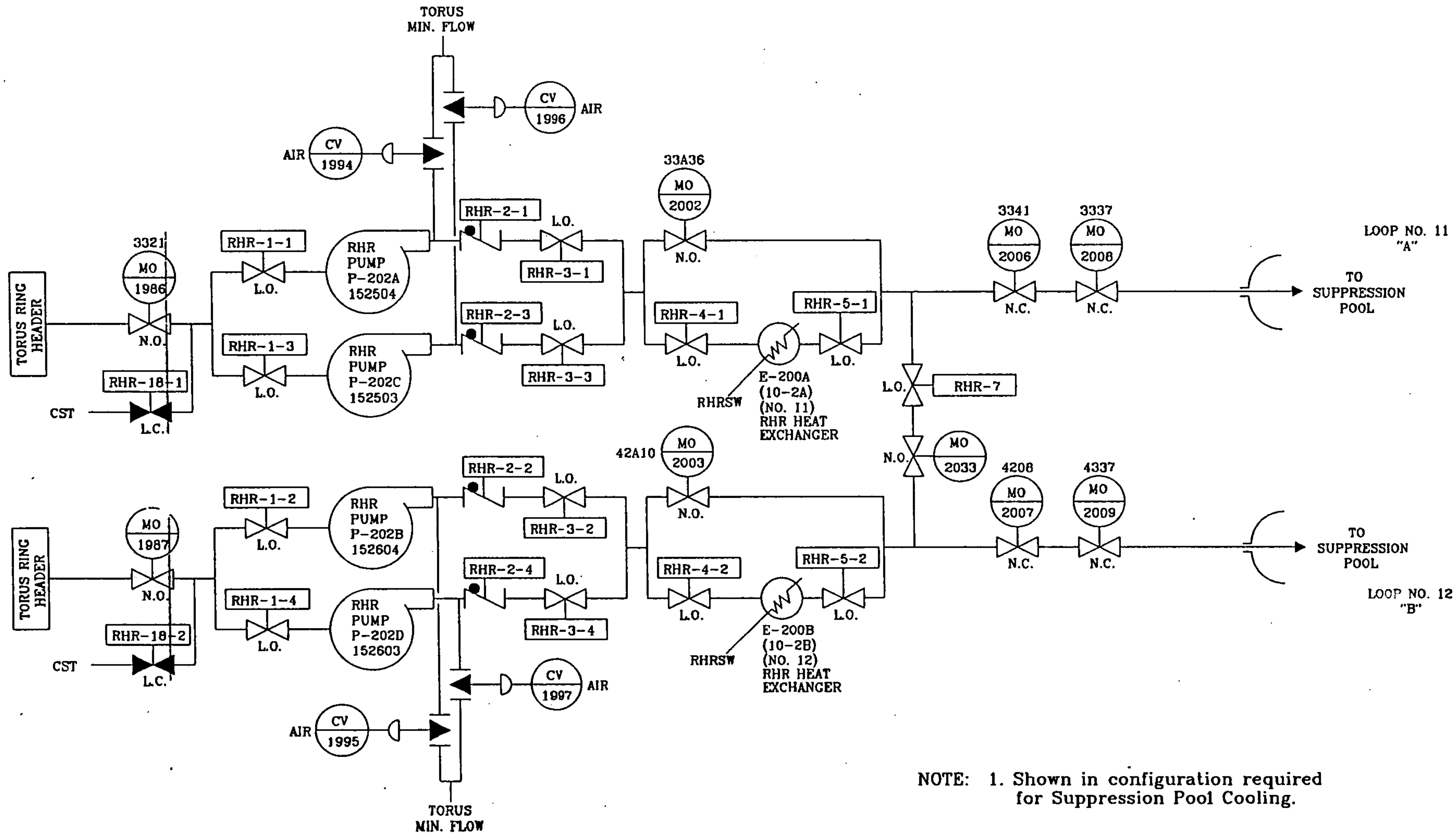


Figure No. 3.2-21
MAIN CONDENSER



NOTE: 1. Shown in configuration required for Suppression Pool Cooling.

Figure No. 3.2-22
Residual Heat Removal System
(Suppression Pool Cooling Mode)

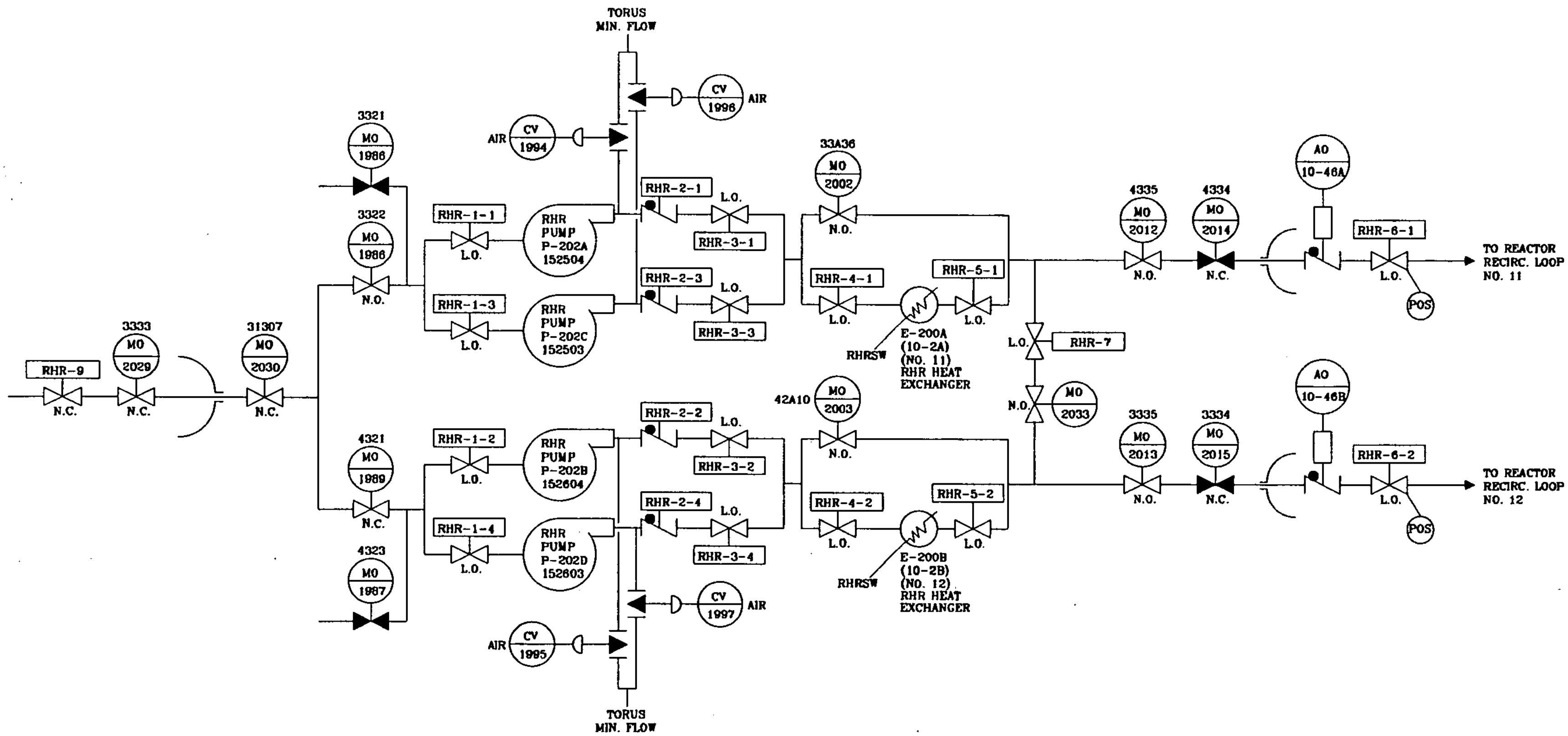


Figure No. 3.2-23
Residual Heat Removal System
(Shutdown Cooling Mode)

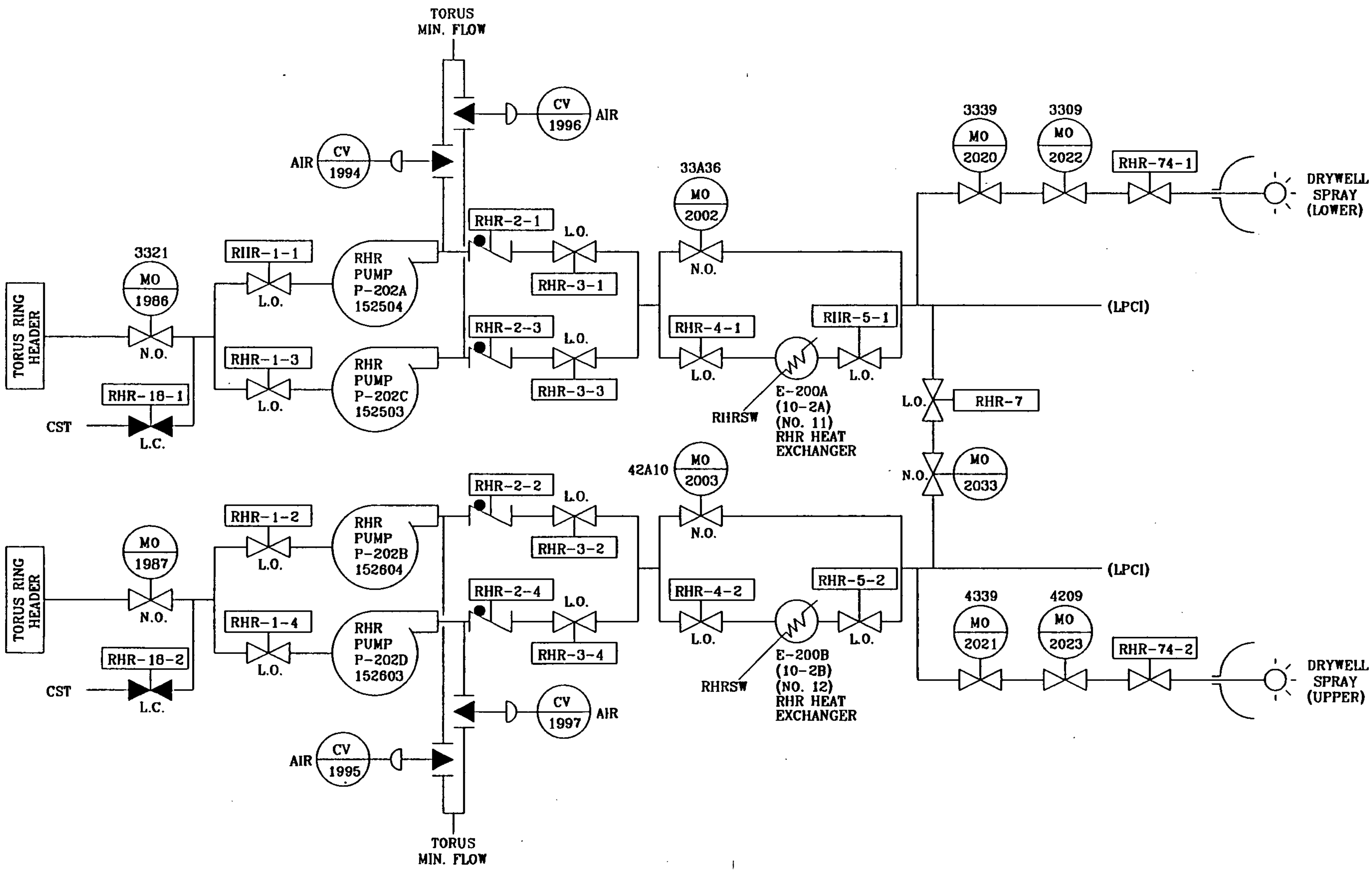


Figure No. 3.2-24
Residual Heat Removal System
(Drywell Spray)

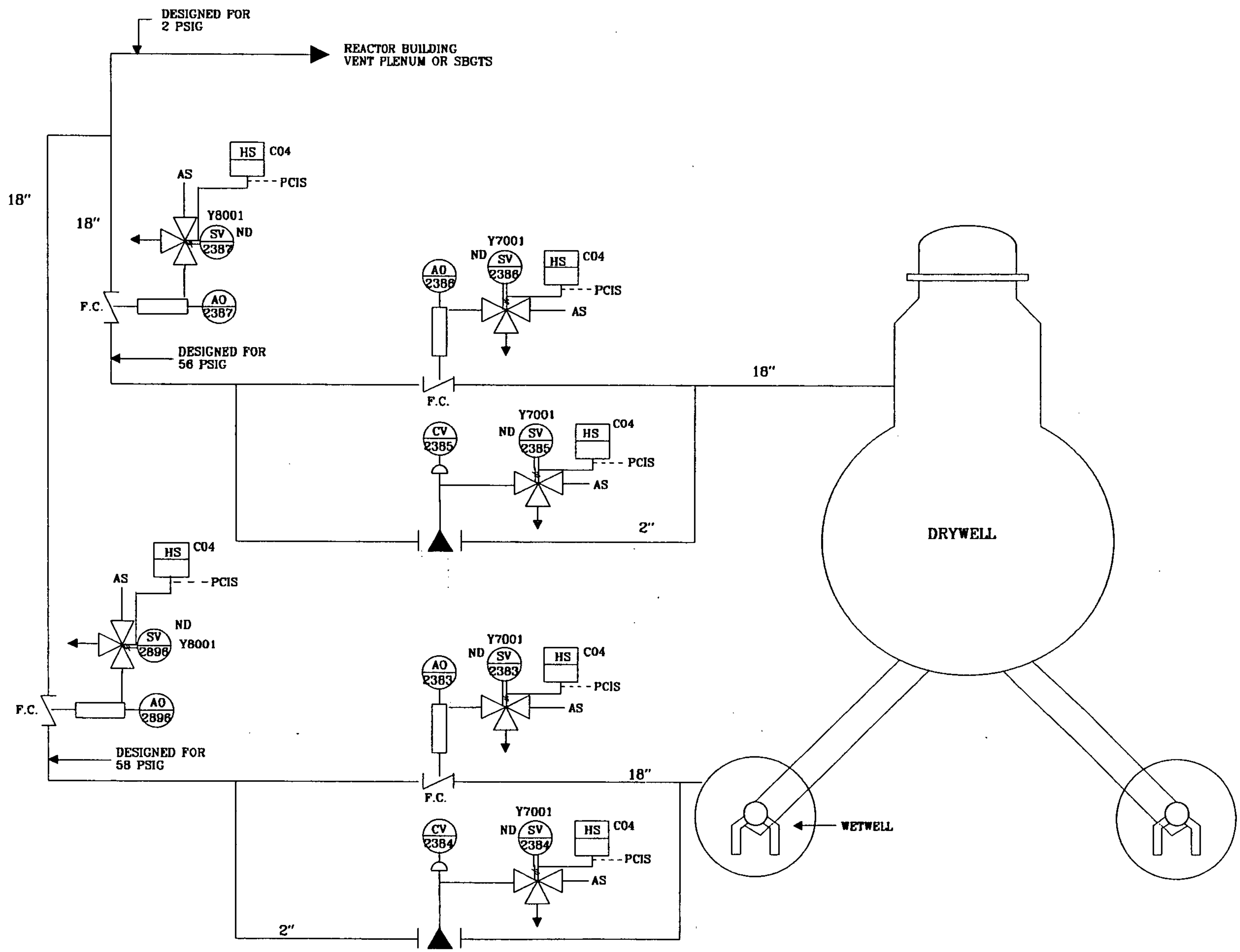


Figure No. 3.2-25
Containment Venting

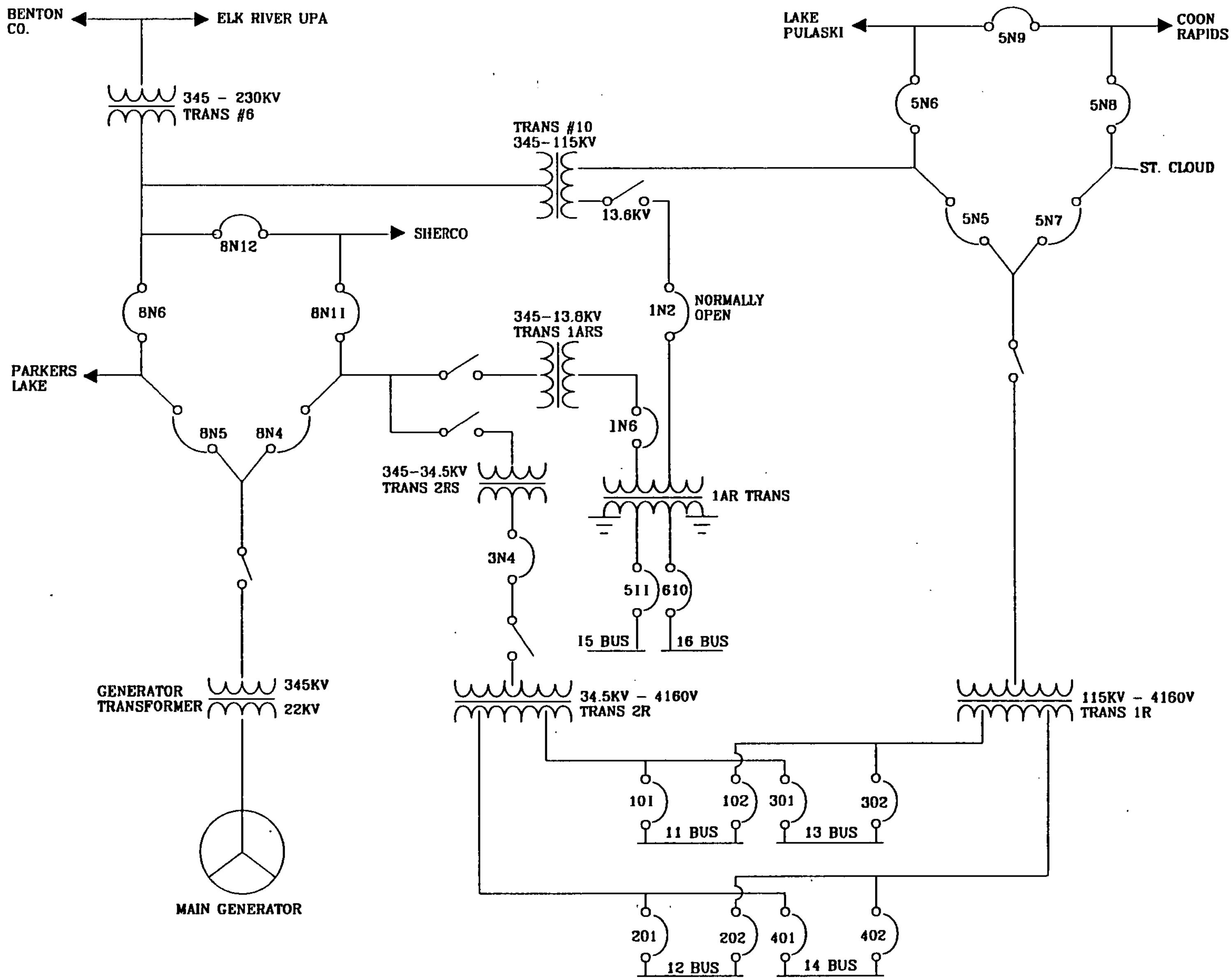


Figure No. 3.2-26
OFFSITE AC POWER

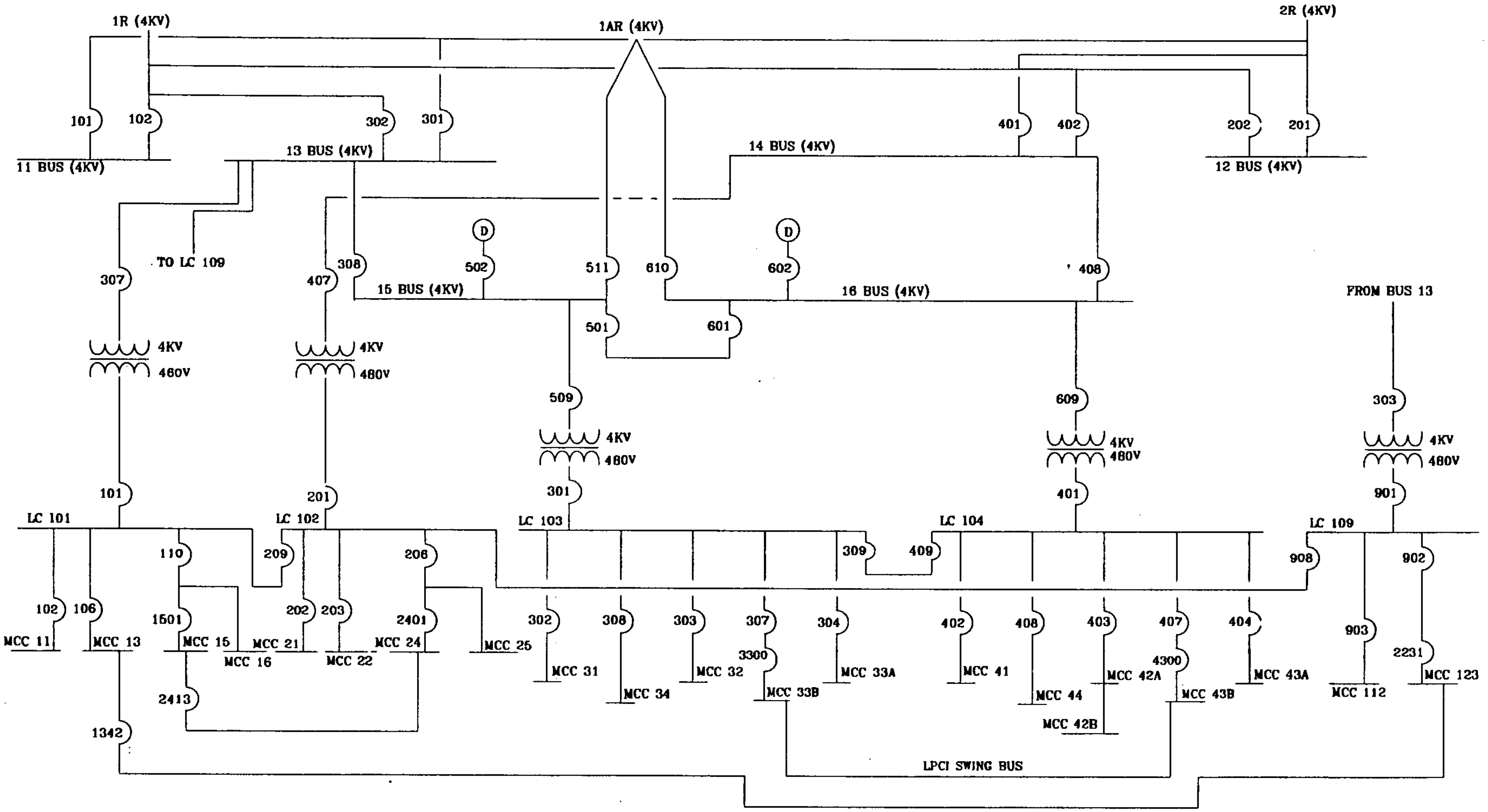


Figure No. 3.2-27
Onsite AC Power

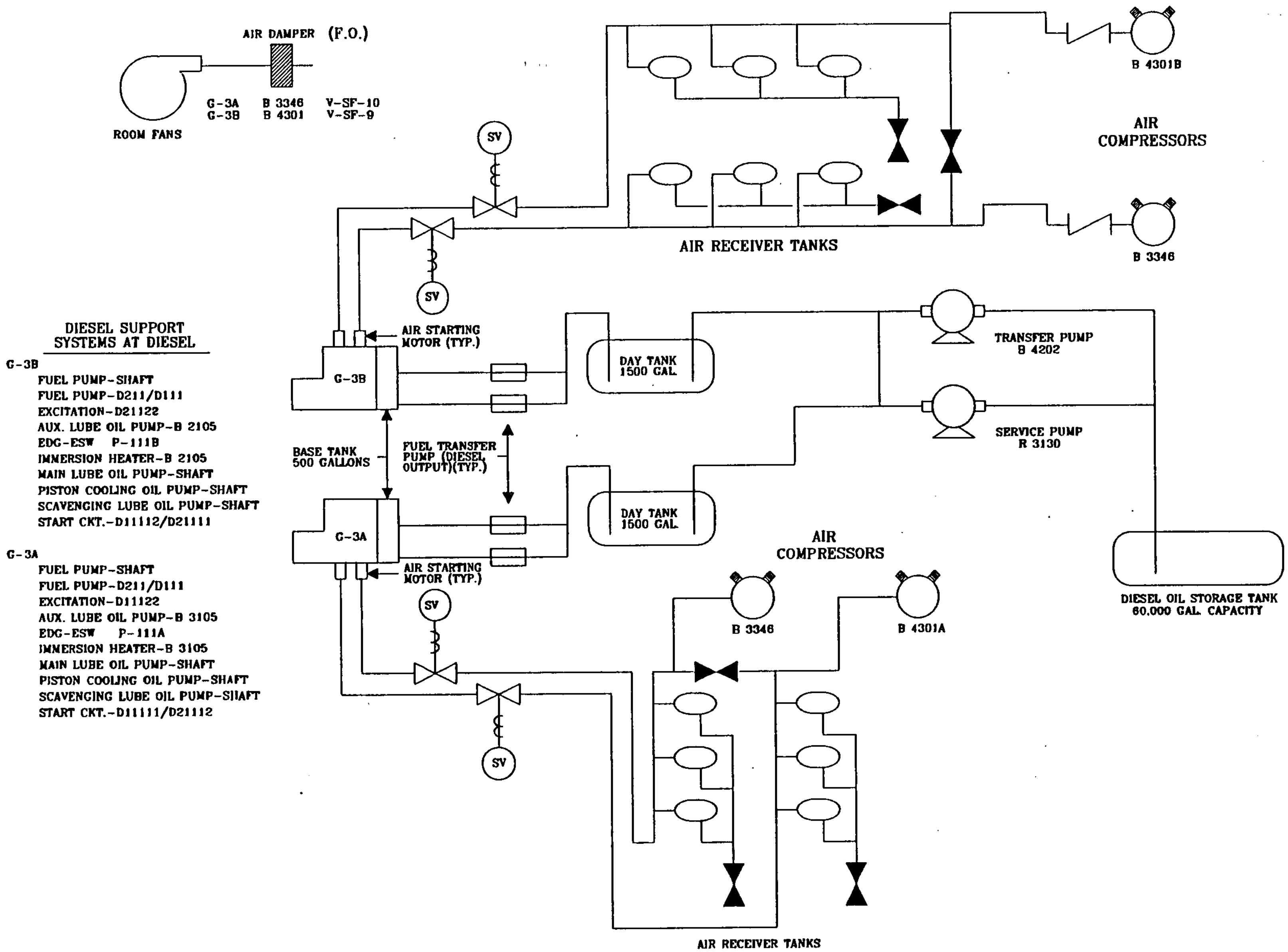


Figure No. 3.2-28
Emergency Diesel Generators

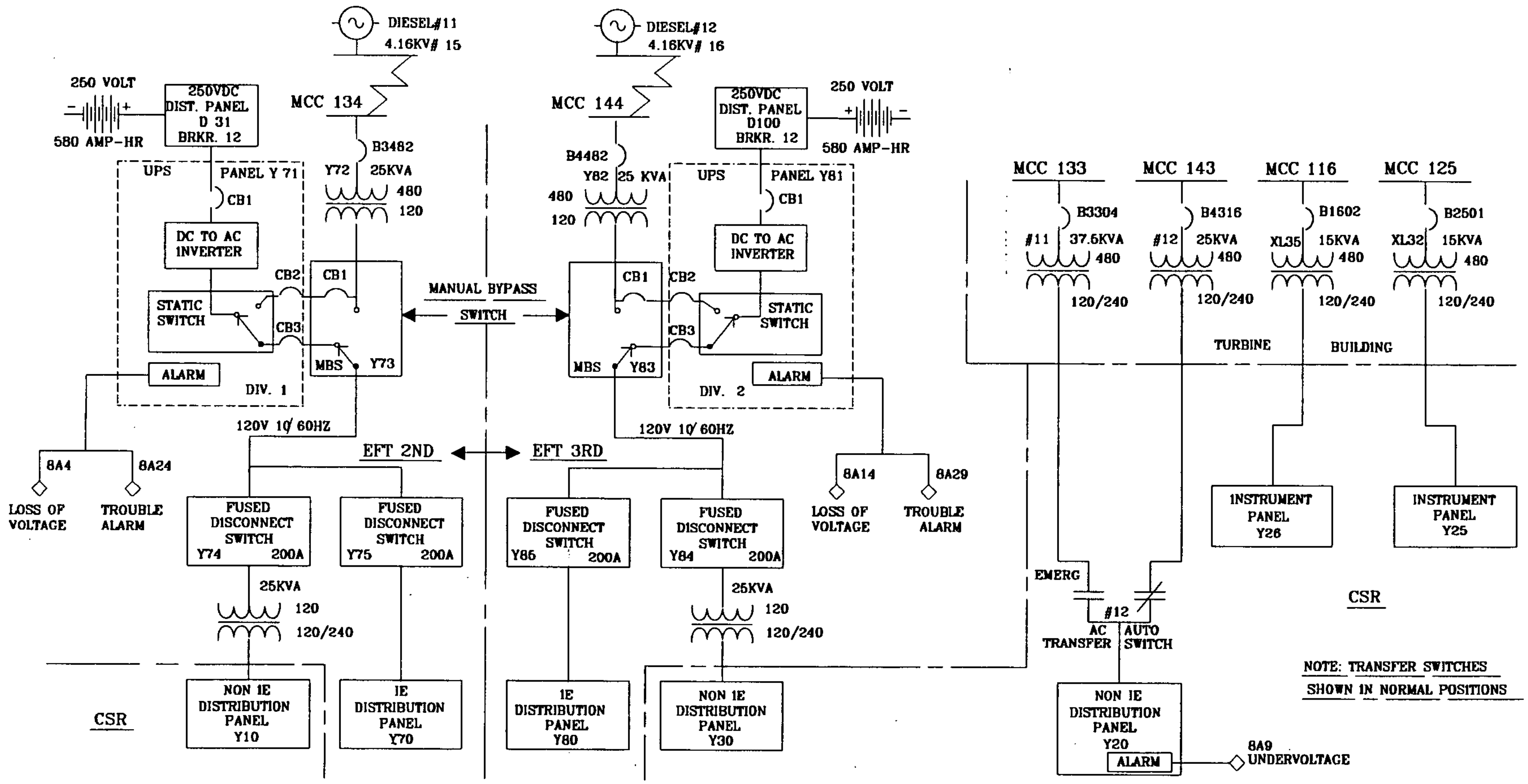
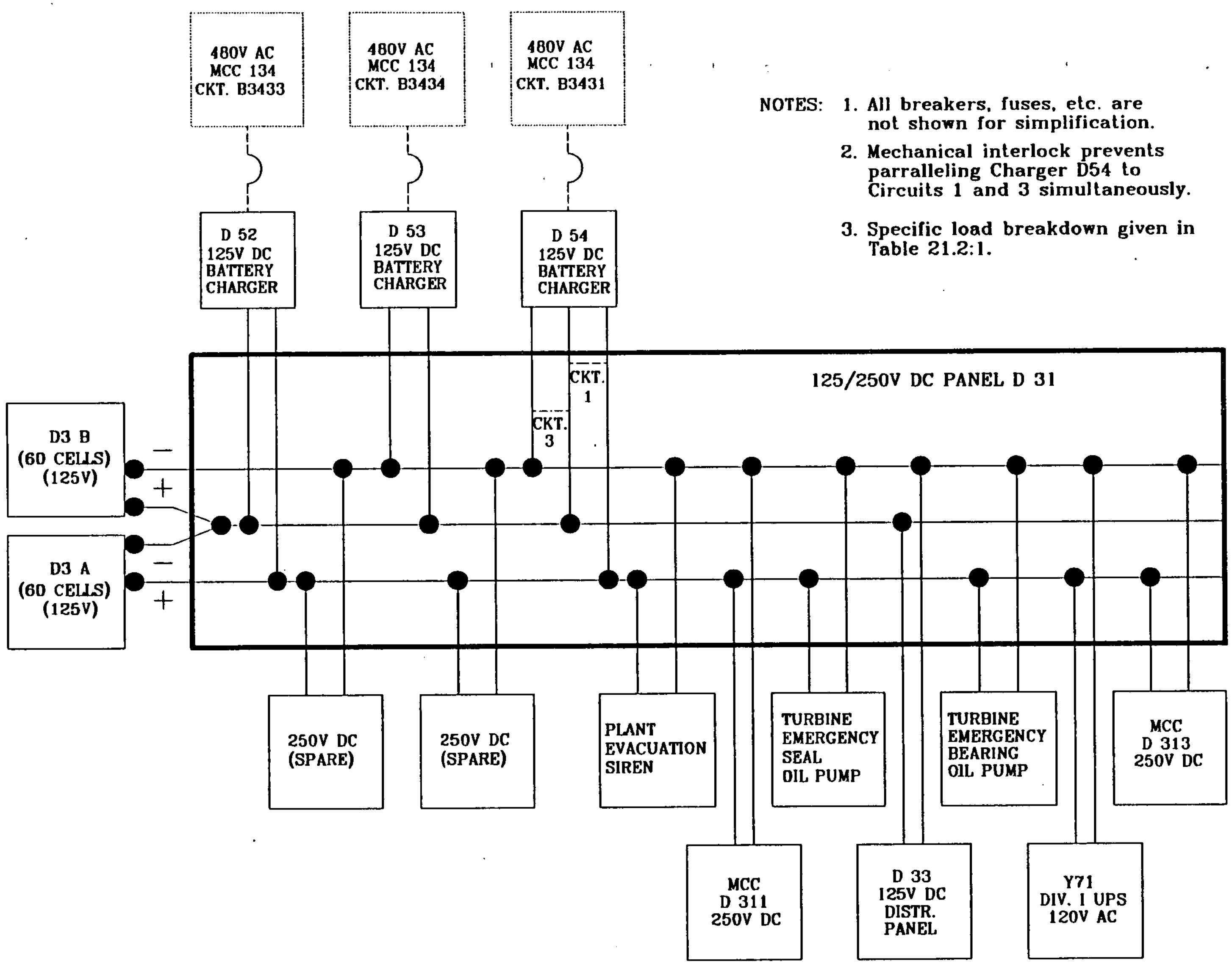


Figure No. 3.2-29
INSTRUMENT AC AND UNINTERRUPTIBLE AC DISTRIBUTION SYSTEM SINGLE LINE DIAGRAM



NOTES:
1. All breakers, fuses, etc. are not shown for simplification.
2. Mechanical interlock prevents parralleling Charger D54 to Circuits 1 and 3 simultaneously.
3. Specific load breakdown given in Table 21.2:1.

Figure No. 3.2-30
250 V DC Distribution - Division I

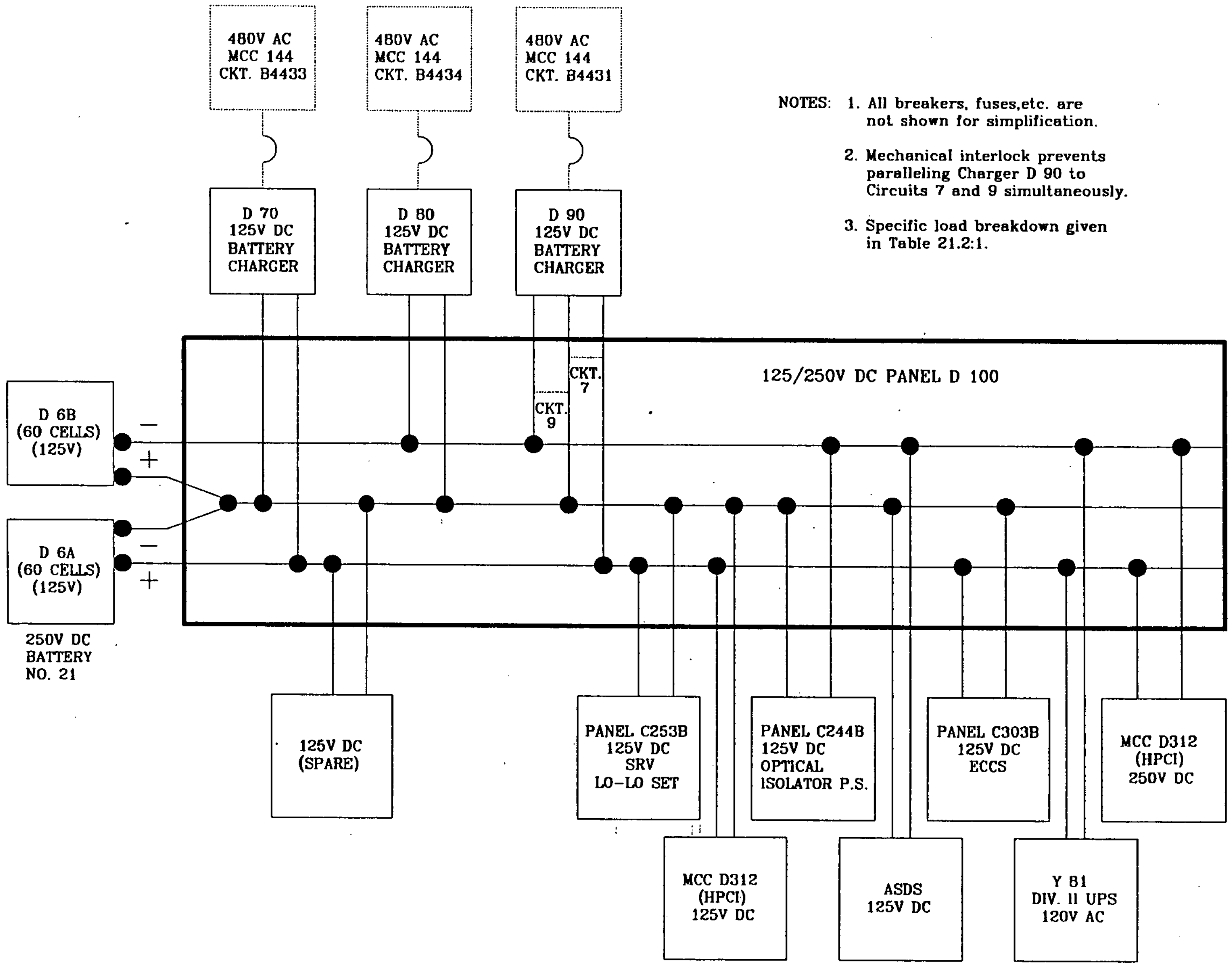
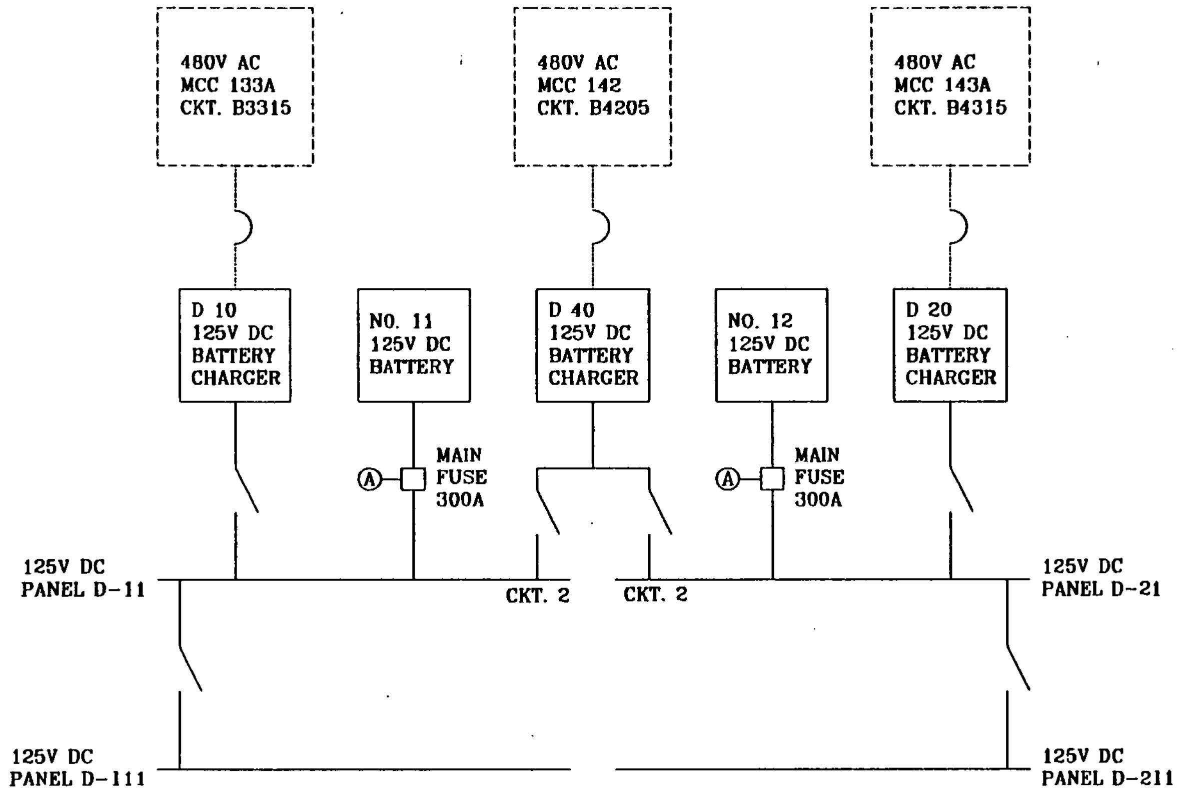


Figure No. 3.2-31
250 V DC Distribution - Division II



NOTES: 1. Circuit 2, Panel D-11 and Circuit 2, Panel D-21 may not be closed simultaneously (by procedure).

2. Specific loads given in Table 21.2:2.

Figure No. 3.2-32
125V DC Distribution

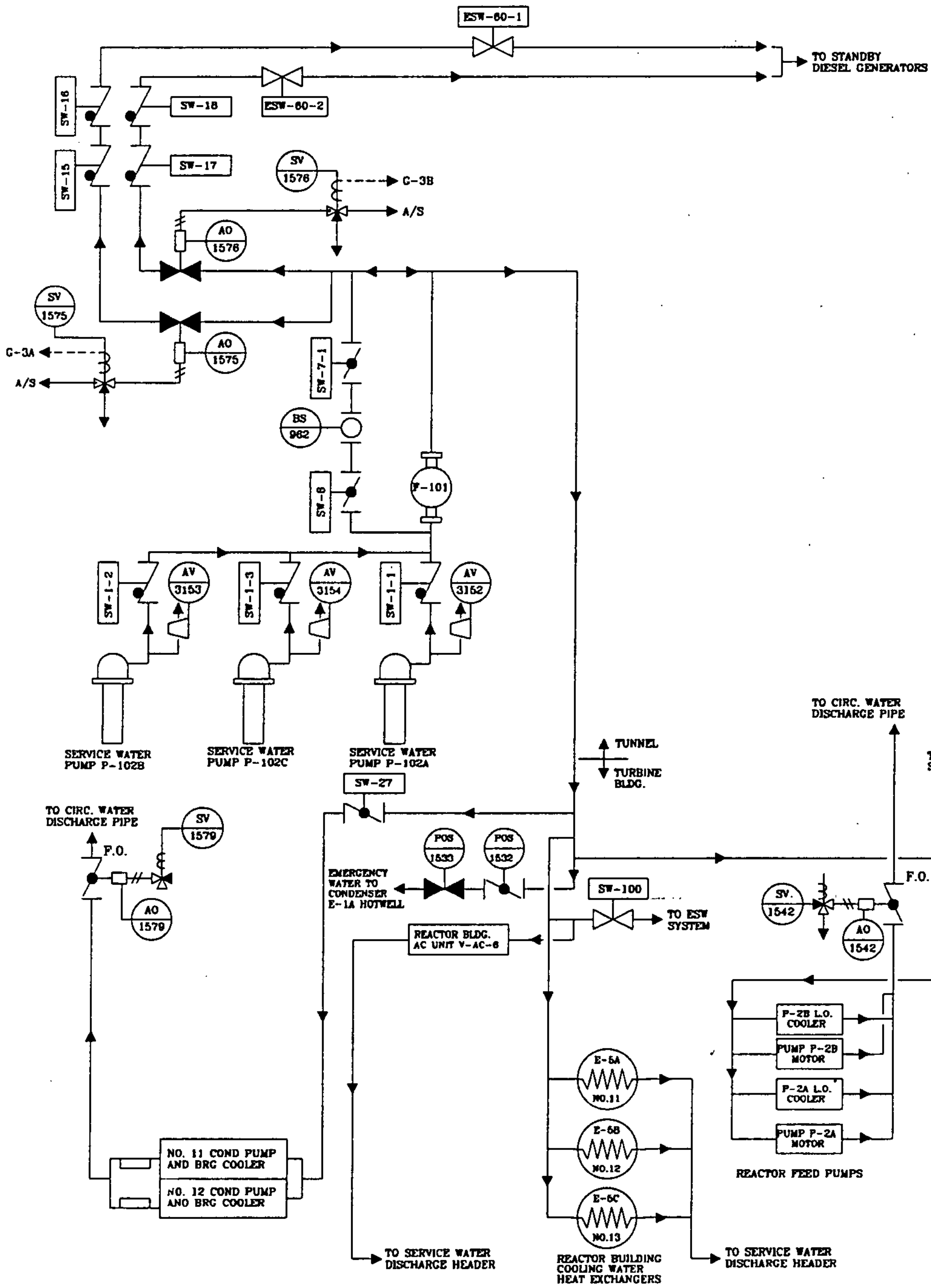


Figure No. 3.2-33
SERVICE WATER SYSTEM

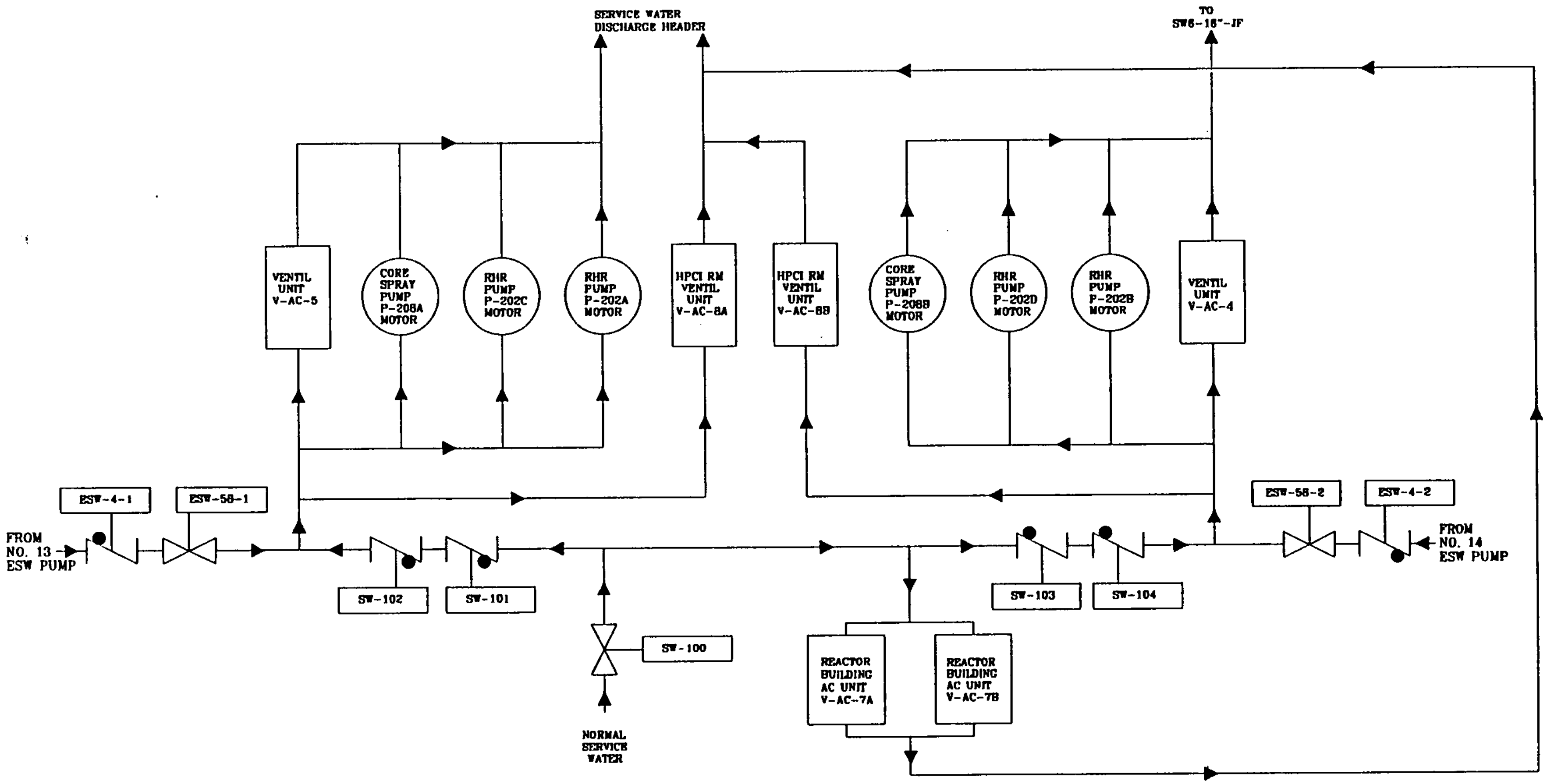


Figure No. 3.2-34
SERVICE WATER SYSTEM

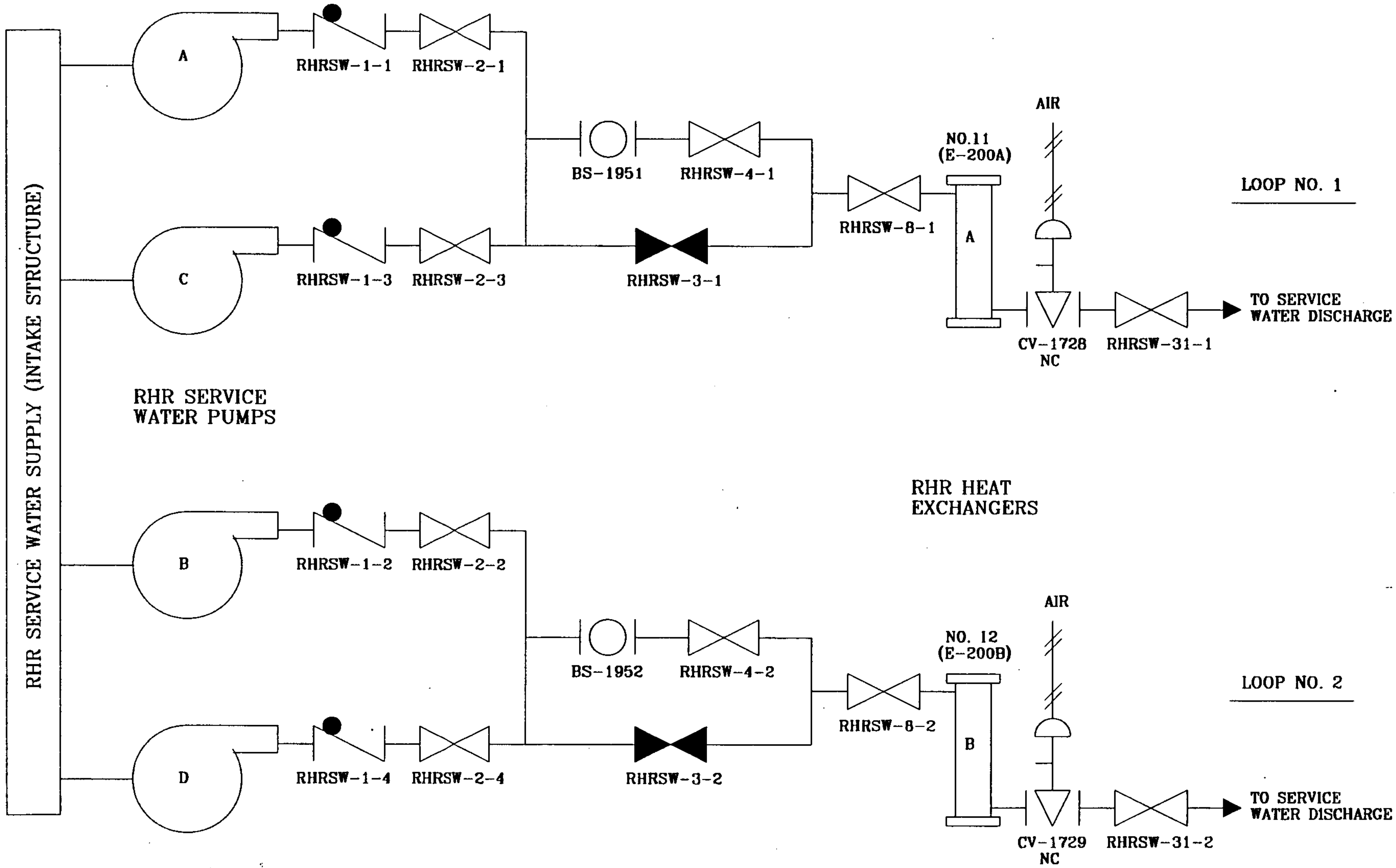


Figure No. 3.2-35
RHR SERVICE WATER SYSTEM

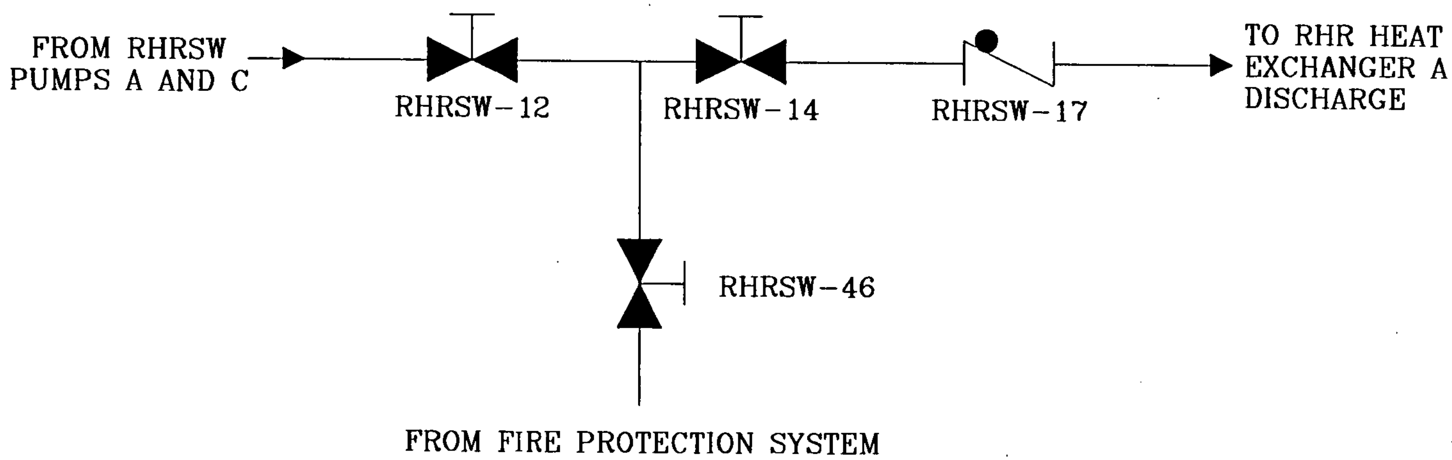


Figure No. 3.2-36
RHRSW/RHR CROSS-TIE FOR
EMERGENCY RPV INJECTION

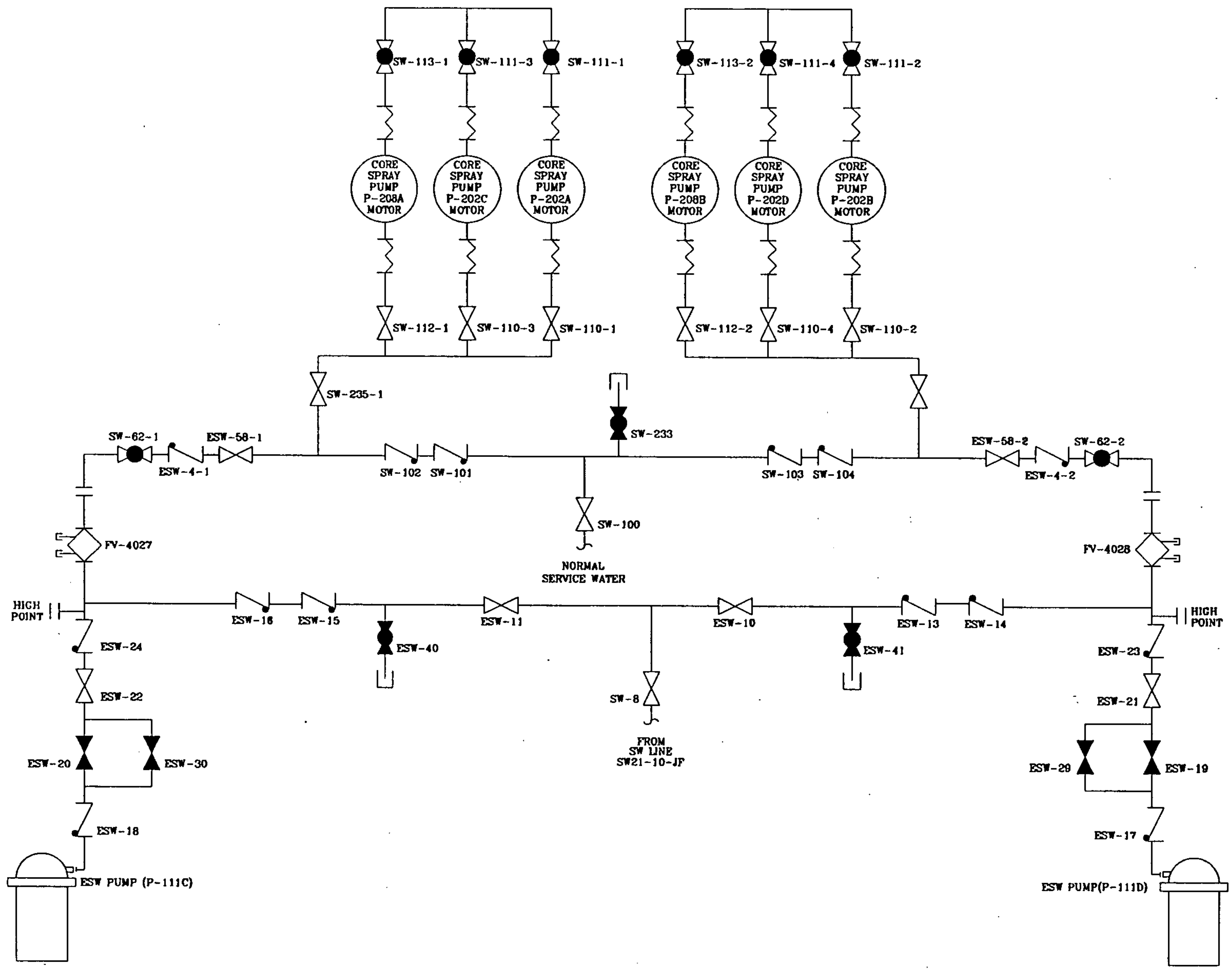


Figure No. 3.2-37
EMERGENCY SERVICE WATER SYSTEM

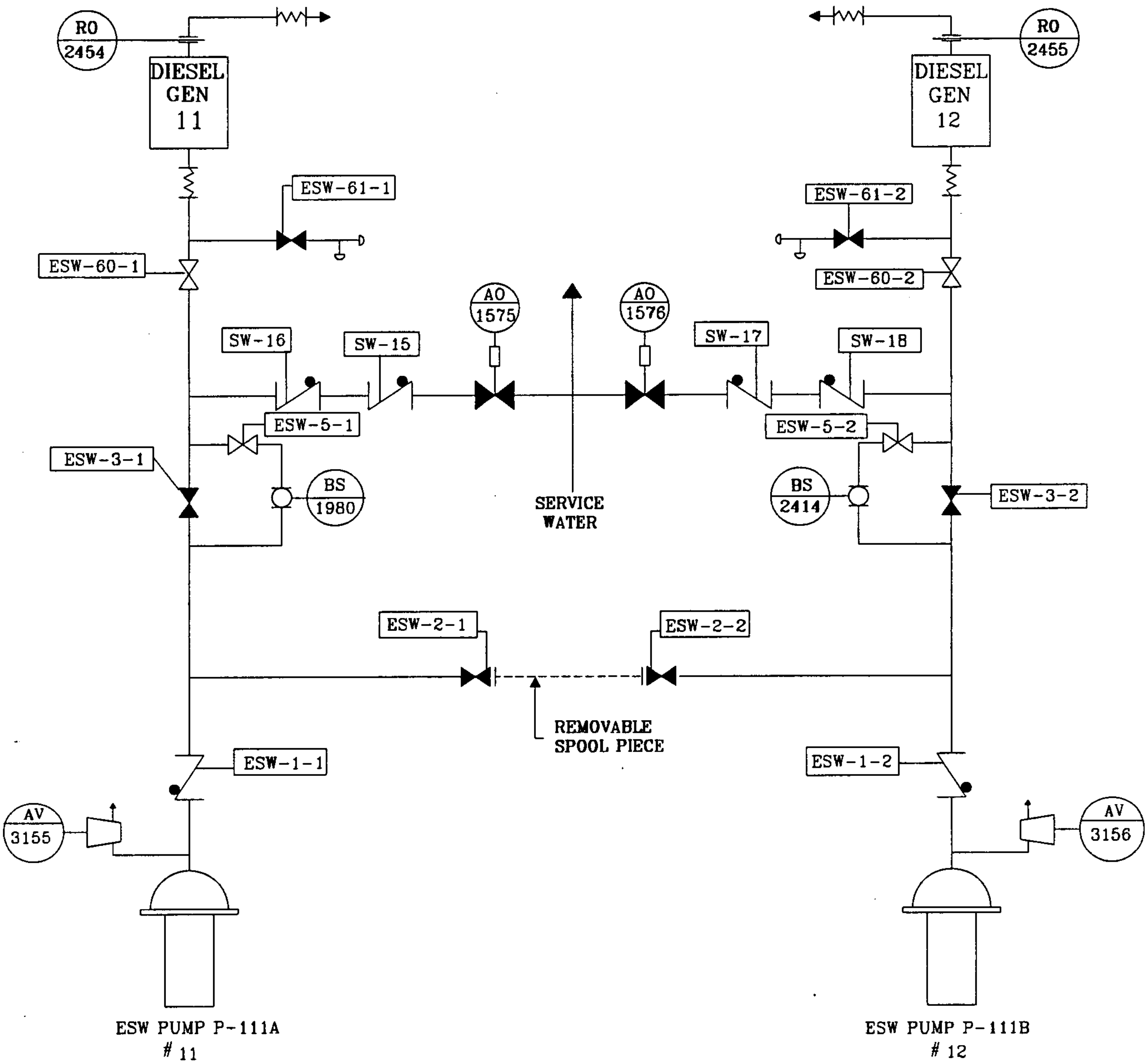


Figure No. 3.2-38
EMERGENCY DIESEL GENERATOR
EMERGENCY SERVICE WATER (EDGESW)

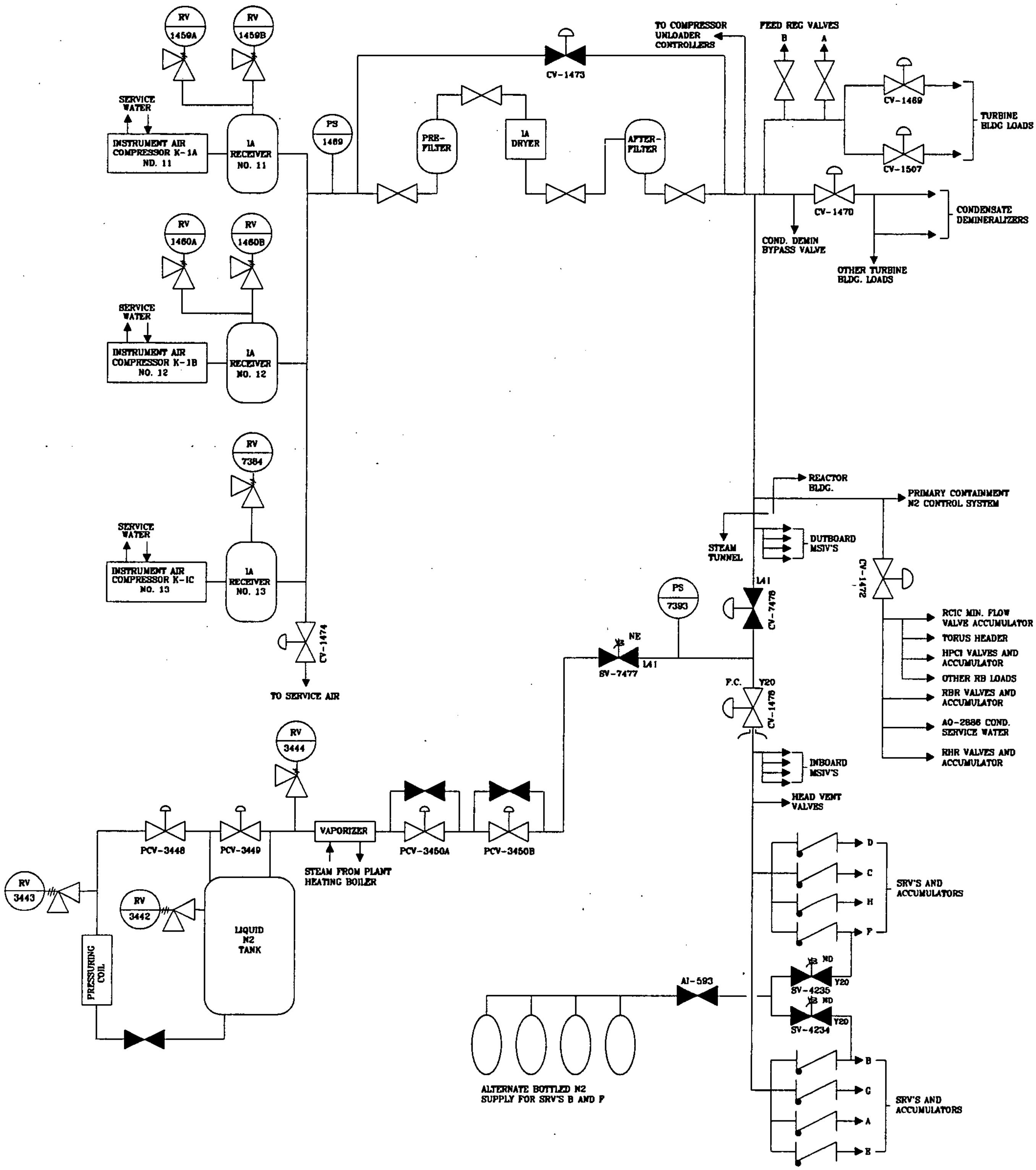


Figure No. 3.2-39
INSTRUMENT AIR AND NITROGEN SYSTEM

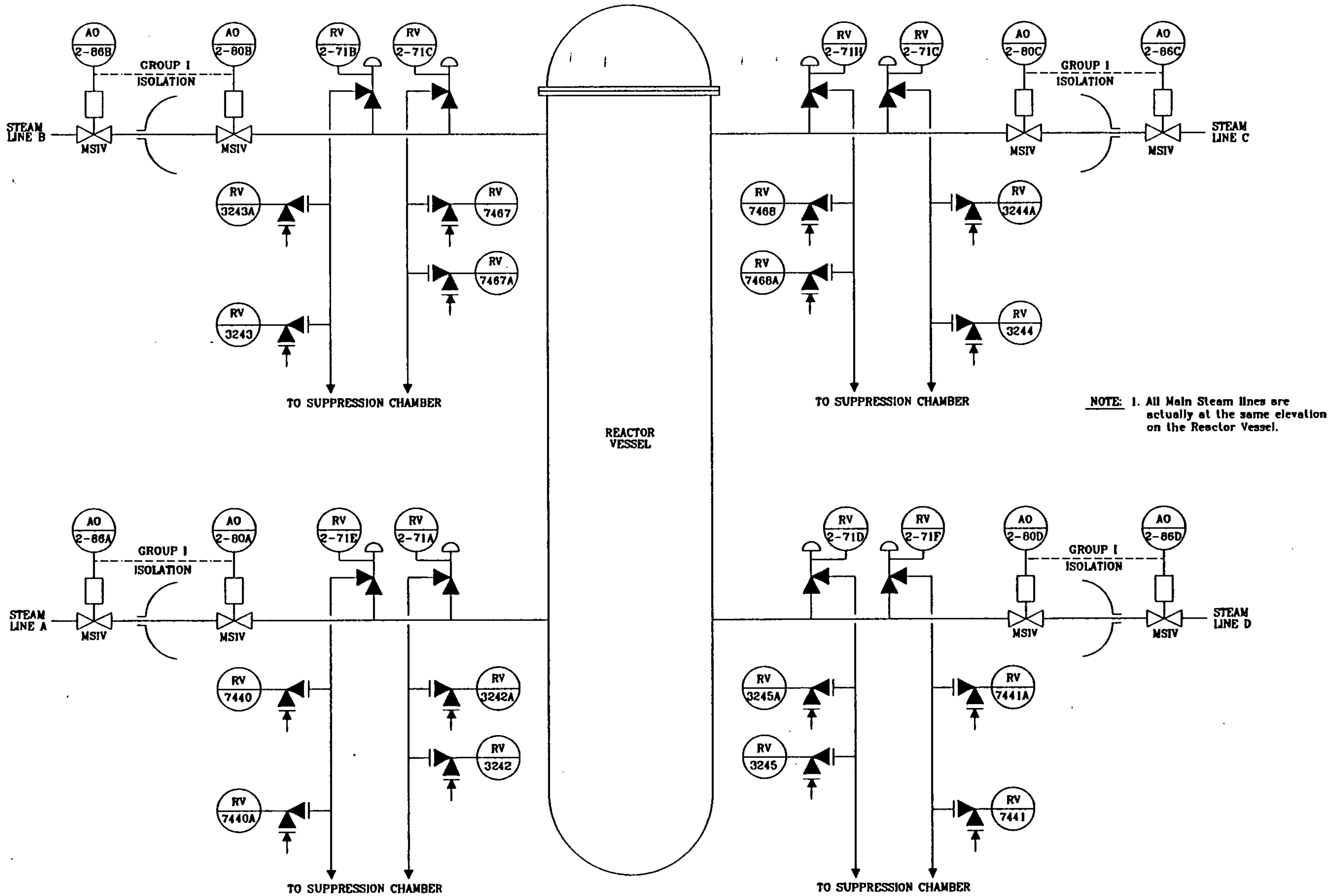


Figure No. 3.2-40
REACTOR PRESSURE RELIEF SYSTEM

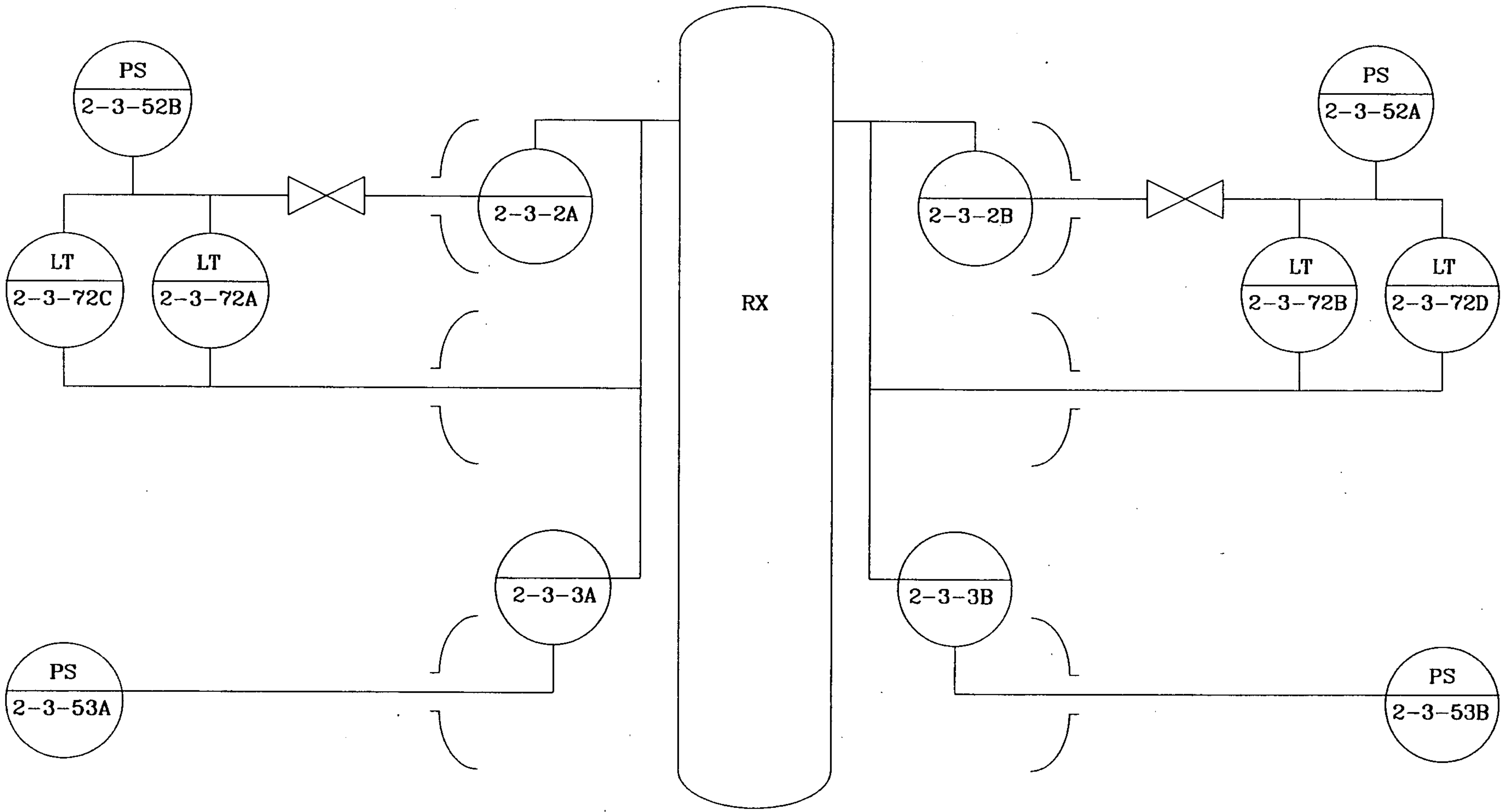


Figure No. 3.2-41
REACTOR LEVEL AND PRESSURE SWITCHES

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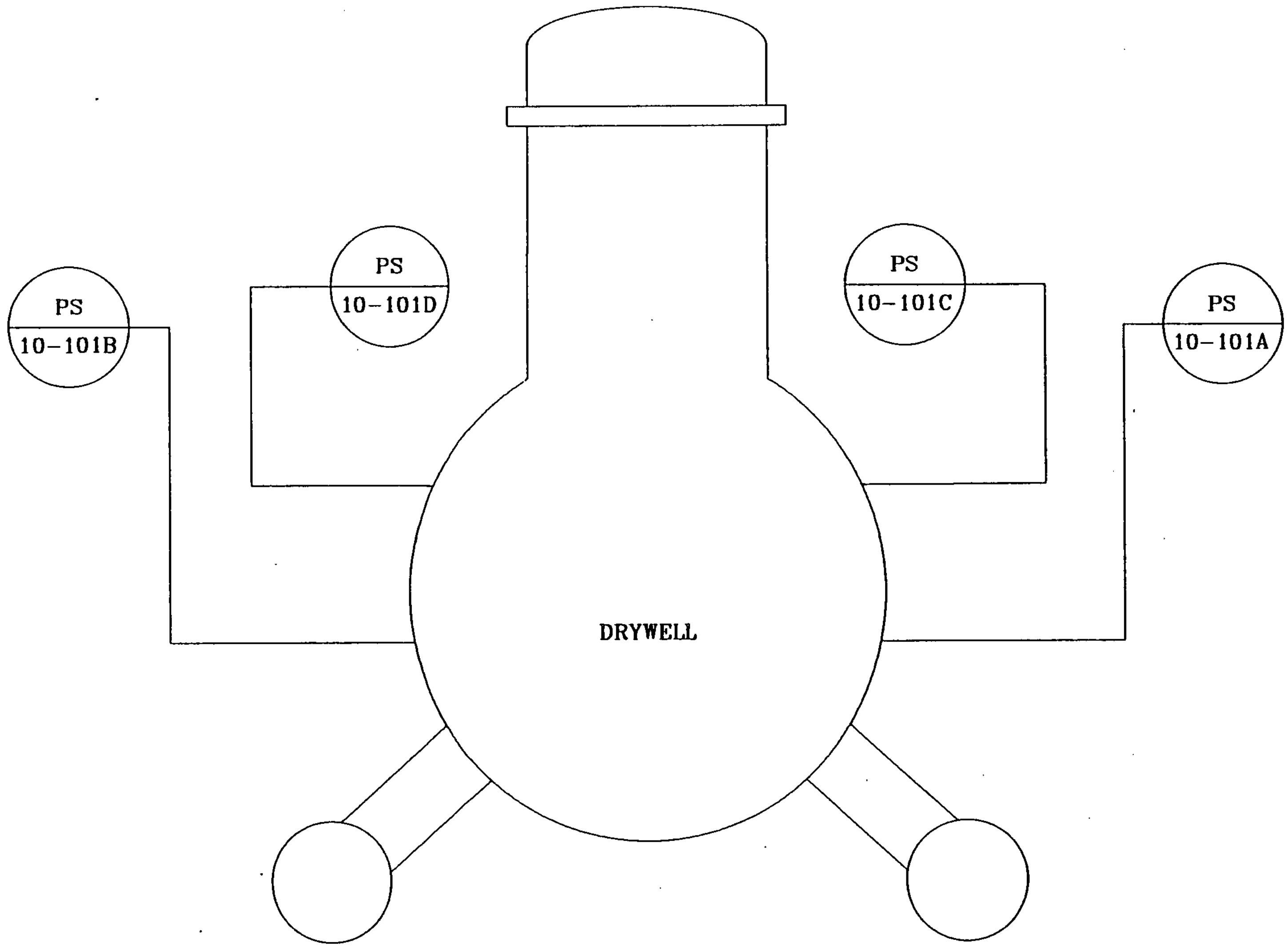


Figure No. 3.2-42
DRYWELL PRESSURE SWITCHES

3.2 System Analysis

3.2.1 System Descriptions

This section provides a brief description of frontline and support systems considered in the IPE.

3.2.1.1 Reactor Protection System (RPS)/Control Rod Drive (CRD) System

Instrumentation associated with the reactor protection system (RPS) monitors key plant parameters to determine whether the plant processes are within the bounds of important parameters associated with normal operation. The system is shown in Figures 3.2-1 through 3.2-4.

The RPS, in conjunction with the primary containment, containment isolation, and ECCS systems, is designed to prevent the release of radioactive materials in excess of the guidelines of 10CFR100 and to prevent fuel damage as a consequence of single operator error or single equipment failure. When specified limits have been exceeded, the RPS initiates a reactor scram.

When an off-normal condition is sensed, the RPS logic sends a trip signal to scram the reactor, shutting down the reactor and annunciating the off-normal condition in the control room. The RPS is arranged as two separately powered trip systems. The trip system has trip logic which produces an automatic reactor scram signal in a 1 out of 2 taken twice configuration.

The RPS trip signal de-energizes the scram pilot solenoid valves in the CRD system which then rapidly vents air from the CRD system scram valves, causing them to open. The opening of the scram valves results in a large differential pressure across the control rod drive piston. This results from the high pressure water on the lower side of the piston and the venting of the top side of the piston to the scram discharge volume. This differential pressure results in rapid insertion of control rods into the core.

3.2.1.2 Alternate Rod Insertion (ARI)

Alternate rod insertion (ARI) is a means of control rod insertion that uses the hydraulic control units and control rod drives as described above, but which is triggered by separate and diverse logic from the RPS. Its purpose is to provide a redundant mechanism for reactor scram in the unlikely event that electrical failure of the RPS or its sensors do not result in rod insertion. ARI initiation signals of high vessel pressure or low-low water level are used to open separate solenoid valves that cause the scram pilot air header to depressurize. Depressurization through the ARI valves takes approximately 25 to 30 seconds after which the reactor is shut down by rod insertion. The ARI system is designed to be a backup system to the electrical portion of the RPS. ARI is assumed not to be effective if the failure to scram is a mechanical problem which prevents control rod insertion. Figures 3.2-5 through 3.2-8 illustrate ARI/RPT logic. Successful actuation of ARI requires that one of two solenoids energize to vent the air supply to the scram valves.

3.2.1.3 Recirculation Pump Trip (RPT)

RPT results in a reduction in reactor flow, increases voids within the core, inserts negative reactivity and thereby limits the reactor power and pressure excursion such as an anticipated transient without scram (ATWS) event. This provides time for scram valve air headers to depressurize while maintaining the primary system pressure within the capacity of the safety relief valves (SRV). Monticello has over 70% full power SRV capability in combination with natural circulation after the RPT trip reduces power to approximately 45% original power if feed pumps continue to run. Tripping the recirculation pumps also provides additional time for operator action to initiate shutdown by means of control rod insertion or SLC. The RPT system is actuated in response to ATWS signals of low-low RPV water level, high RPV pressure or both. Trip of the recirculation pumps following ATWS events limits the pressure excursion and provides negative reactivity insertion by increasing the core void fraction. The primary mechanism for RPT consists of redundant shunt trip devices (trip coils) installed in each recirculation pump motor-generator (MG) set field breaker. These trip coils are normally deenergized, energizing them trips the recirculation pump MG set field breakers causing a trip of the recirculation pumps.

Other signals cause a trip of the MG set drive motor breaker. Low reactor water level is among these signals. This trip would be effective as a backup RPT signal during events involving a loss of feedwater provided the main condenser was also in service. Analysis of this event in the simulator suggested it may be effective in all events, although it was not credited as such.

Successful field breaker or drive motor breaker trip causes a recirculation pump trip. Both pumps tripping would result in natural circulation within the primary system. Given the SRV capacity at Monticello, it is likely that the trip of a single pump is sufficient to limit reactor pressure, assuming the eight SRVs operated.

3.2.1.4 Standby Liquid Control (SLC) System

The purpose of the SLC system is to bring the reactor subcritical by insertion of negative reactivity into the core independent of the CRD, RPS and ARI systems. For ATWS mitigation, SLC is a backup to ARI. Both ARI and SLC mitigate the long term consequences of ATWS. RPT mitigates the initial consequences of ATWS. Figure 3.2-9 is a line diagram of the SLC system.

Successful SLC operation requires one of the two positive displacement pumps to be started and actuation of either of the two explosive squib valves to provide a path to the reactor.

The SLC system is manually initiated from the control room, and injects an enriched boron (sodium pentaborate) solution into the reactor vessel at an effective rate of 86 gpm in accordance with the requirements of 10CFR50.62. At this rate the reactor can be shutdown in less than 15 minutes following SLC actuation. SLC can be initiated any time the operators believe the reactor cannot be shutdown or maintained subcritical with control rods.

Alternately, the sodium pentaborate solution can be injected to the reactor with CRD or RWCU pumps, as directed by procedures referenced in the EOP. The CRD line up is performed by connecting dedicated hoses to the SLC tank and the suction of the CRD pumps through existing connections. This means of SLC injection requires entry to the reactor building and is credited when all steam being produced in the reactor is being directed to the main condenser. No quantitative credit in the IPE models is taken for RWCU as a means of injecting boron to the RPV because credit for only one alternate system is taken in the IPE, and because of the complexity of aligning the system.

3.2.1.5 Feedwater

Following a reactor trip the feedwater system is capable of providing a continuous source of high pressure coolant to the reactor vessel from the main condenser. This is the normal means of ensuring proper coolant inventory control in the RCS during power operation, reactor shutdown and cooldown.

The feedwater portion of the condensate and feedwater systems consists of two one-half capacity motor-driven feedwater pumps, a set of high pressure feedwater heaters, a set of feedwater regulating valves used for normal plant operation, and a start-up feedwater regulating valve (a low-flow valve). The low flow feedwater regulating valve is also used following transient events where the feedwater system is used for coolant inventory control. This system is shown in Figures 3.2-10 and 11.

The Monticello feedwater system is capable of making up to the reactor regardless of the status of the main condenser as a heatsink. The Monticello system incorporates motor-driven pumps in lieu of turbine-driven pumps. Each feedwater pump has a rated capacity of approximately 8,000 gpm @ 1010 psig.

Operation of either of the feed pumps is sufficient to provide makeup to the reactor during events in which decay heat dictates makeup requirements. The feedwater system is expected to trip during ATWS on the level swell subsequent to RPT. On being returned to service during ATWS, feedwater operation is limited to the amount of time it takes to deplete hotwell inventory unless the MSIVs remain open and the main condenser remains in service.

If the main condenser is unavailable, large flow rates to the hotwell are available by makeup from the CSTs or possibly by aligning service water to the condenser.

3.2.1.6 High Pressure Coolant Injection (HPCI)

The HPCI system consists of a steam turbine assembly that drives a constant-flow pump and includes related piping, valves and instrumentation. The system automatically initiates upon sensing high drywell pressure (+2 psig) or low-low reactor water level (-48"). Low-low water level is an indication of a LOCA or an inventory depletion due to decay heat losses, whereas high drywell pressure is indicative to a LOCA within the drywell for the purpose of HPCI initiation. The HPCI turbine is driven by main steam from the reactor vessel and the turbine exhaust is directed to the suppression pool. The HPCI pump is provided with two sources of water for RPV injection, with interlocks to ensure that only one source is aligned at any given time. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. The HPCI system can provide primary coolant makeup at a rate of approximately 3000 gpm. Figures 3.2-12 and 13 are simplified diagrams of the HPCI system.

Operation of the HPCI system at this flow rate is capable of maintaining level above the top of the fuel for transients in which makeup is required due to decay heat, small LOCAs, and ATWS events. During an ATWS event from full power, the resultant RPV water level is expected to be above the top of the fuel but below the reactor low-low level setpoint.

3.2.1.7 Reactor Core Isolation Cooling (RCIC)

The RCIC system consists of a steam turbine assembly that drives a constant-flow pump and includes related piping, valves and instrumentation. The system automatically starts upon sensing low-low reactor water level (-48") utilizing a one-out-of-two taken twice logic. The system delivers design flow within 30 seconds after start. The design flow of the RCIC system is approximately 400 gpm. Since it is relatively low flow, RCIC is considered adequate for transients requiring makeup due to decay heat generation but not for ATWS or LOCA events. The RCIC turbine is driven by steam from the reactor vessel. The turbine exhaust is directed to the suppression pool. The RCIC pump is provided with two sources of water for RPV injection, with interlocks to ensure that only one source is aligned at any given time. The primary source is the condensate storage tank (CST), with the suppression pool providing the secondary source. Figures 3.2-14 and 15 are simplified line diagrams of this system.

3.2.1.8 Control Rod Drive (CRD) Hydraulic

Normal flow through the CRD hydraulic system is from the condenser reject line through the running CRD pump and into the reactor vessel through hydraulic control units. CRD flow is judged to be an important potential coolant makeup source following reactor isolation. One CRD pump is normally operating to maintain pressure in the CRD hydraulic system. Following reactor scram without reset, CRD flow increases to approximately 100 gpm with no operator action, and higher if the alternate bypass injection line is manually aligned. A second CRD pump is available which can provide additional flow if it is manually started from the control room and aligned locally with manual valves and pump flow throttled to prevent low flow suction pressure isolation. In addition, CRD flow may be augmented by other low capacity pumping systems (e.g., standby liquid control) as a means of maintaining core inventory make-up although credit for operation of multiple, low capacity makeup systems are not currently taken in the PRA. See Figure 3.2-16 for a simplified diagram of this system.

CRD makeup was assumed to be adequate only if other high pressure systems had been capable of maintaining inventory early in the event. Accident sequences involving use of CRD for makeup include stuck open safety valves (in which HPCI is assumed to trip following depressurization of the reactor) and loss of decay heat removal sequences (in which HPCI and RCIC are assumed to be inadequate late in the event due to high suppression pool temperatures). As noted in Section 3.1.6, CRD has a significant effect on the time available for operator action to depressurize the reactor and initiate low pressure injection should other high pressure systems be unavailable. Although not credited at this time, CRD may, in fact, be effective in preventing core damage should high pressure systems be lost following a transient.

3.2.1.9 Low Pressure Coolant Injection

The low pressure coolant injection (LPCI) system is an operating mode of the residual heat removal (RHR) system. The primary function of the LPCI mode of RHR system operation is to inject water into the reactor vessel to restore and maintain water level in the event of a large LOCA. In addition, the LPCI system can provide coolant inventory make-up following a small break or other demand for inventory make-up if the reactor vessel is depressurized to less than 325 psig.

The RHR system is divided into two loops. Each loop contains two pumps and one heat exchanger with an associated heat exchanger bypass valve. Each RHR pump has a rated capacity of approximately 4000 gpm. For the LPCI mode, the RHR pumps take suction from the suppression pool and discharge the water into the reactor vessel via the reactor recirculation loop piping. Each pump has a dedicated suction line from the suppression pool and each loop has a dedicated injection path to the reactor vessel through one of the reactor recirculation loops. A normally open cross-tie line which connects both loops is provided downstream of the heat exchangers so that the Loop A pumps can provide flow to the reactor vessel through the Loop B discharge line, and vice-versa. The Monticello LPCI system has "loop selection logic" which is designed to automatically select an undamaged recirculation loop for injection. The purpose of the loop selection logic is to prevent injection flow from bypassing the reactor core if a break occurs in one of the recirculation loops. A simplified drawing of the LPCI mode of the RHR system is included as Figure 3.2-17.

The LPCI system pumps are designed to start automatically upon receipt of any of the following conditions: (1) a low-low reactor vessel water level signal (-47") with low pressure (460 psig), or (2) a high drywell pressure signal, or low-low level (-47") sustained for 20 minutes. When the pressure in the reactor vessel drops below 450 psig, the selected loop injection valves will open, the non selected loop injection valves close, and flow will be delivered to the reactor at 325 psig when pump head exceeds Rp pressure.

3.2.1.10 Core Spray (CS)

The core spray system functions automatically to spray water onto the top of the core at a sufficient rate to cool the core and limit fuel clad temperatures in the event of a large LOCA. In addition, the core spray system can provide coolant inventory makeup following a small break or any other demand for inventory makeup if the reactor vessel is depressurized to less than 350 psig.

Two independent loops are provided as part of the core spray system. Each loop consists of a core spray pump, a sparger ring, spray nozzles, and the necessary piping, valves and instrumentation. Each core spray pump has a rated capacity of approximately 3020 gpm at 130 psig. During core spray system operation, the core spray pumps take suction from the suppression pool and deliver this water to spray spargers above the fuel. The core spray system is depicted in Figure 3.2-18.

The same signals which automatically start the LPCI system also initiate core spray. The core spray system can be manually initiated at any time during operation if the reactor vessel pressure is less than the discharge valve opening permissive of 450 psig.

3.2.1.11 Condensate System

The purpose of the condensate system following transient conditions that require reactor shutdown is to provide a continuous source of water from the condenser hotwells to the suction of the reactor feed pumps for continued coolant inventory makeup to the reactor vessel. Under low reactor vessel pressure conditions (approximately 600 psig), the condensate system can serve as a low pressure injection system provided offsite power is available and the flowpath through the feedwater portion of the system is open.

The condensate portion of the condensate and feedwater system consists of two half capacity motor-driven pumps which take the condensate from the condenser hotwells and pump it through the air ejector condensers, the gland seal steam condenser, the condensate demineralizers, and the low-pressure feedwater heaters to the reactor feedwater pumps. Each condensate pump has a rated capacity of approximately 7150 gpm. A single condensate pump is sufficient for continued operation of the condensate system following reactor shutdown. In the event that the condensate system is required for low pressure injection, the operating condensate pump can discharge through non-operating feedwater pumps to provide coolant inventory to the vessel. Figure 3.2-19 contains a simplified drawing of the condensate system.

The condenser hotwells normally contain 43,000 gallons of water for continued operation of the condensate system as a low pressure source. Makeup from the condensate storage tank is required for long term operation if the condenser is unavailable.

3.2.1.12 Condensate Transfer System

The condensate transfer system utilizes two condensate transfer pumps and one transfer jockey pump to supply water to any connected system that requires water makeup. The system is shown in Figure 3.2-20. Each condensate transfer pump has a rated capacity of approximately 800 gpm. The jockey pump has a capacity of approximately 75 gpm. These pumps draw water from the CSTs and provide water for various station systems. The systems supplied by the condensate transfer system include the main condenser mechanical vacuum pump, core spray and RHR pressurizing systems. The condensate transfer system connection to the core spray and RHR systems can serve as an alternate source of low pressure injection via the RHR and core spray discharge headers. The connection is through a two-inch line and the flow rate to the core spray and RHR systems is limited and has not been analyzed. Hence, sufficient injection from the condensate transfer system may only be possible at times when decay heat removal requirements have decreased to the point where the flow rate through the system would be adequate. Therefore, this system was not credited in the IPE as a viable low pressure injection source because of the low flow capacity. However, the system is used in the PRA as a support system for the mechanical vacuum pump.

3.2.1.13 Power Conversion System (PCS)

The power conversion system consists of the main steam isolation valves (MSIVs), the main steam lines, the turbine bypass valves (TBVs), the main condenser and its auxiliaries, and the condensate and feedwater systems. The PCS has more than adequate capability to remove the decay heat from the reactor thereby preventing steam discharge to the suppression pool. The PCS is the preferred method of decay heat removal because it is the normal means of heat removal during power operation. Operation of the circulating water system to remove heat from the condenser, and the condensate system to support steam seal and air ejector operations, is required for successful heat removal by the main condenser. A simplified drawing of the PCS is shown in Figure 3.2-21.

3.2.1.14 RHR Suppression Pool Cooling Mode

In the suppression pool cooling mode of RHR system operation, the RHR pumps take suction from the suppression pool and discharge the heated suppression pool water through the shell side of the RHR heat exchangers and back into the suppression pool. Flow from the RHR service water system passes through the RHR heat exchanger tubes and cools the suppression pool water.

Use of the suppression pool cooling mode is the preferred method of decay heat removal if the PCS is unavailable. Suppression pool cooling may be initiated at any time following reactor scram since it is independent of the reactor vessel pressure. Operating in the suppression pool cooling mode, only one loop of the RHR system with one RHR pump and one RHRSW pump, is required for sufficient heat removal from the suppression pool for most initiating events. Figure 3.2-22 is a simplified drawing illustrating this mode of RHR operation.

3.2.1.15 RHR Shutdown Cooling Mode

In the shutdown cooling mode of RHR system operation, the RHR pumps take suction directly from the reactor vessel and discharge the heated reactor coolant through the shell side of the RHR heat exchangers and back into the reactor vessel. Flow from the RHR service water system passes through the RHR heat exchanger tubes and cools the reactor coolant. A simple drawing of this mode of RHR operation is included as Figure 3.2-23.

Because of the pressure rating of the suction piping, the shutdown cooling mode is not placed into operation until the reactor vessel pressure is less than 40 psig. Since many of the components associated with shutdown cooling are the same as those required for the suppression pool cooling mode of RHR, it was conservatively assumed during the development of the Monticello IPE that the shutdown cooling system would not be available given failure of the RHR system.

3.2.1.16 Containment Spray System

If the PCS and the RHR suppression pool cooling mode fail to adequately remove decay heat following plant scram or manual shutdown, then the containment pressure will rise. The containment pressure increases during an event such as this would be very gradual. The containment design pressure of 56 psig may be reached about a day into the event. This provides substantial time for recovery of equipment associated with normal means of decay heat removal prior to actuation of backup systems such as containment sprays or venting.

The principal system used for containment spraying at Monticello is the RHR system. Connections are available which enable the operators to use the RHRSW system as well, if necessary. Use of the RHR system for containment spraying is preferred over use of the RHRSW system because no additional water is injected into the containment; RHRSW is river water; and the RHR system may be aligned in the spray mode from the control room. Alignment of the RHRSW system requires manual actions which must be performed outside of the control room.

In the containment spray mode of RHR system operation the RHR pumps take suction from the suppression pool, transport the water through or around the RHR heat exchangers, and discharge the water through one spray header located in the suppression pool airspace, or either (or both) of the two spray headers located in the drywell. One train of RHR can be used for LPCI while the other is used for containment spraying. The Monticello IPE quantification assumes that many of the same components and support systems required for the suppression pool cooling mode of RHR are also required for containment spray. This is similar to the assumptions used in the evaluation of the benefits of the shutdown cooling mode of RHR. Thus, successful operation of the containment spray mode of RHR is strongly dependent upon the success of suppression pool cooling. No quantitative credit for drywell sprays is taken in the level 1 PRA because of this dependence.

The RHR service water system may be used as a back-up to the RHR system for containment spraying. This mode of drywell spraying would most likely be used if the RHR pumps were unavailable. Figure 3.2-24 contains an illustration of the RHR system in the containment spray configuration.

3.2.1.17 Containment Venting

The Monticello containment venting arrangement consists of two existing 18" vent and purge lines from the drywell and wetwell airspace to the standby gas treatment system. This pathway contains duct work and has the potential to release steam to the reactor building, should venting be required. Operating procedures instruct the operator to control this release, maintaining the containment pressure below the primary containment pressure limit as opposed to depressurization of containment. Because of this manner of controlled venting, the effect on the environment in the reactor building is limited, and the operation of core cooling equipment located in the reactor building is assumed to be unaffected by venting activities. Environmental conditions resulting from venting operation were investigated as part of the IPE and are presented in Section 4.1. The Monticello venting arrangement is shown in Figure 3.2-25.

A final option available to the operators for controlling containment pressure if other means of heat removal are unavailable is to vent the containment. Monticello can vent at any pressure up to the containment design pressure of 56 psig. This system is capable of successfully maintaining containment pressure below the design pressure for transients in which the energy addition to the containment is at decay heat rates. The vent may be effective in minimizing the pressure rise in containment during more rapidly evolving events such as ATWS. However, the effectiveness of the vent in preventing core damage was assumed to be limited during these scenarios because of potential environmental concerns in the reactor building.

3.2.1.18 Offsite Power

The offsite power system is the normal power supply to the plant 4KV buses. Power is supplied to four balance of plant buses by one of two reserve transformers. These transformers are supplied by substation ring buses. Each of the two essential buses are supplied by a balance of plant bus. The two essential buses can each be supplied from three sources, respective balance of plant bus; individual breaker from 1AR transformer; and its own emergency diesel generator. There is also the capability to crosstie the two essential buses. The 1AR transformer has multiple supplies from the substation. Plant electrical output is supplied to the grid via the substation. Loss of the 345KV ring bus results in a turbine trip and loss of the 2R supply transformer and transfer to the 1R supply transformer. This transfer is only a one way break before make transfer. If 1R transformer were supplying the plant and it were lost, the 1AR transformer or diesel generators would be required to pick up the essential buses. (Refer to next section for a description of this process.) The various transmission lines, the ring buses, and various transformers have been designed to minimize the probability of a voltage disturbance on one line affecting the power supply to the plant. The IPE simplified the subyard and offsite power alignments and connections. Loss of offsite power is defined as a loss of offsite AC power to all plant buses. The offsite power system is shown in Figure 3.2-26.

Monticello has experienced only one loss of offsite power event in it's history. This event involved a human error when a running pump was racked out on an essential bus. This caused all subyard transformers supplying the plant to trip. A modification was performed to assure a plant bus fault could not cause this again.

3.2.1.19 Onsite AC Power

The Onsite AC Power System is made up of two emergency diesel generators and the plant AC distribution system. The AC distribution system is made up of eight 4KV buses feeding the large motors, and various 480V load centers. This system is shown in Figure 3.2-27 with the exception of the cooling tower supplies that come directly from the switchyard. There are various transfer and load shed schemes associated with these supplies. On a loss of offsite, in this case, loss of 1R and 2R transformer, load shed relays open all bus supplies and strip the 4KV buses of all non-essential loads to allow the 1AR transformer to supply the essential 4KV buses. Loss of voltage or degraded voltage on the essential buses will start the emergency diesel generators and initiate a load shed to allow the diesels to supply their respective bus. ECCS load shedding is another series of relays that act to isolate loads when actuated by reactor low-low level and low pressure or high drywell pressure. The same ECCS signals also start both diesels. The diesel support systems are shown in Figure 3.2-28. The 480 volt supplies to the instrument panels are shown in Figure 3.2-29.

3.2.1.20 DC Power

The DC power system, as modeled for the IPE, consists of two divisions of 250VDC batteries and two divisions of 125VDC batteries. Major loads powered by the 250VDC batteries considered for the IPE were HPCI, RCIC, and uninterruptable power supplies for instrumentation and system controls. Major loads powered by 125VDC batteries considered by the IPE were SRVs, HPCI and RCIC control power, 4KV motor control power, emergency diesel generators, and miscellaneous control power. The impact of loss of one 125VDC bus is greater than the loss of a 250VDC battery and the IPE models loss of a 125VDC bus as an initiating event, as a result. Battery capacity during blackout is 4 hours without load shedding. A test is done each cycle on all 250VDC and 125VDC batteries to verify their capacity. The load profile used for the test discharge is updated as required to reflect plant changes. Loss of a 250VDC battery is assumed to result in loss of the respective HPCI or RCIC system. The systems are shown in Figures 3.2-30, 31, and 32.

3.2.1.21 Service Water

The service water system supplies filtered river water to various loads in the reactor building and turbine building. Key components cooled by service water are the air compressors, feed pumps, mechanical vacuum pump, RBCCW system. It also provides backup cooling water to the emergency service water system and backup to the emergency diesel generator service water system. The assumed success criteria is two service water pumps out of three. The service water system can also be aligned to the condenser as a makeup source. The system is shown in Figures 3.2-33 and 34.

3.2.1.22 RHR Service Water

The RHR service water system is the heat sink for various modes of RHR system operation. One RHRSW pump is required for successful system operation. The system consists of two separate loops with two pumps each. The RHRSW system can also be cross tied to the RHR system by local manual valve manipulation in the turbine building plus manipulation of RHR injection valves to supply an alternate injection source or alternate drywell or wetwell spray source. The system is shown in Figures 3.2-35 and 36.

3.2.1.23 Emergency Service Water

The purpose of the Emergency Service Water (ESW) system is to supply cooling water to the Emergency Filtration Train (EFT) system, ECCS pump motor oil coolers, and ECCS room coolers during a loss of offsite power events coincident with a LOCA. The pumps start automatically. For the IPE, the RHR motor oil coolers were the only components specifically assumed dependent on the ESW system. There are two ESW pumps, each ESW pump supplies water to specific RHR pumps. Even without ESW, the RHR pumps could operate for several hours and could be operated intermittently or staggered to prolong pump operation.

Service water is a backup to the ESW system unless there is a loss of offsite power coincident with an ECCS initiation signal. The IPE assumes service water will not be available as a backup to ESW. The system is shown in Figure 3.2-37.

3.2.1.24 EDG Emergency Service Water

The purpose of the EDG ESW system is to supply cooling water to the diesels. There is one pump required for each diesel to be operational. Service water is a backup to the EDG ESW system unless there is a loss of offsite power coincident with high drywell pressure (+2 psig) or low-low level (-47") and low pressure (460 psig). The IPE assumes service water is not available as a backup. The EDG ESW pump is started automatically when the diesel reaches 125 RPM and the bus is energized. The system is shown in Figure 3.2-38.

3.2.1.25 Instrument Air and Nitrogen

The instrument air system consists of three air compressors powered from essential buses. The system normally operates between 90 to 95 psig. Frontline systems affected by loss of the air system are the main condenser, vent system, condensate and feedwater, and SRV operation for depressurization. Loss of air will result in a scram either from loss of the scram air header, MSIV closure, loss of vacuum, or level control problems. One air compressor is assumed to be adequate to handle the air system loads. The air compressors are load shed on high drywell pressure (+2 psig) or low level (-47") and low pressure (460 psig) if the plant is supplied by the 1AR transformer or the diesels. This would not happen if the plant were supplied by 2R or 1R transformer. Loss of air is significant because of the impact on containment pressure control and high pressure injection. The air system also provides a backup pneumatic supply to drywell pneumatic loads such as SRVs and MSIVs.

The nitrogen system consists of a cryogenic tank external to the reactor building. The nitrogen system maintains the containment inerted as well as supplies pneumatic pressure for SRV manual operation and inboard MSIVs to maintain them open. Normally, loss of the nitrogen system will not have a significant impact on plant operation because the air system is an automatic backup supply. In the event of a loss of all AC power, pneumatic supply to the drywell will be lost. If essential AC power is available, SRVs B and F can be aligned manually with an alternate nitrogen source. The air and nitrogen systems are shown in Figure 3.2-39. Accumulators are assumed to provide for SRV operation on loss of air or nitrogen for a limited time period. The IPE assumes that SRV manual operation will continue for approximately 30 minutes using the SRV air accumulators.

3.2.1.26 Room Cooling

Loss of room cooling is assumed to have limited impact on the capability of the Monticello plant to respond to accident conditions during the time periods of interest for this evaluation. Specific reasons for areas considered are outlined below. Coupling these assessments with a quantitative evaluation of the probability of randomly failing the room cooling systems, and failing to recover these systems, leads to the conclusion that no further evaluation of room cooling is necessary.

a. The RCIC room cooler is not required based on a study done by Bechtel dated 2/23/79. This indicates that the RCIC pump can operate for greater than 72 hours without the RCIC room exceeding 145 degrees.

b. The HPCI room cooler is not required based on a study done by Bechtel dated 5/2/88. This indicates that the HPCI pump can operate for greater than 15 hours without the HPCI room exceeding 135 degrees. This assumes the HPCI room door is opened.

c. Losing steam tunnel cooling will result in significant heatup of this area, the MSIV's will isolate reducing the main heat source in the steam tunnel.

d. It is assumed that room cooling is not mandatory in the turbine building. Operating experience indicates that the condensate pumps, for example, have run for one day without room cooling during operation. Scenarios considered by the IPE will have reduced or no steam flow to the main condenser which will eliminate the major heat source. The condensate pumps would be the most limiting component in the turbine building. The 4KV rooms are open to the surrounding area.

e. Cooling in the CRD pump area is not required because the CRD pump room is large and open to the 935' elevation of the reactor building. A significant heat up of this area should not occur as a result.

f. The RHR room cooling was not considered to be important based on examination of the results of a special test run at Monticello. During this test, operation of the pumps without room cooling for an extended period was performed. The room temperature leveled off towards the end of the test run without room cooling in operation. As the room temperature increased, the heat flux from the room to the surroundings increased until an equilibrium was reached. Three pumps were run during the test. The door was never opened to help cool the room, even though this is a reasonable action to assume during loss of room cooling. A hot RHR heat exchanger, that may exist in some sequences, was not considered. However, even for loss of DHR sequences, makeup requirements to the reactor and flow through RHR or core spray piping will be limited, minimizing the flow of hot suppression pool water to these rooms.

3.2.1.27 Reactor Vessel Pressure Relief System

The reactor vessel pressure relief system consists of eight relief valves, all located on the main steam lines within the drywell. Figure 3.2-40 shows the valve arrangement. The relief valves are self-actuating at 1120 psig but three may also be opened automatically by the automatic depressurization system (ADS) actuation logic or low-low set logic. The relief valves can also be operated individually with remote manual controls in the main control room and from the remote shutdown panel. The discharge of each relief valve is piped into the suppression pool to permit the steam to condense in the pool. T-Quencher devices are used on each SRV discharge. There are no unpiped safety valves which discharge to the drywell in the Monticello pressure relief system.

Automatic ADS occurs upon coincident signals indicating reactor vessel low-low water level, indication that a core spray or LPCI pump is operating, and a 107 second time delay, if not inhibited. The ADS relief valves can also be operated individually by remote manual controls from the main control room.

3.2.1.28 ECCS Initiation Logic

The ECCS initiation logic for RHR, core spray, HPCI, RCIC, ADS, and emergency diesel generators was included in the PRA analysis. Instrumentation supplying the logic is shown in Figures 3.2-41 and 42.

3.2.2 Fault Tree Methodology

Fault trees were used to model plant systems as part of the PRA. They were used to produce system failure equations which were then linked into core damage sequence equations as dictated by the event tree logic. Fault trees developed for the Monticello IPE are listed in Table 3.2-1.

Prior to development of the system fault tree models, various information sources were reviewed and summarized in system notebooks. These were then the basis for the development of the fault trees.

Front-line systems were generally characterized as providing some critical safety functions relating to accident mitigation such as reactor vessel injection or decay heat removal. Support systems provided necessary functions to ensure operability of front-line systems. Separate system fault trees were developed using EPRI's CAFTA fault tree manager which were later linked together using Logic Analyst's PCSETS. The frontline system fault trees were developed to allow the support system fault trees to be linked directly into the logic when quantification was performed.

A prime consideration in developing the fault tree models was the level of detail to include. One criterion for determining the level of detail was the available data concerning components. For example, it was not necessary to model a pump down to the bearings and its control circuit down to the contacts, if all failures of the pump and control circuit were encompassed by one failure mode of interest, such as, pump fails to start, and no further insights would be gained by more detailed modeling.

Data was used to determine what to model based on its relative importance. Faults associated with passive components such as pipes and manual valves were eliminated from further consideration if, for example, the system had a pump with

Table 3.2-1
Monticello IPE Fault Trees

Frontline	Support
Reactivity Control	Offsite Power
Alternate Rod Injection	Onsite AC Power
Recirculation Pump Trip	DC Power
Standby Liquid Control	Service Water
	RHR Service Water
High Pressure Injection	Emergency Service Water
HPCI	EDG Emergency Service Water
RCIC	RBCCW
Feedwater	Instrument Air/Nitrogen
CRD	ECCS Initiation Logic
	Condensate Service Water
Reactor Pressure Control	
SRVs (Depressurization)	
Low Pressure Injection	
Condensate	
RHR (LPCI mode)	
Core Spray	
RHRSW	
Containment Pressure Control	
Main Condenser	
RHR (Suppression Pool Cooling)	
Containment Vent	
Vapor Suppression	

Table 3.2-2
Components/Failure Modes/Transfers Included in the PRA Fault Trees

Component	Failure Mode	Support System Transfer
Pump*, Fan*, Air Compressor*	FTS FTR	Pump/Motor/Engine Cooling AC Bus DC Panel (May be required for breaker operation)
Diesel Generator	FTS FTR	Engine Cooling DC Panel HVAC
Motor Operated Valve*	FTO FTC FTRO** FTRC**	AC Bus DC Panel
Air Operated Valve (Includes Solenoid Valve)	FTO FTC FTRO** FTRC**	AC or DC Panel (for Solenoid Operation) Instrument Air/Nitrogen
Check Valve	FTO FTC FTRC**	
Manual Valve	FTRO** FTRC**	
Filter/Screen/ Basket Strainer/ Heat Exchanger	Plug	
Bus, Batter, Inverter, Charger, Transformer	FTE FTRE	AC Bus DC Panel

Instrumentation and Control components should be modeled based on the criteria in Section 3.2.2.

Notes:

FTS - Fails to Start	FTRO - Fails to Remain Open
FTR - Fails to Run	FTRC - Fails to Remain Closed
FTO - Fails to Open	FTE - Fails to Energize
FTC - Fails to Close	FTRE - Fails to Remain Energized

*Circuit breaker faults were included with these components, i.e., circuit breakers were not be explicitly modeled for these components.

**FTRO and FTRC failure modes would not be included if an associated demand failure existed for the valve, or if the operating status of the valve was identified by a surveillance test on a quarterly basis or more frequently.

Table 3.2-3
INITIATING EVENT TO FRONTLINE SYSTEM DEPENDENCY MATRIX

INITIATING EVENT	REACTIVITY CONTROL				HIGH PRESSURE COOLANT MAKEUP				RPV DEPRESS	LOW PRESSURE COOLANT MAKEUP					CONTAINMENT PRESSURE/TEMPERATURE CONTROL					
	RPS	ARI	RPT	SLC (3)	FEED WATER	HPCI	RCIC	CRD	SRV/ ADS (7)	COND. (4)	LPCI	CS	RHRSW*	COND SW (KEEP-FULL)	MSIV REMAIN OPEN	MAIN COND.	TORUS COOLING	TORUS/ DRYWELL SPRAYS	DRYWELL COOLERS	CONTAIN VENTING
TURBINE TRIP (1)																				
MSIV CLOSURE															X	X(40)				
LOSS OF COND. VACCUM																X(40)				
LOSS OF FEEDWATER					X											P(27)				
MANUAL SHUTDOWN																				
LOSS OF OFFSITE PWR			(50)		X			P(49)		X				X	X(51)	X(51)			X(25)	
STATION BLACKOUT (2)		D(45)	(50)	X	X	D(19)	D(19)	X	X(8,9)	X	X	X	X	X	X(51)	X(41)	X	X	X	
SMALL LOCA								X(21)						X(21)					X(39)	
MEDIUM LOCA						D(35)	X(35)	X(21)						X(21)	X(36)	X(34)			X(39)	
LARGE LOCA					P(44)	X	X	X(21)		P(44)				X(21)	X(36)	X(34)			X(39)	
RPV RUPTURE				X	X(18)	X(18)	X(18)	X(21)		X(18)	X(18)	X(18)	X(18)	X(21)	X(36)	X(34)			X(39)	
LOCA OUTSIDE CONT.					P(17)	X(5)	X(5)	X(21)		P(17)				X(21)	X(36)	(42)			X(39)	
LOSS OF SERVICE WTR			(52)		X(16)	(29)	(29)	D(22)	P(32)(9)		(29)	(29)	P(10)	AIR LOSS?	X(32)	X(43)	(29)	(29)	X(38)	X(32)
LOSS OF RBCCW			(52)					D(22)			(30)	(30)					(30)	(30)	X(38)	
LOSS OF 125VDC BUS	P(13)	P(46)	P(46)		P(15)	X(6)	X(6)	P(20)	P(8)	P(15)	P(12)	P(12)	P(12)				P(12)	P(12)		
LOSS OF ALL 125VDC (2)	P(13)	X(46)	X(46)		P(15)	X	X	P(12)	P(8)	P(15)	P(12)	P(12)	P(12)				P(12)	P(12)		
LOSS OF INST. AIR					P(24)	(31)	(31)	P(23)	P(9)	P(24)	(31)		P(11)	AO-2886?	X(37)	X(28,37)	(31)	(31)		X
LOSS OF NITROGEN									P(9)						P(9)					
RPV LVL REF LEG LEAK	(47)	P(48)	P(48)		P(14)	P(26)	P(26)		P(26)		P(26)	P(26)								
IORV/SORV						D(35)	X(35)	X(21)						X(21)	X(36)	X(34)			X(39)	

X = COMPLETE DEPENDENCE. FRONTLINE SYSTEM NOT AVAILABLE FOLLOWING INITIATOR.

P = PARTIAL DEPENDENCE. FRONTLINE SYSTEM PARTIALLY UNAVAILABLE FOLLOWING INITIATOR (E.G., ONE LOOP OR DIVISION AVAILABLE). FOR LOSS OF SUPPORT SYSTEMS (SERVICE WATER, RBCCW, DC POWER, AC POWER, AIR, NITROGEN) THIS MEANS THAT ONE OF TWO OR MORE REDUNDANT SUPPORT SYSTEMS IS UNAVAILABLE.

D = DELAYED DEPENDENCE. DELAYED IMPACT ON FRONTLINE SYSTEM UNAVAILABILITY (E.G., LOSS OF COMPONENT COOLING).

* = THE NORMAL FUNCTION OF THE RHR SERVICE WATER SYSTEM IS TO REMOVE HEAT FROM THE RHR HEAT EXCHANGERS. THE RHRSW PUMPS MAY ALSO BE USED FOR LOW PRESSURE COOLANT MAKE-UP OR CONTAINMENT SPRAYING IF THEY ARE MANUALLY ALIGNED TO INJECT THROUGH THE RHR SYSTEM FLOW PATHS.

Table 3.2-3 (continued)
INITIATING EVENT TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(1) - EVENTS IN THIS CATEGORY ARE EVENTS THAT CAUSE TURBINE TRIP AND REACTOR SCRAM (AUTOMATIC OR MANUAL) BUT DO NOT CAUSE MSIV CLOSURE, LOSS OF CONDENSER VACUUM, LOSS OF FEEDWATER, OR ANY OTHER SIGNIFICANT IMPACT.

(2) - THIS EVENT IS NOT CONSIDERED AS AN INITIATING EVENT IN THIS STUDY, BUT IS A PLANT STATE THAT IS EVALUATED FOLLOWING LOSS OF OFF-SITE POWER. LOSS OF BOTH 125V DC BUSES IS INCLUDED ON THIS TABLE BECAUSE IT IS EVALUATED FOLLOWING A LOSS OF ONE 125V DC BUS INITIATING EVENT.

(3) - RWCU AND CRD PUMPS MAY BE USED TO INJECT BORON IF SLC PUMPS FAIL. THE RWCU SYSTEM REQUIRES AC POWER, INST. AIR, AND RBCCW FOR OPERATION. THE CRD DEPENDENCIES ARE IDENTIFIED UNDER THE HIGH PRESSURE COOLANT MAKE-UP HEADING ON THIS TABLE.

(4) - THE SERVICE WATER SYSTEM MAY BE MANUALLY ALIGNED TO SUPPLY THE CONDENSER HOTWELL FOR RPV INJECTION VIA THE CONDENSATE SYSTEM.

(5) - HPCI AND RCIC ARE ASSUMED TO BE UNAVAILABLE DUE TO HARSH ENVIRONMENTAL CONDITIONS IF THE BREAK OCCURS IN THE REACTOR BUILDING. A MAIN STEAM BREAK OR FEEDWATER LINE BREAK IN THE STEAM TUNNEL WILL AUTOMATICALLY ISOLATE THE HPCI/RCIC STEAM LINES DUE TO HIGH TEMPERATURE.

(6) - LOSS OF ONE 125V DC BUS WILL DISABLE HPCI OR RCIC, DEPENDING UPON THE BUS THAT IS LOST. LOSS OF BUS A WOULD DISABLE RCIC; LOSS OF BUS B WOULD DISABLE HPCI. FOR THIS STUDY, RCIC IS ASSUMED TO BE UNAVAILABLE AS A CONSERVATIVE ASSUMPTION; HPCI HAS A HIGHER RANDOM FAILURE RATE AND IS THEREFORE MORE LIKELY TO FAIL FOR OTHER REASONS.

(7) - THE REACTOR VESSEL OVERPRESSURE PROTECTION FUNCTION OF THE SRVs DOES NOT DEPEND ON ANY OTHER PLANT SYSTEMS FOR OPERATION.

(8) - THE RPV DEPRESSURIZATION FUNCTION WILL BE AVAILABLE UNLESS BOTH DIVISIONS OF 125V DC POWER AND ONE DIVISION OF 250V DC POWER ARE UNAVAILABLE. (MOST LIKELY DURING AN EXTENDED SBO)

(9) - THE PRIMARY PNEUMATIC SUPPLY FOR THE INBOARD MSIVs AND THE SRVs IS INSTRUMENT NITROGEN. INSTRUMENT AIR IS AUTOMATICALLY ALIGNED TO PROVIDE PNEUMATIC CONTROL IF NITROGEN PRESSURE IS TOO HIGH OR LOW. THE MSIVs AND SRVs ALSO HAVE ACCUMULATORS RATED FOR 30 MINUTES OF VALVE OPERATION. SRVs B & F ALSO HAVE BACK-UP NITROGEN BOTTLES. IF STATION BLACKOUT OCCURS, THE SUPPLIES (AIR AND NITROGEN) TO ALL SRVs AND THE INBOARD MSIVs WILL BE ISOLATED BECAUSE THE AIR SUPPLY SOLENOID VALVES ARE NOT BACKED UP BY DC POWER.

(10) - THE RHRSW PUMPS MAY BE COOLED FROM EITHER OF THE FOLLOWING SOURCES: (1) SERVICE WATER, OR (2) FLOW TAPPED FROM OUTLET OF PUMP AND ROUTED BACK THROUGH PUMP COOLER.

(11) - PRIMARY AIR SUPPLY TO RHRSW FLOW CONTROL VALVES (CV-1728, CV-1729) IS INSTRUMENT AIR. AUXILIARY AIR COMPRESSORS K-10A & K-10B ARE BACK-UP SUPPLIES.

(12) - 125V DC POWER REQUIRED FOR 4160 VOLT BREAKER CLOSURE. BREAKERS MAY BE CLOSED MANUALLY IN SWITCHGEAR ROOMS IF DC POWER IS UNAVAILABLE.

(13) - BACK-UP SCRAM PILOT VALVES REQUIRE DC POWER.

(14) - FEEDWATER PUMPS ARE ASSUMED TO TRIP DUE TO INDICATION OF FALSE HIGH REACTOR VESSEL WATER LEVEL (+48"). (TWO OUT OF TWO ONCE LOGIC).

(15) - IF FEEDWATER OR CONDENSATE PUMPS ARE TRIPPED OFF, DC POWER IS REQUIRED FOR 4160 VOLT BREAKER CLOSURE. BREAKERS CAN BE MANUALLY CLOSED IN SWITCHGEAR ROOMS.

(16) - SERVICE WATER PROVIDES FEEDWATER PUMP COOLING.

(17) - LOCA OUTSIDE CONTAINMENT DISABLES FEEDWATER AND CONDENSATE IF THE BREAK OCCURS IN A FEEDWATER LINE.

(18) - COOLANT MAKE-UP REQUIREMENTS FOR THE RPV RUPTURE EVENT ARE ASSUMED TO BE BEYOND THE CAPABILITY OF THE AVAILABLE MAKE-UP SYSTEMS.

(19) - HPCI, RCIC AFFECTED DURING EXTENDED LOSS OF AC POWER DUE TO LOSS OF BATTERY CHARGERS.

Table 3.2-3 (continued)
INITIATING EVENT TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(20) - ONE CRD PUMP IS NORMALLY RUNNING; ONE IS IN STANDBY. LOSS OF DC POWER CAUSES LOSS OF REMOTE BREAKER CONTROL FOR THE CRD PUMPS.

(21) - CRD AND CONDENSATE SERVICE WATER ("KEEP-FILL") PUMP FLOW CAPACITY INSUFFICIENT FOR LOCAs AND IORV/SORV.

(22) - RBCCW REQUIRED FOR CRD PUMP COOLING. SERVICE WATER PROVIDES RBCCW HEAT EXCHANGER COOLING.

(23) - CRD SYSTEM FLOW CONTROL VALVES FAIL CLOSED ON LOSS OF AIR. HOWEVER, THE TEST BYPASS VALVE CAN BE MANUALLY OPENED TO INJECT TO THE VESSEL, BASED ON PROCEDURAL INSTRUCTIONS.

(24) - ON LOSS OF AIR, THE CONDENSATE DEMINERALIZER VALVES FAIL CLOSED AND THE BYPASS VALVE IS OESIGNED TO OPEN BUT HAS FAILED TO DO SO IN THE PAST. IN ADDITION, THE FEEDWATER REGULATING VALVES FAIL AS IS. THIS RESULTS IN FEEDWATER PUMP TRIP DUE TO HIGH REACTOR LEVEL BECAUSE OF THE HIGH FW INJECTION RATE FOLLOWING SCRAM AND THE FLOW REGULATING VALVES CANNOT BE CLOSED TO THROTTLE DOWN THE FEEDWATER FLOW. THEREFORE, A HIGH LEVEL TRIP WILL RESULT. CONDENSATE PUMPS WILL REMAIN AVAILABLE IF THE CONDENSATE DEMINERALIZER BYPASS VALVE OPENS.

(25) - DRYWELL COOLERS ARE SHED ON LOOP. THEY MAY BE MANUALLY RESTARTED.

(26) - ONE OF TWO DIVISIONS OF AUTO ECCS INITIATION LOGIC (HPCI, RCIC, LPCI, CORE SPRAY, ADS) IS UNAVAILABLE DUE TO FALSE HIGH WATER LEVEL INDICATION. MANUAL OPERATION FROM THE CONTROL ROOM IS POSSIBLE.

(27) - LOSS OF CONDENSATE FLOW CAUSES LOSS OF TURBINE SEAL, AIR EJECTOR, AND H2 RECOMBINER COOLING.

(28) - CIRCULATING WATER PUMP LEVEL CONTROL SENSOR FAILS LOW ON LOSS OF AIR, TRIPPIHG THE CIRC WATER PUMPS.

(29) - LOSS OF SERVICE WATER WILL RESULT IN LOSS OF ROOM COOLING FOR RHR/CS, HPCI, AND RCIC, AND BEARING OIL COOLING FOR THE RHR/CS PUMPS, IF ESW IS ALSO UNAVAILABLE. (THE ONLY SUPPLY TO RCIC IS SW). LOSS OF ROOM COOLING HAS A MINIMAL IMPACT ON THE OPERATION OF THESE SYSTEMS FOR 24 HOURS.

(30) - RBCCW COOLS THE RHR AND CORE SPRAY PUMP SHAFT SEALS. LOSS OF SHAFT SEAL COOLING FOR THESE PUMPS IS ASSUMED TO HAVE NO IMPACT ON PUMP OPERATION DURING THE 24 HOUR MISSION TIME.

(31) - THE RHR, HPCI, AND RCIC MINIMUM FLOW BYPASS VALVES FAIL OPEN ON LOSS OF AIR. THIS RESULTS IH A 10% FLOW DIVERSION TO THE SUPPRESSION POOL. A 10% FLOW DIVERSION IS ASSUMED TO HAVE NO ADVERSE IMPACT ON THE OPERATION OF THESE SYSTEMS.

(32) - LOSS OF SERVICE WATER CAUSES LOSS OF INSTRUMENT AIR COMPRESSOR AND AFTERCOOLER COOLING, WHICH WILL RESULT IN LOSS OF AIR.

(33) - THE DEPENDENCIES SHOWN IN THIS TABLE FOR THE TORUS COOLING MODE OF THE RHR SYSTEM OPERATION ARE ALSO APPLICABLE TO OTHER MODES OF THE RHR SYSTEM NOT LISTED HERE: SHUTDOWN COOLING, ALTERHATE SHUTDOWN COOLING, AND TORUS LEVEL CONTROL. THE SHUTDOWN COOLING AND TORUS LEVEL CONTROL MODES ALSO REQUIRE 250V DC POWER FOR VALVE OPERATION.

(34) - LOSS OF CONOENSER VACUUM WILL RESULT DURING MEDIUM AND LARGE LOCAs AND IORV/SORV DUE TO LOSS OF STEAM FLOW TO THE CONDENSER, EITHER DUE TO MSIV CLOSURE OR DEPRESSURIZATION OF THE PRIMARY SYSTEM. THE LOW REACTOR PRESSURE (<850 #) CLOSURE OF THE MSIVs IS BYPASSED IF THE OPERATOR PLACES THE MODE SWITCH IN SHUTDOWN.

(35) - MEDIUM LOCA AND IORV/SORV ARE ASSUMED TO EVENTUALLY DEPRESSURIZE THE RPV BELOW THE HPCI TURBINE OPERATING LIMIT; THE RCIC FLOW CAPACITY IS INSUFFICIENT FOR THESE EVENTS.

(36) - THE MSIVs ARE ASSUMED TO CLOSE DURING MEDIUM LOCAs, LARGE LOCAs, AND IORV/SORV EVENTS DUE TO LOW STEAM PRESSURE OR LOW-LOW VESSEL LEVEL.

(37) - THE OUTBOARD MSIVs WILL EVENTUALLY CLOSE ON LOSS OF INSTRUMENT AIR WHEN THE ACCUMULATORS BLEED DOWN (AFTER AT LEAST 30 MINUTES).

(38) - THE DRYWELL COOLERS ARE COOLED BY RBCCW, WHICH IS COOLED BY SERVICE WATER.

Table 3.2-3 (continued)
INITIATING EVENT TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(39) - AN ECCS SIGNAL (HIGH DW PRESSURE, OR LOW-LOW VESSEL LEVEL AND LOW PRESSURE) WILL ISOLATE THE DRYWELL COOLERS.

(40) - THE MAIN CONDENSER MAY BE MANUALLY RECOVERED DEPENDING UPON THE CAUSE OF MSIV CLOSURE OR CONDENSER VACUUM LOSS.

(41) - THE MAIN CONDENSER CANNOT MAINTAIN VACUUM BECAUSE THE CIRCULATING WATER PUMPS ARE UNAVAILABLE.

(42) - THE MAIN CONDENSER IS NOT REQUIRED FOR AN UNISOLATED LOCA OUTSIDE CONTAINMENT. IF THE BREAK IS ISOLATED, THE CONDENSER CAN BE RESTORED.

(43) - LOSS OF SERVICE WATER CAUSES LOSS OF AIR WHICH WILL RESULT IN CIRCULATING WATER PUMP TRIP WHEN THE INTAKE STRUCTURE LEVEL SENSOR FAILS LOW. ALSO, AIR IS REQUIRED FOR AIR EJECTOR OPERATION AND SERVICE WATER COOLS THE SEALS OF THE MECHANICAL VACUUM PUMPS.

(44) - LARGE LOCAs ISOLATE THE MAIN CONDENSER. MAKE-UP FLOW FROM THE CSTs MAY NOT BE SUFFICIENT FOR LONG TERM MAKE-UP DURING A LARGE LOCA. NORMAL HOTWELL INVENTORY IS INSUFFICIENT FOR LONG TERM OPERATION.

(45) - ATWS CHANNEL A POWER SUPPLIES: 120 VAC Y70 IS PRIMARY SOURCE, 125 VDC D11 IS BACK-UP.
ATWS CHANNEL B POWER SUPPLIES: 120 VAC Y80 IS PRIMARY SOURCE, 125 VDC D21 IS BACK-UP.

(46) - 125 VDC D11 IS ONLY POWER SOURCE FOR ATWS CHANNEL A RPT TRIP COILS AND ARI SOLENOID VALVES.
125 VDC D21 IS ONLY POWER SOURCE FOR ATWS CHANNEL B RPT TRIP COILS AND ARI SOLENOID VALVES.

(47) - REACTOR WATER LEVEL REFERENCE LEG LEAK WILL CAUSE ONE LEVEL INSTRUMENT TO READ HIGH. THIS IS ASSUMED TO RESULT IN REACTOR TRIP DUE TO TURBINE TRIP AND STOP VALVE CLOSURE ON HIGH LEVEL (+48").

(48) - REFERENCE LEG LEAK WILL RESULT IN HIGH LEVEL INDICATION ON THE ASSOCIATED LEVEL INSTRUMENT. THEREFORE, THE LOW-LOW LEVEL ATWS TRIP PERMISSIVE FOR THIS REFERENCE LEG WILL NOT BE RECEIVED. IN ADDITION, SINCE THE ATWS PRESSURE TRANSMITTERS TAP OFF OF THE REFERENCE LEG, THE 1135 PSIG ATWS TRIP PERMISSIVE MAY NOT BE REACHED DUE TO THE BREAK IN THE REFERENCE LEG. HOWEVER, THE ATWS TRIP INSTRUMENTS ASSOCIATED WITH THE OTHER REFERENCE LEG WILL REMAIN AVAILABLE, AND AN ATWS TRIP MAY BE INITIATED MANUALLY FROM THE CONTROL ROOM IF THE AUTO TRIP LOGIC FAILS.

(49) - THE CRD PUMPS ARE LOAD SHED ON LOSS OF OFF-SITE POWER AND THEY ARE HOT AUTOMATICALLY RELOADED ONTO THE DIESELS.

(50) - THE RECIRC PUMPS WILL TRIP ON LOSS OF OFF-SITE POWER.

(51) - THE MSIVs RECEIVE A SIGNAL TO CLOSE IF OFF-SITE POWER IS LOST (SEE OPERATIONS MANUAL C.4-B.9.2.B).

(52) - RBCCW COOLS THE SEALS OF THE RECIRC PUMP. HIGH SEAL TEMPERATURE WOULD ALARM IN THE CONTROL ROOM. SERVICE WATER COOLS THE MG SETS FOR THE RECIRC PUMPS AND SUPPORTS RBCCW. THE MG SETS TRIP WHEN LUBE OIL TEMPERATURE REACHES 190°F.

Table 3.2-4
FRONTLINE SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

FRONTLINE SYSTEM	REACTIVITY CONTROL				HIGH PRESSURE COOLANT MAKEUP				RPV DEPRESS	LOW PRESSURE COOLANT MAKEUP					CONTAINMENT PRESSURE/TEMPERATURE CONTROL					
	RPS	ARI	RPT	SLC(1)	FEED WATER	HPCI	RCIC	CRD	SRV/ADS	COND. (2)	LPCI	CS	RHRSW*	COND SW (KEEP-FULL)	MSIV REMAIN OPEN	MAIN COND.	TORUS COOLING	TORUS/ DRYWELL SPRAYS	DRYWELL COOLERS	CONTAIN VENTING
RPS																				
ARI			P(17)																	
RPT		P(17)																		
SLC																				
FEEDWATER															P(22)					
HPCI															P(22)					
RCIC															P(22)					
CRD														P(24)	P(22)					
SRV/ADS						P(18)	P(18)			P(8)	P(8)	P(8)	P(8)	P(8)						
CONDENSATE					X(25)			P(23)							X(5,7)	X(5)				
LPCI									P(16)				P(3)	P(20)	P(22)		(9)	(3)		
CORE SPRAY									P(16)					P(20)	P(22)					
RHRSW															P(22)		X(4)	P(3,4)		
COND. SW (KEEP-FULL)											(15)	(15)			P(22)		(15)	(15)		
MSIV OPEN					P(14)					P(14,8)	P(8)	P(8)	P(8)	P(8)		X(7)				
MAIN CONDENSER					P(14)					P(14,8)	P(8)	P(8)	P(8)	P(8)	X(7)					
TORUS COOLING						P,D(10)	P,D(10)				D(9,10)	D(10)	P(11)					(9)		
TORUS/DW SPRAYS						P(12)	P(12)				P(12)	P(12)	P(3)				P(12)			
DRYWELL COOLERS																				
CONTAINMENT VENTING						D(19)	D(19)	D(19)	(21)											

X = COMPLETE DEPENDENCE. FRONTLINE SYSTEM NOT AVAILABLE FOLLOWING INITIATOR.

P = PARTIAL DEPENDENCE. FRONTLINE SYSTEM PARTIALLY UNAVAILABLE FOLLOWING INITIATOR (E.G., ONE LOOP OR DIVISION AVAILABLE).
FOR LOSS OF SUPPORT SYSTEMS (SERVICE WATER, RBCCW, DC POWER, AC POWER, AIR, NITROGEN) THIS MEANS THAT ONE OF TWO OR MORE REDUNDANT SUPPORT SYSTEMS IS UNAVAILABLE.

D = DELAYED DEPENDENCE. DELAYED IMPACT ON FRONTLINE SYSTEM UNAVAILABILITY (E.G., LOSS OF COMPONENT COOLING).

* = THE NORMAL FUNCTION OF THE RHR SERVICE WATER SYSTEM IS TO REMOVE HEAT FROM THE RHR HEAT EXCHANGERS.
THE RHRSW PUMPS MAY ALSO BE USED FOR LOW PRESSURE COOLANT MAKE-UP OR CONTAINMENT SPRAYING IF THEY ARE MANUALLY ALIGNED TO INJECT THROUGH THE RHR SYSTEM FLOW PATHS.

Table 3.2-4 (continued)
FRONTLINE SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(1) - RWCU AND CRD PUMPS MAY BE USED TO INJECT BORON IS SLC PUMPS FAIL.

(2) - THE SERVICE WATER SYSTEM MAY BE MANUALLY ALIGNED TO PROVIDE MAKE-UP TO THE CONDENSER HOTWELL FOR RPV INJECTION VIA THE CONDENSATE SYSTEM.

(3) - THE RHRSW SYSTEM MAY BE MANUALLY ALIGNED FOR REACTOR VESSEL INJECTION OR TO SPRAY THE TORUS OR DRYWELL IF THE RHR SYSTEM IS UNAVAILABLE.

(4) - FAILURE OF RHR SERVICE WATER RESULTS IN FAILURE OF THE TORUS COOLING MODE OF RHR SYSTEM OPERATION. FAILURE OF THE RHRSW SYSTEM MAY RESULT IN A LONG TERM IMPACT ON THE EFFECTIVENESS OF TORUS OR DRYWELL SPRAYING WITH THE RHR PUMPS BECAUSE NO HEAT REMOVAL FROM THE RHR SYSTEM IS OCCURRING. THE RHRSW PUMPS ARE A BACK-UP METHOD OF SPRAYING THE DRYWELL.

(5) - LOSS OF THE CONDENSATE SYSTEM CAUSES LOSS OF STEAM SEAL COOLING, STEAM JET AIR EJECTOR COOLING, AND H2 RECOMBINER COOLING. THIS WILL RESULT IN LOS OF CONDENSER VACUUM.

(6) - DELETED.

(7) - THE CONDENSER IS UNAVAILABLE FOR HEAT REMOVAL IF THE MSIVs ARE CLOSED. LOSS OF CONDENSER VACUUM WILL EVENTUALLY RESULT IN MANUAL MSIV CLOSURE.

(8) - LOW PRESSURE INJECTION SYSTEMS CANNOT OPERATE IF THE REACTOR VESSEL CANNOT BE DEPRESSURIZED USING SRV/ADS, THE MAIN CONDENSER, OR ALTERNATE DEPRESSURIZATION METHODS.

(9) - LPCI, TORUS COOLING, AND DRYWELL/WETWELL SPRAYS ARE OPERATING MODES OF THE SAME SYSTEM, THE RHR SYSTEM.

(10) - LOSS OF TORUS COOLING MAY RESULT IN LONG TERM LOSS OF NPSH FOR PUMPS TAKING SUCTION FROM THE SUPPRESSIOH POOL: E.G., THE LPCI AND CORE SPRAY PUMPS, AND THE HPCI AND RCIC PUMPS IF ALIGNED TO THE POOL.

(11) - LOSS OF TORUS COOLING MAY BE DUE TO LOSS OF RHRSW.

(12) - DRYWELL SPRAYING MAY RESULT IN LOSS OF NPSH FOR PUMPS TAKING SUCTION FROM THE SUPPRESSION POOL: E.G., RHR (LPCI) AND CORE SPRAY PUMPS, AND THE HPCI AND RCIC PUMPS IF THEY ARE ALIGNED TO THE SUPPRESSION POOL.

(13) - DELETED.

(14) - LOSS OF MAIN CONDENSER VACUUM IS ASSUMED TO HAVE NO ADVERSE IMPACT ON THE OPERATION OF THE FEEDWATER AND CONDENSATE SYSTEM PROVIDED HOTWELL MAKE-UP FROM THE CSTs OR THE SERVICE WATER SYSTEM IS MAINTAINED.

(15) - LOSS OF THE CONDENSATE SERVICE WATER SYSTEM IS ASSUMED TO HAVE NO ADVERSE IMPACT ON RHR AND CORE SPRAY SYSTEM OPERATION.

(16) - VERIFICATION OF RHR OR CORE SPRAY PUMP OPERATION PROVIDES A PERMISSIVE FOR AUTOMATIC INITIATION OF ADS.

(17) - ARI AND RPT ARE ACTUATED BY THE SAME INSTRUMENTATION.

(18) - IF A STUCK OPEN SRV FAILS TO CLOSE AT LOW PRESSURE, IT IS ASSUMED TO RESULT IN REACTOR DEPRESSURIZATION TO THE POINT THAT HPCI AND RCIC DO NOT HAVE SUFFICIENT STEAM PRESSURE FOR OPERATION.

(19) - CONTAINMENT FAILURE OR RUPTURE OF THE VENTING DUCTWORK MAY RESULT IN LOSS OF HPCI AND RCIC DUE TO HIGH AREA TEMPERATURE. CRD IS ALSO ASSUMED TO FAIL DUE TO THE HARSH ENVIRONMENT.

(20) - THE CONDENSATE SERVICE WATER SYSTEM INJECTS INTO THE REACTOR VESSEL THROUGH THE RHR/CS INJECTION LINES.

Table 3.2-4 (continued)
FRONTLINE SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

(21) - THE SRV/ADS LOGIC PANELS AND POWER SUPPLIES ARE LOCATED OUTSIDE OF THE REACTOR BUILDING.

(22) - THE MSIVs RECEIVE A SIGNAL TO CLOSE ON LOW-LOW LEVEL, THEREFORE, FAILURE OF THESE SYSTEMS TO OPERATE ON DEMAND MAY IMPLY THAT THE MSIVs GO CLOSED BECAUSE THESE SYSTEMS EITHER AUTO START ON LOW-LOW LEVEL OR MIGHT NOT BE USED UNLESS LOW-LOW LEVEL HAD BEEN REACHED.

(23) - IF THE CONDENSATE PUMPS ARE NOT OPERATING, THE CONDENSATE REJECT LINE SUCTION SOURCE FOR THE CRD PUMPS IS LOST. THE CRD PUMPS MUST THEN B E ALIGNED TO TAKE SUCTION FROM THE CSTs.

(24) - THE CONDENSATE SERVICE WATER SYSTEM CAN INJECT 75 GPM INTO THE REACTOR VESSEL THROUGH THE CRD SYSTEM.

(25) - FAILURE OF THE CONDENSATE SYSTEM WILL FAIL THE FEEDWATER SYSTEM.

Table 3.2-5
SUPPORT SYSTEM TO FRONTLINE SYSTEM DEPENDENCY MATRIX

	REACTIVITY CONTROL				HIGH PRESSURE COOLANT MAKEUP				RPV DEPRESSURIZATION (9)			LOW PRESSURE COOLANT MAKEUP					CONTAINMENT PRESSURE/TEMPERATURE CONTROL					
SUPPORT SYSTEM	RPS A B	ARI A B	RPT A B	SLC (1) A B	FEEDWATER A B	HPCI	RCIC	CRD (1) A B	ADS/SRVs A,C,D	SRVs E,G,H	SRVs B,F	COND. A B	LPCI A B	CS A B	RHRSW* A B	COND SW (1) A B	MSIV REMAIN OPEN	MAIN CONDENS	TORUS (2) COOLING A B	TORUS/DW SPRAYS A B	DRYWELL COOLERS A B	TORUS/DW VENTING
BUS 11 (4160 VAC)					X(41)																	
BUS 12 (4160 VAC)					X(41)																	
BUS 13 (4160 VAC)												X(41)						X(40)				
BUS 14 (4160 VAC)												X(41)						X(40)				
BUS 15 (4160 VAC) E								X(41)					X(41)	X(41)	X(41)				X(41)	X(41)		
BUS 16 (4160 VAC) E								X(41)					X(41)	X(41)	X(41)				X(41)	X(41)		
BUS 17 (4160 VAC)																						
BUS 18 (4160 VAC)																						
MCC 11 (480 VAC)																X(41)						
MCC 31 (480 VAC)																		(51)				
MCC 32 (480 VAC)				X(20)																		
MCC 33A (480 VAC) E				X(41)										X(43)					X(43)	X(43)	X(44)	
MCC 33B (480 VAC) E						(3)							X(43)									
MCC 34 (480 VAC) E																						
L35 (480 VAC)					X(28)							X(28)										
MCC 21 (480 VAC)																		(51)				
MCC 41 (480 VAC)									X(55)	X(55)	P(55)					X(41)						
MCC 42A (480 VAC) E				X(41)															X(43)	X(43)		
MCC 42B (480 VAC)																						
MCC 43A (480 VAC) E						(4)	(39)							X(43)					X(43)	X(43)	X(44)	
MCC 43B (480 VAC) E													X(43)									
MCC 44 (480 VAC) E																						
PANEL Y10 (120 VAC)					P(53)																	
PANEL Y20 (120 VAC)					B(53)				X(54)	X(54)	X(54)				X(46)						P(47)	
PANEL Y30 (120 VAC)					X(29)																	
PANEL Y70 (120 VAC)							X(5)								X(46)		B(15)					X(45)
PANEL Y71 (120 VAC)																						
PANEL Y80 (120 VAC)						X(5)									X(46)		B(15)					X(45)
PANEL Y81 (120 VAC)																						
250 VDC PANEL D31							X(6)															
250 VDC PANEL D100						X(6)				B(10)	B(10)											
125 VDC PANEL D11	B(21)	X(42)	X(42)				X(6)		P(10)		P(10)						B(15)					
125 VDC PANEL D111					P(30)			P(31)				P(30)	X(17)	X(17)	X(17)				X(17)	X(17)		
125 VDC PANEL D21	B(21)	X(42)	X(42)			X(6)			B(10)								B(15)					
125 VDC PANEL D211					P(30)			X(31)				P(30)	X(17)	X(17)	X(17)				X(17)	X(17)		
125 VDC PANEL D33							X(6)			X(10)	B(10)											
SERVICE WATER					X(27)	(7)	(7)	X(26)	B(18)	B(18)	B(18)		(12)	(12)	(13)		X(18)	X(18,33)	(12)	(12)	X(34)	X(18,19)
RBCCW								X(26)					(32)	(32)					(32)	(32)	X(34)	
EDG-ESW (A)																						
EDG-ESW (B)																						
ESW (A)						(7)							(12)	(12)					(12)	(12)		
ESW (B)						(7)							(12)	(12)					(12)	(12)		
RHRSW (A)													D(35)	D(35)					X(36)	D(36)		
RHRSW (B)													D(35)	D(35)					X(36)	D(36)		
CONDENSATE SW													(37)	(37)					(37)	(37)		
INST. NITROGEN									P(11)	P(11)	P(11)						P(16)					
SRV NITROGEN BOTTLES											B(11)											
INSTRUMENT AIR					X(24)	(38)	(38)	P(25)	B(11)	B(11)	B(11)	P(24)	(38)		P(14)	AO-2886	X(16)	X(16,33)	(38)	(38)		X(19)
AUX. AIR COMP. K-10A															B(14)							
AUX. AIR COMP. K-10B															B(14)							
AIR/N2 ACCUMULATORS									B(11)	B(11)	B(11)						B,D(16)					
V-AC-3A (COND PUMP A)					(23)							(23)										
V-AC-3B (COND PUMP B)					(23)							(23)										
V-AC-4 (RHR/CS B)													(48)	(48)					(48)	(48)		
V-AC-5 (RHR/CS A)													(48)	(48)					(48)	(48)		
V-AC-6 (RCIC)							(49)															
V-AC-7A (CRD)								(22)														
V-AC-7B (CRD)								(22)														
V-AC-8A (HPCI)						(8)																
V-AC-8B (HPCI)						(8)																
V-AC-10A (STEAM CHASE)						(52)	(52)										(52)	(52)				
STEAM JET AIR EJECT.																		P(50)				
MECH. VACUUM PUMP																		B(50)				
CIRC. WATER PUMPS																		X				
STEAM SEAL SYSTEM																		X				
BYPASS VALVE CONTROL																		X				
H2 RECOMBINER																		P				

X = COMPLETE DEPENDENCE. FRONTLINE SYSTEM, LOOP, OR DIVISION NOT AVAILABLE FOLLOWING LOSS OF SUPPORT SYSTEM.
P = PRIMARY DEPENDENCE. SUPPORT SYSTEM IS THE PRIMARY SOURCE OF SUPPORT FOR THE FRONTLINE SYSTEM, LOOP, OR DIVISION.
B = BACK-UP DEPENDENCE. SUPPORT SYSTEM IS A BACK-UP SOURCE OF SUPPORT FOR THE FRONTLINE SYSTEM, LOOP, OR DIVISION.
D = DELAYED DEPENDENCE. DELAYED IMPACT ON FRONTLINE SYSTEM, LOOP, OR DIVISION UNAVAILABILITY DUE TO LOSS OF SUPPORT SYSTEM (E.G., LOSS OF ROOM OR PUMP COOLING).

E = DENOTES AN ESSENTIAL BUS THAT IS SUPPLIED BY THE EMERGENCY DIESEL GENERATORS DURING LOSS OF OFF-SITE AC POWER.

* = THE NORMAL FUNCTION OF THE RHR SERVICE WATER SYSTEM IS TO REMOVE HEAT FROM THE RHR HEAT EXCHANGERS.
THE RHRSW PUMPS MAY ALSO BE USED FOR LOW PRESSURE COOLANT MAKE-UP OR CONTAINMENT SPRAYING IF THEY ARE MANUALLY ALIGNED TO INJECT THROUGH THE RHR SYSTEM FLOW PATHS.

3.3 Sequence Quantification

3.3.1 Introduction

The process of quantifying fault trees involved calculating a probability for each basic event on each fault tree. These calculations were based on either historical failure and demand data for Monticello, or on an acceptable source of generic data. Plant specific data were preferred, because they provided a greater potential to gain plant-specific insights than generic data sources.

The plant-specific or generic failure rates fell into one of three categories.

1. Demand-type failures (such as pump failure to start or valve failure to open).

2. Failure during standby (in which case the failure probability was the failure rate times one-half the interval of the test which will detect the failure).

3. Random failure of the component to perform its function during the course of the transient (the failure probability is the failure rate times the mission time of the component, typically 24 hours).

The mission time of components was needed to calculate the probability of failures of operating equipment which occurred randomly subsequent to the initiating event. It is a common practice in the nuclear power industry to use a mission time of 24 hours for PRA activities unless specific considerations dictate otherwise. Successful operation for 24 hours of the equipment required to respond to accident conditions would place the plant in a condition where decay heat levels are very low and long times are available for mobilization of people and equipment for recovery of failures occurring beyond this point in time. Successful 24 hour operation of plant equipment leaves the plant in a state where subsequent failures are expected, and have a very low contribution to the Core Damage Frequency.

3.3.2 List of Generic Data

When plant-specific failure rates could not be calculated, several sources of generic data were available. They were, in order of preference:

1) NUREG/CR-2815
2) GE BWR data
3) NRC LER data for BWRs
4) WASH-1400 data
5) IEEE Standard 500 data

The generic failure rates from these sources were frequently expressed as failures per hour. When a failure rate estimate for a demand failure was given in generic data as failures per hour, the failure rate was converted to failures per demand by treating it as a random failure during standby. The failure rate per hour was multiplied by one-half the test interval to get the probability of failure on demand. This method was considered appropriate because demand failures can be viewed as failures to function on demand due to some preexisting fault which actually developed during the idle period since the last use of the equipment. Table 3.3-1 provides a summary of events which used generic data and the source of that data.

Generic initiating events were used when no plant data was available to estimate an initiating event frequency. The generic initiating events are discussed in more detail in Section 3.1.1. The following generic initiating events were used for the Monticello IPE:

LOCAs
Loss of a DC bus
Reference leg leak

3.3.3 Plant Specific Data and Analysis

Plant specific data was collected primarily for major mechanical components. Classical statistical methods of estimating component failure rates were used.

Plant specific data sources were exhausted before generic data sources were used for any component failure rate. This extensive use of plant specific data means that in general, all major and significant mechanical component failure rates and initiating events were generated using plant specific data.

The plant specific demand failure rate of a given component was estimated by dividing the failures of that component during a specified time interval by the demands for that component to act or operate during the same period of time. Time dependent standby failures were calculated by multiplying the failure rate per hour by one-half the test interval. Time dependent operating failures were derived by dividing the total number of failures by the total number of hours the component had run. This was done for each failure mode of interest to produce rates for several failure modes of each component. The result was expressed either as a rate per demand, if the failure mode is a demand-type failure, or a rate per hour if the failure mode was standby or operating. This process was also used for pools of similar components which were coalesced to obtain statistically broader based results. Examples of component types that were pooled are motor-operated valves, circuit breakers, etc.

The sample period for which plant specific data was collected was nine years, between early 1978 and late 1987. This time span was the period when failure, maintenance, and demand data were readily accessible using electronic data retrieval techniques. The data collected was considered to be significant and provides meaningful failure rates.

The hours of actual plant operation during the sample period were compiled. Random and demand failures were counted during the entire sample period, whether or not the plant was in an outage. These were divided by all the demands during the sample period in order to obtain a failure rate estimate.

Some components and failure modes experienced no failures recorded in the plant data. In order to use the component data to derive a non-zero failure rate estimate in these cases, a value of 0.5 was used to represent the number of "failures" in deriving the estimate. This treatment allowed derivation of a conservative non-zero estimate while giving some credit to the component for not failing. Sometimes, however, this treatment resulted in an estimate that was much higher than generic estimates for the same type of component and failure mode. In such cases, generic estimates were used because using the conservative estimated derived from plant data would penalize the component for having experienced no failures. These cases were indicative of insufficient plant data on which to base an estimate, so generic estimates were considered more appropriate.

Table 3.3-2 provides a summary of components which used plant specific data.

3.3.4 Human Actions

Human actions figure into the PRA due to errors made in restoring systems to normal operating status following test or maintenance activities, activities in progress at the time of the initiating event which influence its outcome such as maintenance or testing, and actions performed in responding to an accident.

3.3.4.1 Restoration Errors

In some cases test or maintenance activities require a component to be temporarily removed from service such that it cannot perform its intended function. Failure to properly restore the component to the proper position or condition could result in the component being unavailable when required. To account for component unavailability due to restoration errors, several factors relating to each test and maintenance action were examined. Many tests and maintenance procedures require an operational test of a system or component following completion of work to verify that the system is operable. In some cases, even if a component were to be left in the wrong position or condition it would automatically return to the proper condition when required. An example of this would be a normally open isolation valve in an injection line which gets a signal to open even if it was inadvertently shut. If these factors were present, restoration failure modes were not included in the models.

Other methods for correcting restoration errors were credited if the operator could tell from the control room that the component was in the wrong position or condition and the operator was required to check the component status routinely (once per hour or once per shift). A reduction in the probability of failing to correctly restore a component to service was made in these cases.

3.3.4.2 Maintenance Activities

There were two general categories of maintenance actions of importance:

1. Routinely scheduled maintenance. These actions occurred periodically and were intended to ensure that a component operates at peak efficiency. Actions such as oil changes, bearing replacement, filter replacement, etc. are examples of this type of maintenance.

2. Unscheduled maintenance. These actions involve repair or replacement of a component due to failure during normal operation or upon detection during periodic testing. Generally speaking, unscheduled maintenance actions require a longer time to complete than scheduled actions. The frequency of both scheduled and unscheduled maintenance can vary significantly from system to system depending on the operating philosophy, e.g., waiting until scheduled outages rather than taking components out of service during normal operations.

Plant specific data was used to derive the fraction of time a given component or train of equipment could be expected to be out of service for maintenance. The period of time over which this data was collected was the same as that for which component data was collected.

3.3.4.3 Testing Activities

Testing actions refer to those periodic operations or inspections of components to verify that they were operable. These acts were usually performed to satisfy requirements contained in the Technical Specifications for the plant. A system could be unavailable because of the test. Information used to derive component unavailability during testing was obtained from a review of plant surveillance procedures.

3.3.4.4 Human Error

With few exceptions, the only human actions credited in the PRA are those currently proceduralized or covered in operator training. A list of human actions considered in the IPE is listed in Table 3.3-3. In deriving human error probabilities for these actions, a screening process was used. In screening these values, human actions were assumed to be influenced by several key performance shaping factors:

1. Time available to perform the action.

2. Degree of difficulty in performing the action (complexity of action or number of procedural steps).

3. Stress under which action must be taken.

The diagnosis phase of a given operator action was based on the most conservative time related diagnosis curve from NUREG-1278. Operator reliability was determined based on plant specific estimates of the time available to initiate and perform the action. Performance shaping factors associated with degree of difficulty and stress were applied to each failure rate.

In addition to the quantification of operator actions, Table 3.3-4 provides a list of potentially important repair and recovery activities that were included in the PRA. Repair and recovery data was either derived from plant specific information (such as for feedwater, instrument air, and the main condenser) or generic sources (such as offsite power recovery or repair of mechanical equipment).

On developing these HEPs, quantification of fault trees and sequences were performed and an identification of the most important operator actions was made to focus detailed HEP development and human factors review. Operator actions were identified as important if they contributed significantly to the baseline core damage probability or if a change in the failure rate could cause a significant increase in overall core damage probability. This method is similar to the approach suggested in NUREG-1335 for identifying important actions that have a significant effect on sequence frequency. The importance measures used to identify these operator actions are Risk Reduction Worth (or Fussel-Vesely) and Risk Achievement Worth (or Birnbaum). Operator actions which contributed to a spectrum of sequences totaling more than 1E-6/yr or could increase the total of a spectrum of sequences by more than 1E-6/yr were considered for further evaluation (Risk Reduction worth of .03 or a Risk Achievement Worth of 1.03).

A summary of the most significant operator actions and recovery actions included in the Monticello IPE is in Table 3.3-5.

The initial detailed human reliability analysis work involved walkdowns of the most significant human actions as identified above. The simulator was used to confirm plant response to specific accident sequence types and compare these to the MAAP analysis. Most of the above work was done without operator involvement. Interviews with operators were completed in the plant. This review included a review of the dominant sequences and key assumptions. These interviews, show the operators have a very good understanding of the initiating events, success criteria and EOPs as used in the IPE. In addition, the function and task analysis for the Monticello Control Room Design Review was reviewed for human factors considerations that would affect the quantification of the I&EPs. No significant human factors concerns were identified for the human actions modeled in the IPE.

A detailed human reliability analysis was performed on the most significant operator actions and recovery actions identified in Table 3.3-5. This table includes the time available for operator recovery actions, the improved error probability from detailed analysis, and the risk achievement work for that action. The detailed HRA was performed using the EPRI SHARP framework for human reliability analysis. The quantification of the human error probabilities (HEPs) was performed using the method in NUREG/CR-4772 (commonly known as the ASEP method), which provides a step-by-step version of the method in NUREG/CR-1278, "The Handbook of Human Reliability Analysis". The ASEP method produces HEP estimates that are somewhat more conservative than would be realized from a full scope application of the THERP method, but less conservative than the screening estimates used in the initial sequence quantification.

The HEP results from the detailed HRA were compared to the results from the screening analysis. In all cases, the detailed HRA produced HEPs at the same value or lower than the screening analysis results. Therefore, none of the screening HEP values used in the accident sequence quantification for the most significant human actions were underestimated. As a result, the accident sequences were not requantified with the new, lower HEPs. Therefore, the current Monticello accident sequences incorporating the important human actions are considered to be bounding.

As part of the HRA, a review of the operator actions in the EOPs was also completed, with the use of a consultant. The only area of concern he brought up was in the ATWS trip area (C.5-1103). There could be some confusion in reading the IRMs. This however is not the only indication available, the operators are trained to also look at the following:

1. APRM and/or SPDS
2. Steam Flow
3. Bypass Valve Position
4. Relief Valve Operation.

No other human factors concerns were brought up in the EOP reviews.

3.3.5 Common Cause

This section outlines the steps for evaluation of common cause failure probabilities in the system models developed for the PRA. The discussion describes how common cause events were included in the fault trees. The PRA common cause failure analysis was part of a wider evaluation aimed at analyzing and estimating the potential effects of dependencies in and among plant systems. Important common cause dependencies were those that may compromise existing redundancy to prevent and mitigate a severe accident.

The common cause failure analysis treated those dependencies that were not explicitly modeled in other phases of the PRA. The list below gives dependencies that are explicitly treated in other phases of the PRA and their method of treatment:

Support System Dependencies: Transfers to support system fault trees were included at appropriate points in system fault trees. Linking of fault trees during fault tree reduction and cutset generation ensured such dependencies were expressed correctly in PRA results.

Shared Components Among Frontline Systems: As with shared support systems, this type of dependency was evaluated correctly by the linking of fault trees in the sequence quantification phase of the analysis.

Human Errors: Human errors, considered in the IPE, were discussed in the previous section. Human errors such as incorrect calibration of sensors or instruments were included as explicit events in system fault trees. Human errors such as failure to restore components to service after their isolation for maintenance were also explicitly included as explicit events in fault trees. Operator errors occurring subsequent to an accident initiator were explicitly treated in the system fault trees if the system was not automatic and required manual action for initiation. Events associated with operator response to initiate automatic systems or repair failed systems were included as recovery actions after sequence cutset results were generated.

Maintenance and Testing: Unavailability of multiple components due to maintenance, repair (unscheduled, corrective maintenance), and testing were included as explicit events in the system fault trees.

External Events: Dependencies among component failures due to the effects of spatially dependent or "external" events (earthquake, fire, external flood, tornado, and heavy wind) are not evaluated as a part of the PRA at this time. The effects of these events will be considered in response to the IPEEE. Common dependencies due to internal flooding were evaluated, however.

Inclusion of other common cause failure modes involved defining additional events representing common cause failures of components to be added to system fault trees. Common cause events were defined and their probabilities estimated to capture the dependence among component failures (both within a system and among separate systems) arising from causes other than those listed above. Some potential causes of dependent component failure other than those listed above included common design, manufacture, installation errors, adverse environment, internal physical similarities such as identical parts, and common human impacts during maintenance, testing, or operation.

The component groups for which common cause events were defined are largely those that have proved important in previous PRAs and reliability studies and are given in Table 3.3-6.

The common cause investigation examined equipment within individual systems and included some components with potential dependencies in multiple systems. Some common cause groups which cross system boundaries include:

1. HPCI and RCIC pumps, turbine steam admission valves, and injection valves

2. LPCI and Core Spray pumps and injection valves

After common cause events were included in the system models, probability estimates were calculated for each event for fault tree quantification and cut set generation. This required selection of a common cause probability model, data analysis to derive parameter estimates for the model, and the evaluation of event probabilities according to the model and the data.

The common cause failure (CCF) analysis estimated CCF probabilities in the framework of the "multiple Greek letter" (MGL) model. This model's parameters (the Greek letters beta, gamma, delta, etc.) are defined as conditional probabilities of the failure of additional components in a common cause group, given the failure of a certain number of components. Thus, for example, the MGL model parameter "beta" is defined as the probability of common cause failure of two or more components in a common cause group, given that at least one has failed; the parameter "gamma" is defined as the CCF probability of three or more components, given the failure of at least two. The basic event probabilities of the common cause events were simply the product of the single-component failure probability estimated from plant data or generic sources, and the MGL parameter estimates.

The primary data for the common cause factor estimates were found in published studies, EPRI NP-3967, NUREG/CR-2770, NUREG/CR-2098, EGG-EA-5623 and NUREG/CR-2099 as well as NPRDs data, which have sorted and classified events as individual or common cause component failures. The multiple greek letter results for the CCF analysis are presented in Table 3.3-7.

3.3.6 Support States

Fault trees were developed for the support systems required by the front line systems. The support system fault trees were prepared and quantified in the same manner as the frontline system fault trees. The effects of support system component failure on frontline systems and sequences was accomplished by linking the support system fault trees directly into the frontline and other support systems they affect. Using the linking process there is no need to produce a support state event tree model to account for the effects of support systems.

3.3.7 Sequence Quantification

After all of the system fault trees were completed, minimal cutset equations for the top events of the fault trees were produced. Equations for the functional headings of the fault trees were then derived where combinations of more than one fault tree top event for a given safety function were required. The functional equations representing the headings for the level 1 Event Trees were then combined with the various initiating events as defined by the event trees to produce core damage sequences.

The computer programs CAFTA and PCSETS were used for this work. Cutsets for all systems, functions and sequences were retained down to the 1E-9 level.

3.3.8 Internal Flooding Analysis

The purpose of the Monticello IPE internal flooding analysis was to determine potential vulnerabilities due to flooding from sources such as tank overfilling, hose and pipe ruptures, and pump seal leaks. The analysis used bounding and conservative assumptions to simplify the analysis. Qualitative and quantitative analyses were performed to identify potentially important vulnerabilities. Attention was focused on the major flood sources in the plant which could affect multiple systems and propagate to other areas. Low capacity systems which had limited impact on other systems and flood initiators which were bounded by other flooding events were qualitatively screened for further consideration.

In performing the internal flooding evaluation, various documents were reviewed which discussed the possibility of internal flood such as the High Energy Line Break Analysis (HELB) and SOER 85-5. SOER 85-5 was a deterministic assessment of flooding which provided significant input to this analysis. The plant response to the SOER concluded there were no vulnerabilities with respect to flooding. The response to the SOER also involved training plant personnel on internal flooding and the need to ensure adequate isolation of equipment. SOER 85-05 identified that maintenance events were the primary cause of flooding events based on industry experience. An extensive part of the plant evaluation of the SOER involved reviewing procedures to see if they adequately addressed flooding and to identify the need for training in this area. The actions taken as a result of the SOER were credited in the performance of this study.

Plant walkdowns were conducted which observed various factors such as the length and diameter of water piping systems, number of valves, tanks, room drains, room sumps, presence of equipment for systems considered in the PRA, propagation to and from other areas, door arrangement, curbs, and more. The primary objective of the walkdown was to determine potential flooding sources and equipment affected. Flood zones were determined. A table was generated showing the flooding sources in the flood zones. A failure modes and effects analysis (FMEA) was performed showing which systems and components would be unavailable if a flood occurred in a specific zone.

Various calculations were prepared which estimated the flooding rate an area could tolerate considering factors such as drains, sump capacity, door leakage, and room volume. Drain and sump capacity for various areas was determined in order to determine the cutoff for systems that need not be considered further in the pipe break analysis because of low capacity. For the flood initiators that remained, calculations were prepared to determine what size piping in those systems needed to be considered further.

The flow rate around closed doors was calculated. Other calculations were made to determine room volumes. These were used to determine what flow rate an area could tolerate and what level the room would reach for a given volume of water. These calculations were used to eliminate low capacity systems from further consideration, either because of low pump capacity or by calculating head loss and determining the flow rate in a given area.

Once the low capacity systems were eliminated, attention could be focused on the higher capacity systems which would affect multiple systems. One other criteria used to eliminate systems from further consideration was they were eliminated if they were in a standby status with no automatic pump start capability.

The equipment affected, the length of piping, and the number of valves or tanks in an area were estimated from plant walkdowns. Generic pipe, valve, and tank rupture frequencies were used to estimate the initiating event frequency due to pipe break. Realizing there was a great deal of uncertainty in the pipe and valve rupture frequencies, a detailed analysis to account for every foot of pipe in the plant was unnecessary because important insights would be apparent regardless of the exact initiating event frequency.

Maintenance events were considered, but no explicit calculations could be generated to add to flood initiating event frequencies derived from pipe, tank and valve ruptures. There have been no significant flooding events due to maintenance since Monticello began operation. An upper bound estimate on the total flood initiating event frequency was derived to evaluate this source of flooding. The total initiating event frequency used in the quantification was distributed among the various areas as a weighted average in proportion to the amount of equipment present in an area which could cause a flood.

On completion of the initiating events analysis and the FMEA, sequence quantification was performed using the internal events event trees and sequence results as a basis. Failures postulated to occur as a result of the flood were related to components represented by basic events within the sequence cutsets. Detailed results from the internal flooding analysis are provided in Section 3.4.5.

Table 3.3-1

Generic Failure Rates Used in this Study

Component	Failure[1]	Failure Rate[2]	Source
Relay	Energize (E) De-energize Remain E Spurious E	1E-6/hr 1E-6/hr 3E-6/hr 3E-6/hr	NUREG/CR-2815 NUREG/CR-2815 NUREG/CR-2815 NUREG/CR-2815
Pressure Transmitter	R	2.68E-6/h	WASH-1400
Level Transmitter	R	2.68E-6/h	WASH-1400
Check Valve	C N	2E-6/hr 2E-7/hr	NUREG/CR-2815 NUREG/CR-2815
Electrical Contacts	C L	1E-6/hr 8.04E-8/h	NUREG/CR-2815 WASH-1400
Strainer	F	3E-5/hr	NUREG/CR-2815
Torque Switch	C	2E-7/hr	NUREG/CR-2815
Level Switch	C F	3.9E-4/hr 2.68E-8/h	GE data WASH-1400
Solenoid Valve	E Remain E	2E-6/hr 1.25E-4/d	NUREG/CR-2815 WASH-1400
Automatic Transfer Switch (modeled as electrical contacts)	C	1E-6/hr	NUREG/CR-2815
Fuse	Premature Blow	3E-6/hr	NUREG/CR-2815
ARI Inverter	R	6E-5/hr	NUREG/CR-2815
Squib Valve	E/N	3E-3/dem	NUREG/CR-4550
Pipe	Rupture(<3") Rupture(>3")	8.6E-9/hr 8.6E-10/h	WASH-1400 WASH-1400
Heat Exchanger	F	5.7E-6/hr	NUREG/CR-4550
ESW Pump	R	1E-4/hr	NUREG/CR-2815
Demineralizer (modeled as filter)	F	3E-5/hr	NUREG/CR-2815
Reference Leg Leak (pipe <3")	Rupture	8.6E-9/hr	WASH-1400
Recombiner (modeled as a filter)	F	3E-5/hr	NUREG/CR-2815
EPR Oil Pump	R S	1E-4/hr 1E-5/hr	NUREG/CR-2815 NUREG/CR-2815
Condensate Service Water Pump	S	1E-5/hr	NUREG/CR-2815

Component	Failure[1]	Failure Rate[2]	Source
Motor Operated Disconnect (modeled as motor S + contacts N)	N	3E-4/dem	WASH-1400
Pressure Switch	C L	2E-7/hr 8.04E-8/h	NUREG/CR-2815 WASH-1400
Level Switch	C F	3.9E-4/h 2.68E-8/h	GE data WASH-1400
Trip Coil (modeled as relay coil)	E	2.68E-4/d	WASH-1400
Manual Switch	C N	1.25E-5/d 1.25E-5/d	WASH-1400 WASH-1400
Filter	F	3E-5/hr	NUREG/CR-2815
Steam Trap (modeled as orifice)	F	6E-7/hr	NUREG/CR-2815
Relief Valve (Except SRVs)	L	6E-6/hr	NUREG/CR-1363
Rupture Disk (modeled as check valve "C" failure)	Premature Rupture	2E-6/hr	NUREG/CR-2815
Louvre (modeled as damper)	N	1E-6/hr	NUREG/CR-2815
Static Switch (modeled as switch contacts)	C	2.68E-7/h	WASH-1400
ARI Power Supply	R	4.2E-6/hr	GE Data
Manual Valve	N	2E-7/hr	NUREG/CR-2815
Tank Heater (modeled as cable open circuit)	R	1E-5/hr	NUREG/CR-2815
Position Indicator (modeled as general instrumentation)	R	2.68E-6/h	WASH-1400
Master Control Circuit	R	3E-6/hr	NUREG/CR-2815
Lube Oil Pump	R S	1E-4/hr 1E-5/hr	NUREG/CR-2815 NUREG/CR-2815
Lighting Panel	R	4.2E-6/h	GE data
Fan	R S	1.8E-6/hr 2.2E-3/d	Palisades PRA Palisades PRA
Auxiliary Oil Pump	R S	1E-4/hr 1E-5/hr	NUREG/CR-2815 NUREG/CR-2815
Exhaust Line Vacuum Breaker (modeled as check valve)	N	2E-7/hr	NUREG/CR-2815
Tank	L	2.7E-7/hr	Seabrook PRA
Orifice	F	6E-7/hr	NUREG/CR-2815

1 Failure codes on this table are:

(Other failure modes are written out.)

C	=	Failure to Close
E	=	Failure to Energize
F	=	Failure to Remain Open/Plugs
L	=	Failure to Remain Closed/Leaks
N	=	Failure to Open
R	=	Failure to Run/Operate
S	=	Failure to Start

2 Hourly failure rates are often given for demand-type failures. These were converted to demand rates by using the conventional practice of multiplying the hourly failure rate by one-half the test interval of the test which demonstrates proper function of the component.

Table 3.3-1

Equipment which used Generic Data

Relay	Pressure switch
Pressure transmitter	Level switch
Level transmitter	Trip coil
Check valves	Manual switch
Contacts	Filter
Strainer	Steam trap
Torque switch	Relief Valve
Level switch	Rupture disk
Solenoid valve	Louvre
Automatic transfer switch	Static switch
Fuse	ARI power supply
ARI invertor	Manual valve
Squib valve	Tank heater
Pipe failure	Position indicator
Heat exchanger	Flow controller
ESW pump	Master control circuit
Demineralizer	Lube oil pump
Reference leg leak	Lighting panel
Circulating water pump bellows	Fan
Recombiner	Auxiliary oil pump
EPR oil pump	Exhaust line vacuum breaker
Condensate service water pump (FTS)	Condensate storage tank
Motor operated disconnect	Orifice
Turbine bypass valve	

Table 3.3-2
Equipment Which Used Plant Specific Data

Offsite power transmission lines	MSIVs
Electrical Bus	Power transformer
Condensate service water pump (FTR)	Circulating water pump (FTR)
RCIC hydraulic valve	RCIC pump
Mechanical vacuum pump	RBCCW pump
Liquid nitrogen tank	Air compressor
Service water pump	Condensate pump
Feedwater pump	EDGESW pump
RHRSW pump	Core Spray and RHR pumps
Maintenance and test unavailabilities	HPCI auxiliary oil pump
Air operated valves	Motor operated valves
RHR pump start permissive limit switch	HPCI pump
HPCI hydraulic valve	HPCI EGM/EGR
Battery bus	Battery charger
Battery ground	Mechanical interlock
Emergency diesel generator	UPS invertor
SRV	Circuit breaker
CRD pump	SBLC pump
Vapor suppression vacuum breaker	EDG breaker

Table 3.3-3

Human Actions Using Screening Process

FUNCTIONAL SEQUENCE	OPERATOR ACTION	FAILURE PROB	DISCUSSION
Accident Class 1A, 3B (Core Damage at High RPV Pressure)	Failure to depressurize the reactor Transients, Small LOCA Medium LOCA	 1E-3 2E-2	Events in which high pressure injection systems are unavailable either due to random failure or the initiating event itself lead to low reactor level and the need to depressurize and enable low pressure injection systems. Manual inhibit of ADS is assumed in accordance with BWR Emergency Operating Procedures thereby making depressurization an operator initiated action. Transients allow 30 min. before the top of the fuel is reached, medium LOCA approximately 10 minutes.
	Alignment of bottled N_2 to SRVs or restoration of MCC41 on loss of offsite power	1E-3	CV 1478 is a fail closed AC powered solenoid valve that supplies nitrogen to the SRVs. On a loss of offsite power to the power supply, the control valve closes. This leaves the SRV accumulators as the pneumatic supply to the SRVs. It is assumed in the PRA that after several hours the accumulators may be depleted unless N_2 or air is restored. More than 5 hours is assumed to be available for this restoration (1 hr for accumulator depletion, 4-6 hrs for RPV heatup and inventory depletion to the top of the fuel).
	Failure to restore feedwater	2.8E-3	Following reactor trip, a shrink in the reactor vessel level occurs followed by an increase in feedwater flow. Without operator action to control feedwater, a level 8 feedwater trip is expected. This operator action represents control of feedwater to prevent feedwater trip or restoration of feedwater following the level 8 trip. The event affects the reliability of feedwater used in quantifying accident sequences.

Table 3.3-3 (Continued)

Human Actions Using Screening Process

FUNCTIONAL SEQUENCE	OPERATOR ACTION	FAILURE PROB	DISCUSSION
Accident Class 1D (Transient with Core Damage at Low Pressure)	Manually initiate low pressure systems	3E-3	On low-low reactor level, core spray and LPCI systems initiate to provide inventory makeup once low pressure is reached in the reactor. Should automatic initiation fail, manual action from the control room is possible.
	Manually align RHRSW to RHR	.75	On loss of high pressure injection (feedwater, HPCI, RCIC) and low pressure injection (condensate, Core Spray, LPCI), RHRSW is capable of being crosstied to LPCI as an injection source. 30 minutes are assumed to be available to perform this alignment.
	Recovery of CRD pumps on loss of offsite power	1E-3	CRD is credited as a makeup system to the reactor for a limited number of sequences (such as stuck open safety valves) in which high pressure systems operated early in the event when decay heat levels were high. During a loss of offsite power, however, a DBA signal occurs on low-low reactor level load shedding the CRD pumps. Operator action is required to establish CRD as an injection source once HPCI and RCIC trip on low reactor pressure.

Table 3.3-3 (Continued)

Human Actions Using Screening Process

FUNCTIONAL SEQUENCE	OPERATOR ACTION	FAILURE PROB	DISCUSSION
Accident Class 1D (continued)	Maximize CRD flow	0.4	On loss of all high pressure injection, CRD can be used as a high pressure injection source. Its capacity (100 gpm SCRAM flow rate) is assumed to be insufficient to make up for decay heat losses early in a transient. A crosstie to the feedwater system is available to maximize flow to more than 200 gpm. Actions to maximize CRD flow are coupled to those requiring reactor depressurization however (both are required on low-low reactor level). A high conditional failure probability with respect to failure to depressurize is assumed as a result.
	Manually crosstie Service Water System to the main Condenser	0.1	Long-term makeup to the condensate system may be performed by hotwell makeup from the service water system. Required only for LOCAs as decay heat demands on CST inventory are not as great for transients.
Accident Class 2 (Loss of Containment Decay Heat Removal)	Failure to align torus cooling	2E-5	The suppression pool cooling made of RHR is a manually initiated system which can be placed in service to remove decay heat independent of reactor conditions. The system is routinely initiated by operators following reactor isolation events. Two to three days are available to activate the system prior to reaching containment failure.
	Failure to align shutdown cooling	1.0	The shutdown cooling mode of RHR is similar to suppression pool cooling except that suction is taken directly from the reactor which must be depressurized at the time the system is placed in service. There is a high degree of coupling assumed with failure to align torus cooling.

Table 3.3-3 (Continued)

Human Actions Using Screening Process

FUNCTIONAL SEQUENCE	OPERATOR ACTION	FAILURE PROB	DISCUSSION
Accident Class 2 (continued)	Reopening MSIVs	.083	Reestablishing the main condenser as a heat sink is a viable way to prevent containment heatup on decay heat. This action is principally credited when random failures of RHR equipment result in failure of DHR. This recovery applies only to MSIV closure initiator. 2 to 3 days are available to initiate this action and prevent containment failure. Failure rate based on prompt recovery.
	Failure to vent the containment	1E-3	On loss of all other means of decay heat removal (main condenser and all modes of RHR), operator action to vent the containment prevents overpressure failure of the containment into the reactor building. Symptoms associated with this action are diverse from those requiring RHR (containment pressure vs. suppression pool temperature). As with other means of DHR, more than 48 hours are available before containment failure.
Accident Class 4 (ATWS)	Failure to manually SCRAM the reactor	1E-3	Included in ARI fault tree as an alternate means of activating ARI solenoids. Routinely practiced by operations during shutdowns and in simulator. Assumed to be important only for ATWS events that do not lead to high reactor pressure or low reactor level (i.e., TT with bypass).

Table 3.3-3 (Continued)

Human Actions Using Screening Process

FUNCTIONAL SEQUENCE	OPERATOR ACTION	FAILURE PROB	DISCUSSION
Accident Class 4 (continued)	Failure to initiate SLC No main condenser With main condenser	 4E-2 5E-3	On failure of the reactor protection system and ARI to insert rods, a diverse means of reactor shutdown is available in SLC. The system is manually initiated from the control room. The operator is required to initiate the system early in an ATWS event. On suppression pool temperature, time available to initiate the system and prevent containment failure is transient dependent but is generally on the order of 20 minutes or more.
	Failure to initiate Alternative Boron Injection	.1	If mechanical or electrical failures within SLC occur, alternate means of boron addition to the reactor are provided by CRD pumps or the reactor cleanup system. These actions take place in the reactor building and are credited only when the main condenser is available condensing most of the steam from the reactor thereby limiting the rate of containment heatup.
	Failure to control reactor level after SLC injection	1E-1	On successful shutdown with SLC, it is important not to dilute the boron by overflowing the reactor to the suppression pool with injection systems. Operator action to maintain level in the reactor near normal precludes this event.
Internal Flooding	Recovery of Service Water after plugged strainers	2.5E-2	This operator action represents back washing service water strainers should they develop a high differential pressure. Actions take place in the intake structure using installed hardware requiring only a few minutes to perform.

Table 3.3-4

Repair and Recovery Actions

OPERATOR ACTION	FAILURE PROB	DISCUSSION
Recovery of feedwater during loss of feedwater	0.11	Based on plant specific data regarding feedwater trips. (8/9 times feedwater has been recovered prior to reaching low-low reactor level.) Used to reduce the loss of feedwater initiator frequency.
Recovery of loss of air	0.5	Based on plant specific experience associated with failure of the relief valves on the air receiver tanks. The one time this has occurred at Monticello, operators were able to isolate these valves prior to depressurization of the air system. Used to reduce the loss of air transient initiator frequency.
Recovery of the main condenser	0.33	Based on plant specific experience with loss of main condenser events. This factor was used in considering reestablishing the main condenser as a heat sink during loss of decay heat removal sequences. Actions involve placing the main condenser in service as opposed to repairing failed equipment.
Recovery of decay heat removal systems	0.15	Applied to sequences involving loss of main condenser, all four modes of RHR (except for failure to initiate the system), and containment venting. Loss of DHR is a slowly evolving sequence requiring two to three days prior to reaching containment failure, providing significant time for repair and recovery. Repair of only one of the DHR systems listed above is credited. Repair model from WASH-1400 is assumed with mean time to recover of 19 hours.

Table 3.3-4 (Continued)

Repair and Recovery Actions

OPERATOR ACTION	FAILURE PROB	DISCUSSION
Recovery of battery charging	.8	Some non SBO sequences may still lead to battery depletion due to random failure of AC equipment supporting the batteries (i.e., chargers and power supplies). Repair over 5-hour period is credited (4 hours for battery depletion, 1 hour for core uncovery following loss of injection by DC dependent systems such as HPCI and RCIC). WASH-1400 mean time to recover of 19 hours assumed.
Recovery of offsite power 30 min 2 hr 4 hr 6 hr	 .64 .29 .15 .10	Station blackout sequences are quantified by breaking up the transient into time phases. The phases at Monticello are selected representing the capacities of systems and equipment to cope with a total loss of AC power (i.e., 30 min for inventory depletion to top of active fuel with no injection; 4 hrs for battery depletion, etc). Non-recovery probabilities for offsite power are derived from NSAC-147.
Recovery of either diesel generator 2 hr 4 hr	 .66 .47	On-site repair of diesel generators during station blackout is also credited. The repair activities are considered to be independent of off-site power recovery and utilize the same time phases. Repair failure rates are taken from the NUREG-CR/1362.

Table 3.3-5

Operator Actions for which
Detailed HEPs were Developed

FUNCTIONAL SEQUENCE	OPERATOR ACTION	DIAGNOSIS TIME	ERROR REDUCTION	RISK ACHIEVEMENT
Class 1A, 3B	Failure to depressurize the reactor	11 minutes	5E-5	150
	Alignment of bottled N_2 to SRVs or restoration of MCC41 on LOOP	240 minutes	2E-2	1.2
	Failure to control feedwater after SCRAM	16.5 minutes	7E-4	2.7
Class 1D	Manual initiation of low pressure systems	24 minutes	2.3E-3	1.2
Class 2	Failure to align torus cooling	2500 minutes	4E-6	60
	Containment Venting	1430 minutes	9.7E-4	1.2
Class 4	Failure to SCRAM the reactor	29 minutes	8.3E-4	2.5
	Manual initiation of SLC	24.5 minutes	4E-2	1.2
	Failure to control reactor level after SLC injection	9.5 minutes	0	2.1

Table 3.3-6
Common Cause Component Groups Modeled in IPE

1. Diesel Generators (failure to start and run)
2. Pumps (failure to start and run)
3. Motor-Operated Valves (failure to open or close on demand)
4. Circuit Breakers (failure to open or close on demand)
5. Batteries
6. Battery Chargers
7. Air-Operated Valves (failure to open or close on demand)
8. Safety/Relief Valves (failure to open or reclose on demand)
9. Check Valves (failure to open on demand; failure to remain closed)
10. Instrumentation and Control Components (failure to send signal or actuate equipment)

Table 3.3-7

Multiple Greek Letter Results

COMPONENT/MODE	NUMBER OF COMPONENTS	MULTIPLE GREEK LETTER BETA	GAMMA	DELTA	OTHER
CHECK VALVE FTO	2	.11			
	3	.13	.2		
	4	.13	.2	1	
STBY SW PUMPS FTS	4	.14	.67	1	
	2	.1			
VACUUM BRKS FTC	2	.3			
DRYWELL FANS FTR	2	.02			
AO VALVES FTO	2	.03			
RBCCW PUMP FTR	2	.04			
CORE SPRAY PUMP FTS	2	.13			
CORE SPRAY PUMP FTR	2	.1			
SLC PUMP FTS/R	2	.11			
LVL/PRESS XMTR	2	.29			
	4	.33	.44	.57	
MOV FTO/C	2	.07			
	3	.08	.17		
	4	.08	.2	.74	
RUNNING SW PUMP FTR	3	.06	.88		
RUNNING SW PUMP FTS	3	.14	.08		
RHR PUMP FTS	4	.08	.11	1	
RHR PUMP FTR	4	.1	.07	1	
RHR/CS PUMP FTS	6	.11	.08		
RHR/CS PUMP FTR	6	.05	.13		
HPCI/RCIC FTS/R	2	.04			
Battery FTR	2	.34			
EDG FTS	2	.03			
EDG FTR	2	.09			
SRV FTO	5	.27	.52	.69	.89

3.4 Results and Screening

3.4.1 Introduction

The purpose of this section is to summarize the overall findings resulting from the quantification of the Monticello front end analysis (level 1 PRA). Internal events and internal flooding are discussed separately. The IPE quantification focused on plant design features and operating characteristics that are most important to preventing core melt. Detailed descriptions of all of the dominant functional accident sequences are provided in this section. The dominant functional sequences are represented by accident class and sub-classes. Definitions of these classes are included in Section 3.1.5. Table 3.4-1 shows a summary of the CDF by accident class. The specific items discussed for each sequence are:

1. Description of accident progression, event timing, and containment failure mode, if applicable.

2. Specific assumptions to which the results are sensitive. Efforts were made to make assumptions consistent with best-estimate information.

3. Significant initiating events, human actions, and sensitive parameters.

The total CDF for Monticello internal events was 1.92E-5/yr. Core damage was conservatively defined as an extended period in which reactor level was less than two-thirds core height.

3.4.2 Application of Screening Criteria

The following screening criteria were used to identify sequences to discuss in this section of the report. This criteria is identical to the functional reporting requirements presented in Generic Letter 88-20.

1. Functional sequences with a CDF greater than 1E-6 per year. The functional sequences are grouped into accident classes. Within each damage class, sequences were generally identified by the dominant initiating events.

2. Functional sequences that contribute 5% or more to total CDF. The total CDF for internal events (excluding internal flooding), was 1.92E-5 per year. Any functional sequence greater than 9.8E-7 per year will be discussed. This criteria is almost identical to screening criteria 1 above.

3. Sequences determined by the utility to be important contributors to CDF or containment performance.

3.4.2.1 Class 1A

The sequences within this class were characterized by a loss of high pressure inventory makeup (QU) with a failure to depressurize the reactor vessel (X). These sequences were typified by the symbols TQUX from the failure headings of the event trees presented in Section 3.1. Class 1A sequences had a total core damage frequency (CDF) of 3E-6 per year, or 15% of the overall internal events CDF, excluding internal flood. The Class 1A sequences were dominated by sequences initiated by a loss of feedwater and loss of offsite power. Loss of feedwater and offsite power resulted in the unavailability of feedwater as an RPV water injection source.

For these sequences, reactivity control was successful (event tree heading C) and the SRVs cycled (event tree headings M and P) to control primary system pressure. Steam flow to the suppression pool through the relief valves occurred throughout the event because of decay heat. Loss of high pressure injection was the first functional failure which occurs for these sequences. Due to equipment failures or maintenance unavailabilities, feedwater, HPCI and RCIC are assumed not to fulfill their function of maintaining reactor water level (event tree heading QU). Failure of high pressure injection required reactor vessel depressurization once reactor level reaches the top of the fuel so that the low pressure injection systems could recover reactor vessel water level. Operator action was expected to be required in depressurizing the reactor (in accordance with the EOPs, the ADS is inhibited once the timer starts) but was not successful. No credit was taken for CRD since it has limited capacity and it must be manually aligned to increase flow. Without sufficient high pressure injection, water level steadily decreases because of cycling SRVs, until the core becomes uncovered and some degree of core damage is initiated with an intact containment. Without depressurization of the reactor and no high pressure injection the top of the fuel is reached about 25 to 35 minutes after the initiating event. Assuming the operation of a CRD pump in an injection lineup, the time to the top of the fuel can be extended to over an hour.

Assumptions applicable to the class were:

1. No credit was taken for alternate depressurization methods such as HPCI through use of the turbine, because failure to depressurize is dependent on operator action and all high pressure injection systems have failed.

2. The operator always inhibited ADS as directed by the EOPs. Plausible arguments may be made that if the operator inhibited ADS, the operator was fully aware of level conditions and would depressurize when required. Inhibiting ADS, making depressurization a manually controlled action, was considered to be a conservative and bounding assumption.

3. No credit was taken for recovering HPCI and RCIC in the time before core damage occurred.

4. The end state involved low reactor water level at the top of active fuel and the initiation of substantial core damage. This occurred between 25 minutes to 1 hour, depending upon whether a CRD pump was running in the scram mode. At this point, core damage was assumed to occur in the vessel with an intact containment. No environmental conditions of concern existed within the containment or reactor building up to the point of core damage.

5. No credit for injection of feedwater at full flow was assumed immediately after a scram. Feedwater was assumed to maintain water level near normal as opposed to filling the reactor. When credit was taken for additional water addition, a significant period of time was available for the operators to consider recovery actions.

6. Credit was taken for recovery of feedwater following a loss of feedwater initiating event, based on plant operating experience. This had the effect of significantly reducing the total contribution of this class of

events to the total CDF. For loss of feedwater events that resulted in a plant scram, 8 out of 9 recorded loss of feedwater events were quickly recoverable. One event in 9 was questionable whether feedwater could have been easily recovered.

7. MSIV closure, as well as HPCI and RCIC actuation all occur at low-low reactor level. Hence, following a transient in which all feedwater to the reactor was lost, it was assumed that there would be no high pressure makeup system available to restore inventory prior to MSIV closure.

8. Feedwater availability during manual shutdown and turbine trip initiating events is conservatively modelled. The potential for a feedwater trip on high level is considered immediately after initiation of the manual shutdown or turbine trip similar to reactor isolation events such as MSIV closure. Realistically, feedwater level control would be performed successfully avoiding this trip. The relatively high initiating event frequencies of turbine trip and manual shutdowns increased the significance of this assumption.

9. Feedwater reliability was affected by assumptions associated with failure of instrument panel Y20 which was assumed to fail the feedwater control system. Panel Y20 is significant because it supplies key SRV and containment vent pneumatic supply valves as well. This feedwater control dependency assumption was conservative because when the feedwater control system is taken to manual, as it would be after a scram, the control system is dependent on a different power supply.

10. The potential for recovery of offsite power within a half hour is considered for loss of offsite power events. If successful, feedwater is assumed to be available as an injection option.

11. SRV accumulators are assumed to allow vessel depressurization and low pressure pump injection for only a limited period of time for the loss of offsite power initiating event (one hour was assumed). The normal pneumatic supply valves for SRVs and inboard MSIVs became deenergized during a loss of offsite power because of the load shed of a non essential

motor control center. The hour made available by the accumulators is used to credit recovery of this MCC and restoration of pneumatic supply to the SRVs.

12. A loss of a single train of 125VDC power was assumed to result in a reactor trip. In reality, loss of DC power should not lead to a reactor trip but instead a manual shutdown after several hours, providing time for manual operation of DC operated components currently not credited.

13. No credit was taken for reactor pressure vessel (RPV) flooding as allowed by the Monticello emergency procedures following a loss of vessel water level indication. This assumption primarily would impact sequence initiated by a reference leg leak, in which instrument failures were assumed to result in operator uncertainty as to reactor level and relatively high error rates associated with failing to depressurize the vessel, or failing to initiate high pressure injection following automatic initiation failure. The resulting sequence frequencies were relatively low even with this assumption.

14. If battery depletion occurred due to unavailability of the chargers or some other reason, the associated battery was assumed to last for at least 4 hours. In that case, HPCI or RCIC was assumed to operate for this period of time with core damage occurring more than 5 hours following charger loss. Credit was taken for power recovery after a 5 hour period of time.

The most significant initiating events were:

1. Loss of offsite power, which accounted for 27% of the class 1A CDF. This event was significant because it caused a loss of feedwater as a high pressure injection system independent of HPCI and RCIC.

2. Manual shutdown, which accounted for 27% of the class 1A CDF. This event was significant because it had a large value compared to the other initiating events.

3. Loss of feedwater, which accounted for 20% of the class 1A CDF. This event was significant because it caused a loss of feedwater as a high pressure system independent of HPCI and RCIC. The significance of this event was reduced with a recovery factor based on plant operating experience.

4. Turbine trip, which accounted for 15% of the class 1A CDF. This event was significant because it had a large value compared to other initiating events.

The most significant operator actions were:

1. Failure to blow down the reactor vessel, which accounted for 79% of the class 1A CDF.

2. Offsite power recovery within thirty minutes. Failure of this recovery accounted for 29% of the class 1A CDF.

3. Prompt recovery of feedwater. Failure to restore feedwater accounted for 23% of the class 1A CDF.

4. Five hour repair factor for HPCI or RCIC on loss of battery charging. This restoration failure appeared in 14% of the class 1A CDF.

5. Two and four hour conditional offsite power recovery. Failure to restore power in these time frames accounted for 9% of the class 1A CDF.

6. Failure to maintain the reactor vessel depressurized following a loss of offsite power, either by failure to restore a motor control center (which is load shed during a loss of offsite power), or failure to align nitrogen bottles. Not successfully performing these actions accounted for 6% of the class 1A CDF.

7. Failure to manually control feedwater after the initiating event (preventing feedwater loss or recovering feedwater after a high level trip during events other than a loss of feedwater). Failure to restore feedwater during these events appeared in 5% of the class 1A CDF.

Important components identified for class 1A sequences were:

1. Random failures of HPCI and RCIC. HPCI and RCIC failures appeared in a large part of the class 1A risk. While available during loss of AC conditions, turbine driven pumps in the HPCI and RCIC systems are not otherwise considered as reliable as motor driven pumps.

2. Instrument panel Y20. Y20 accounted for about 12% of the class 1A CDF. Y20 supplied SRV pneumatic supplies and the automatic portion of the feedwater control system.

3. Common cause failure of feedwater injection check valves. These valves accounted for about 3% of the class 1A CDF. Their failure caused a failure of feedwater, condensate, HPCI, RCIC, and CRD (manually aligned to maximize flow).

3.4.2.2 Class 1B

Sequences within this class were characterized by a loss of offsite and onsite AC power and a loss of coolant inventory makeup. Following a loss of offsite power, the two emergency diesel generators would receive start signals. If the diesels either failed to start or run, a station blackout (SBO) results.

Class 2B sequences make up approximately 62% of the total internal event CDF, excluding internal flooding, with a CDF from all class 1B sequences of roughly 1.2E-5 per year. This class was dominated by a station blackout with a failure of high pressure injection after 4 hours as a result of battery depletion.

Assumptions which could impact the class 1B results included the following:

1. If the DC batteries became unavailable, it was assumed that the HPCI and RCIC pumps were unavailable.

2. No credit was taken for battery replacement. If AC power was not available for the battery chargers, the station batteries would eventually drain. In the IPE models there was no credit taken for replacing the batteries with other charged batteries.

3. No credit was taken for alternate depressurization through the use of the HPCI turbine.

4. The batteries were assumed to last for 4 hours. They may be able to last longer with actions such as load shedding.

5. No credit was taken for low pressure injection systems during station blackout (all rely on AC power).

6. If core damage was a result of random failure of HPCI and RCIC (phase 1 SBO) the sequence was assumed to have the same event timing as a class 1A sequence. Core damage was assumed to occur at about 25 minutes after the initiating event with an intact containment.

7. If the reactor was at pressure when battery depletion occurred, core damage was assumed 2 hours after the failure of high pressure injection systems. This is longer than the 30 minutes assumed for core recovery during other transients due to the lower decay heat load.

8. For a stuck open relief valve sequence with successful high pressure injection, core damage was assumed to occur at 2 hours after the initiating event. HPCI or RCIC was assumed to remain operable for this event until the low pressure trip setpoint was reached, extending the time to core damage.

9. Because AC power supplies SRV pneumatic supply valves, vessel depressurization was assumed to be possible with accumulators for only a brief period of time during an SBO (1 hour). Core melt was assumed to occur at high reactor pressure at 6 hours with an intact containment where HPCI or RCIC were successful.

10. Loss of Y20 was assumed to have minimal impact on the control room operators. Y20 supplies instrumentation in the control room and pneumatic supplies to the SRVs.

Significant events which contributed to class 1B sequences were:

1. Station batteries depleted after 4 hours resulting in failure of high pressure injection. This type of sequence accounted for 83% of the overall class 1B CDF. Offsite power and the diesel generators were not recovered within 6 hours for these sequences. Both 250VDC batteries drained in this time period because of the unavailability of battery chargers. High pressure injection was lost, and depressurization using the ADS was unavailable.

2. HPCI and RCIC random failure mechanisms. These type of events accounted for about 11% of overall class 1B CDF.

3. A stuck open relief valve which eventually depressurized the reactor below the HPCI and RCIC low pressure trip points. This type of event accounted for about 5% of the overall class 1B CDF.

4. Emergency diesel generator mechanical failure. This type of event accounted for more than 50% of the overall class 1B CDF. The CDF was also most sensitive to assumed changes in the overall reliability of the EDG.

5. EDG-ESW Pump unavailability. This type of event accounted for about 17% of the overall class 1B CDF similar to the EDGs. The CDF was also very sensitive to changes in the reliability of these pumps.

6. Maintenance and testing. Removal of EDGs for the purpose of corrective maintenance and testing appears in approximately 10% of the CDF for this accident class.

7. Diesel generator output breaker failure. This type of failure accounted for about 5% of the overall class 1B CDF.

8. Random offsite power failure during other initiating events. This type of event accounted for less than 2% of the overall class 18 CDF. This showed that the overwhelming majority of blackout events were caused by the loss of offsite power initiating event itself.

9. Common cause battery failure. This failure would prevent the diesel generators from starting and loading (this event appeared in approximately 1% of the class 1B sequences).

Important recovery actions identified were:

1. Thirty minute recovery of offsite power. This event in all SBO sequences.

2. Conditional recovery of offsite or onsite power at two hours. These events were in 88.0% of the SBO sequences.

3. Conditional recovery of offsite or onsite power at 4 hours. These events were in 83.0% of the SBO sequences.

4. Conditional recovery of offsite power at 6 Hours. This event was in 83.0% of the SBO sequences.

3.4.2.3 Class 1C

Sequences in this class were characterized by a failure to scram (ATWS) with a coincident loss of all inventory makeup. All events in this class were included in the analysis under class 4.

3.4.3.4 Class 1D

Sequences in this class were characterized by transient initiators with successful depressurization but a loss of both high and low pressure inventory makeup systems. Class 1D sequences made up 2% of the total CDF at Monticello. They had a combined sequence frequency of 3.6E-7 per year, reflecting a high level of redundancy of high and low pressure injection systems.

Assumptions associated with the class 1D sequences in general were:

1. Credit was taken for CRD operation for a SORV with successful high pressure injection. This system is able to inject sufficient flow rates to makeup for decay heat losses after high pressure systems have operated for 20 minutes or more. HPCI would provide adequate flow until it tripped

on low reactor pressure. Operation of HPCI early in this event makes it possible for CRD to be successful later in the event especially at low reactor pressure.

2. Credit was taken for use of the RHR Service water system for vessel makeup. This system had to be manually aligned from outside the control room. It injected through the LPCI injection lines.

3. Credit for offsite power recovery at one half hour was considered for loss of offsite power events.

4. Depressurization could occur at about 10 minutes and core uncovery could occur sooner than class 1A sequences because of the action to depressurize the reactor vessel. Core damage was assumed to occur at 25 minutes with an intact containment and vessel.

5. The operator was credited with starting low pressure ECCS pumps from the control room on a failure of both reference legs.

6. Recovery actions by the operator were considered for loss of offsite power events to regain battery charging, recovery of feedwater in loss of feedwater initiating events, cleaning of a plugged service water filter, and local manual opening of injection valves in the event of common 250 VDC battery failure.

Significant initiating events for this damage class were:

1. Loss of offsite power, which accounted for 68% of the class 1D CDF. Offsite power caused a loss of feedwater and condensate.

2. Loss of service water, which accounted for about 9% of the class 1D CDF. Failure of service water is assumed to cause a loss of feedwater.

3. Reference leg leak, which accounted for 6% of the class 1D CDF. A reference leg leak affected ECCS pump automatic initiation.

Significant events and recovery actions were:

1. Thirty minute recovery of offsite power. This action accounted for 70% of the class 1D CDF. Failure to take credit for this action had a small impact on total class CDF as the potential for recovery during the first 30 minutes is assumed not to be significant (~.5).

2. Common cause failure of all RHR and core spray pumps. These events accounted for about 18% of class 1D CDF. Class 1D results are relatively sensitive to assumptions regarding common cause failure of these pumps.

3. Common cause failure of RHR and core spray valves caused by mechanical or logic failures. Class 1D results are also sensitive to common cause assumptions regarding these valves. These events accounted for about 13% of class 1D CDF. Failure of RHR injection valves also prevented injection with RHRSW which limited the impact of that injection option.

4. Failure to cross-tie RHRSW to RHR. This event accounted for about 11% of the class 1D CDF. Class 1D results are not overly sensitive to this event, primarily because of the common injection line with LPCI and the currently assumed failure probability for this action.

5. Thirty minute action to start ECCS pumps in the control room. This action accounted for 7% of the class 1D CDF. Failure to take credit for this action had a moderate impact on increasing the class CDF as ECCS pumps receive automatic signals to start on low-low reactor level. The sequences in which this action is most significant is the reference line leak.

6. Recovery of feedwater on loss of feedwater initiating event. This action accounted for about 6% of the class 1D CDF. Failure to take credit for this action had a small impact on class CDF because a large fraction of the accident class is associated with loss of offsite power.

7. Failure to align CRD as an emergency injection source. This was used for SORV events. Manual alignment is required only during a loss of offsite power and accounted for about 6.0% of the class 1D CDF. If this option was not used, the CDF would rise very little, primarily because of the assumed failure probability for this action.

8. Recovery actions outside the control room in thirty minutes. This action involved local manipulation of breakers on loss of station batteries. (This action may or may not involve a short term immediate failure of batteries). Failure to take credit for this action had a small impact on increasing the class CDF because it was associated with very rare initiating events.

3.4.2.5 Class 2

Class 2 events were typified by accident sequences involving a loss of containment heat removal. A detailed discussion of this damage class also appears in section 3.4.4. The core damage probability for this accident class was calculated to be <1E-7 per year. The probability of containment failure alone, without consideration to injection after containment failure, was estimated to be less than 1E-5 per year.

The main condenser is the preferred decay heat removal system used during a normal shutdown until reactor pressure drops to the point where RHR shutdown cooling can be placed in service. Important support system requirements for the main condenser include offsite power, circulating water, condensate, instrument air, and service water.

If the main condenser is unavailable, RHR suppression pool cooling is used as an indirect decay heat removal system removing heat from the reactor vessel via the SRVs and the suppression pool. Suppression pool cooling is the principal mode of RHR containment heat removal credited in the IPE. Other operating modes of RHR which can remove decay heat include shutdown cooling, wetwell sprays and drywell sprays. Shutdown cooling can remove decay heat once reactor pressure has been lowered. Wetwell or drywell sprays are initiated per the EOPs at high containment pressures, and temperatures. Modes of RHR other than suppression pool cooling turned out only to have a significant effect on cutsets associated with torus cooling valve failures. Commonalities with the remaining portion of the RHR system reduces the impact of these other modes of RHR. All modes of RHR heat removal depend on RHRSW operation.

The existing containment vent is a system of last resort to prevent containment pressure from rising above the 56 psig design pressure. Because it contains low pressure duct work, the steam release from containment into the vent system would probably go into the reactor building and eventually be released through failure paths in secondary containment. In accordance with emergency procedures, venting would be initiated to maintain containment pressure below 56 psig. The vent would not be used to depressurize the containment. As a result, steaming rates to the reactor building would be approximately decay heat levels. This procedural guidance provided by the EOPs limits the environmental impact of venting on equipment in the reactor building. Required support systems include service water and the plant air system. Analysis of the environmental effects of venting were performed to confirm assumptions made in the IPE and are presented in Section 4.1.3 and 4.2.2-2.

An analysis was performed to determine how long it would take for the containment to pressurize to 56 psig assuming the core was adequately cooled but containment heat removal was lost. One to two days were required depending on whether makeup to the reactor was from the suppression pool or from sources external to the containment.

During this time, containment pressure would gradually increase if the main condenser, torus cooling, and shutdown cooling system were all unavailable. Recovery actions would be underway to correct existing failures. Containment sprays would eventually mitigate the containment pressure rise. If containment sprays and recovery actions were all unsuccessful, venting the containment would occur at the 56 psig design pressure.

If all DHR systems including venting failed, containment pressure would continue to increase at a slow rate driven by the decay heat rate to approximately 103 psig. Two to three days are necessary to pressurize the containment to its ultimate capacity. At this point the containment is assumed to fail at the drywell head enclosure or torus expansion bellows. Releases from these locations, would primarily affect the refuel floor and the torus area. Because of these failure locations, injection systems in the turbine building and some systems in the corner rooms of the reactor building are likely to remain operable after containment failure. It is highly probable that continued injection to the vessel after containment failure will prevent core damage. Again analysis of environmental conditions in the reactor building were performed to confirm equipment survivability assumptions (see Sections 4.1.3 and 4.2.2-2).

Long term equipment recovery action was considered based on the time to reach 56 psig in the containment. At 56 psig it was assumed that all reactor building recovery actions would stop for personnel safety reasons. Actions in the turbine building or elsewhere may continue but this was not credited.

Actions which could significantly prolong the event, but not necessarily prevent it, also were not credited. Those actions included use of the RWCU system, either in feed and bleed or heat exchanger heat removal modes, or use of an external spray source such as RHRSW, which would increase the water mass in the torus.

Assumptions associated with Class 2 events included:

1. Given the significant time frame available for RHR initiation, operator failure to diagnose and initiate RHR was considered but did not have a significant contribution to risk. The limited significance of the operators contribution to RHR system initiation failure is further supported by the structure of the EOPs which specify actuation of various DHR methods based on multiple diverse indications associated with loss of DHR (ie, torus temperature, containment pressure and containment temperature).

2. Containment failure pressure was estimated to be 103 psig. It is recognized that the actual containment failure pressure and failure location are uncertain. The Monticello IPE used the best available plant and generic information to determine what the failure pressure and location should be. When the containment fails, it is also unclear whether the failure would be just enough to relieve only decay heat so the containment would remain pressurized, or large enough to completely depressurize the containment in a short period of time. For this

analysis, the occurrence of a catastrophic rupture large enough to depressurize the containment is assumed. Best estimate analysis of the structural capability of containment suggests that the failure location would most likely be in the drywell head area or torus expansion bellows.

3. The consequences of containment failure are also uncertain. The Monticello IPE assumed a number of systems would remain operable after containment failure. Turbine building systems were assumed to be unaffected by containment failure. However, continued operation of equipment located in the reactor building was also assumed and is based on both the size and location of the expected containment failure noted in 2 above.

4. The ability to use SRVs above 70 psig containment pressure to use low pressure systems was recognized as a potential concern. The Monticello IPE assumed that the SRVs would be unavailable above 70 psig because of pneumatic backpressure, requiring the use of high pressure injection systems such as feedwater or CRD for the duration of the event. If the containment were to depressurize as a result of a large failure in containment on overpressure, SRV operation and reactor depressurization would no longer be prohibited.

5. Even though containment failure is not expected for several days during a total loss of DHR, recovery actions above 56 psig in containment were not credited for personal safety reasons.

6. The impact of prolonging the event by using external water spray sources or RWCU was not considered. No credit was taken for use of the RHRSW system as a source of containment spray. This conservatism was believed to be appropriate because even if successful, service water spray may only delay the need for some other means of heat removal. Since operation of the sprays cannot be initiated if the vacuum breakers become covered with water, it was recognized that spray operation from sources external to the containment cannot continue indefinitely. It was also understood RHRSW has commonalities with suppression pool cooling which must be unavailable in order to require spray for external sources.

7. The success of venting, or the use of the containment sprays with water from sources external to the containment, were assumed to have no negative effect upon the net positive suction head (NPSH) for injection systems taking suction from the suppression pool. EOPs instruct monitoring NPSH to protect operation of injection systems. As injection to the reactor is required only at decay heat makeup rates, the potential for NPSH concerns is further limited.

8. CSTs normally contain over 220,000 gal of water thus having the capability to make up for nearly 3 days of decay heat. This combined with the capability to makeup from numerous other sources (demineralized water, radwaste, condensate makeup, service water) resulted in limiting the importance of CST capacity as a potential failure mode.

Significant initiating events for this damage class were:

1. Loss of Service Water - 50%.

2. Manual Shutdown - 32%.

3. Turbine Trip - 10%.

Manual shutdown and turbine trip make up a significant portion of this accident class simply because they are the most frequent initiating events requiring decay heat removal. Loss of service water has a relatively low frequency but impacts the operation of two of the potential decay heat removal systems; the main condenser and containment venting, plus makeup to the reactor in the form of feedwater and CRD.

Significant recovery actions identified were:

1. Recovery of DHR equipment over approximately a two day period of time. This action was involved with all cutsets in this functional damage class.

2. Initiation of torus cooling. Failure to perform this action appears in only a few percent of this accident class and indicates that mechanical or electrical failures of RHR dominant the reliability of this system.

Again, the time available to initiate the system, combined with multiple diverse indications suggesting the need for its initiation, leads to limited potential for this failure.

3. Recovery of instrument air given a loss of service water. This action appears in approximately 50% of the class 2 CDF sequences and is a result of service water impact on not only DHR but high pressure makeup.

4. Failure to align makeup sources to the condensate storage tank. A bounding value of 0.1 was assigned to this operation action. In fact, sufficient inventory is normally available in the CSTs that the importance of this action is limited.

Significant equipment and components identified were:

1. Power supplies associated with essential buses, essential load centers, and UPS panel Y20. These AC power supplies feed containment vent valves, feedwater control valves, SRV pneumatic supply valves, and RHRSW heat exchanger discharge valves. Various combinations of power supply failures can result in a loss of one or more DHR systems.

2. RHRSW components such as pumps and heat exchanger discharge valves. RHRSW is the ultimate heatsink for all modes of RHR.

3. Torus cooling injection valves.

4. Air system components, such as air compressors and receiver tank relief valves.

5. Service water components, such as pumps and the strainer.

Significant initiating events were manual shutdown, turbine trip, and loss of service water. The relative importance of the sequences involving manual shutdown and turbine trip is a result of those initiators having a larger frequency than other initiators. The fact that they dominate the results indicated that random failures of equipment had a much greater impact than relatively infrequent initiating events which could disable DHR systems. The analysis suggests that assumptions regarding the operability of injection equipment following containment failure drive the results of this accident class. Sensitivity Studies on the environment in the reactor building following containment failure were performed to verify these assumptions.

3.4.2.5 Class 3

No individual class 3 damage classes met any of the level 1 screening criteria. However, the individual damage classes will still be discussed here. The total CDF for all LOCAs was 1.1E-6 per year, or about 6% of the total CDF, excluding internal flooding.

Important initiating events for all of class 3 LOCA events were:

1. Medium LOCAs. They accounted for 44% of the total LOCA CDF.

2. Large LOCAs. They accounted for 26% of the total LOCA CDF.

3. Small LOCAs. They accounted for 20% of the total LOCA CDF.

Class 3A sequences involved sequences initiated by a reactor vessel rupture and failure of ECCS injection systems, with the containment intact after the rupture. The initiating events frequency consisted principally of transient type events combined with common cause failure of all SRVs to open as opposed to random failure of the vessel itself. This class of sequences contributed less than 1% to the total estimated CDF at Monticello and 11% of the total LOCA CDF, with a total CDF of 1.1E-7 per year.

Class 3B sequences were initiated by small or medium LOCAs for which the RPV could not be depressurized after failure of high pressure injection systems. These sequences contributed 2% to the overall CDF at Monticello and 43% of the total LOCA CDF, with a combined sequence frequency of 4.7E-7 per year. These sequences resulted in core uncovery in 10 minutes with an intact containment. Sequence characteristics which dominate this damage class were those in which manual depressurization of the reactor vessel failed to occur (like the transient events it is assumed ADS is inhibited for LOCAs).

Class 3C sequences were initiated by a medium or large LOCA. High pressure injection systems may be successful, but eventually the LOCA causes vessel depressurization. Vessel depressurization results in the turbine driven high pressure systems to be unavailable. Core damage is assumed to result from inadequate low pressure inventory makeup to the RPV. Vapor suppression was adequate during RPV blowdown. Core damage occurred with an intact containment at low reactor pressure. These sequences contribute 2% to the overall CDF at Monticello, with a combined sequence frequency of about 4E-7 per year.

The dominant class 3C sequence was a medium LOCA with loss of all injection systems (S1V). In this sequence, a medium LOCA occurred. Vapor suppression was successful, but low pressure injection systems failed. The sequence frequency accounted for 82% of the risk in this damage class.

Class 3D sequences were initiated by a large loss of coolant accident or RPV rupture for which vapor suppression was inadequate (AD). These sequences contributed less than 2% to the overall CDF at Monticello, with a combined sequence frequency of about 3E-7 per year. The large LOCA made up about 73% of the CDF for this damage class. In this sequence the LOCA occurs coincident with loss of multiple vacuum breakers between the drywell and wetwell airspace. This is assumed to result in suppression pool bypass and over-pressurization of containment. Continued makeup after containment failure was also assumed to be unavailable. The frequency of this event was 2E-7 per year. As noted in the discussion of important assumptions below, success criteria for this accident class is believed to be conservative.

Assumptions and uncertainties associated with this damage class were:

1. It was assumed that 10 minutes were available to manually depressurize for a medium LOCA with failure of high pressure injection as opposed to the 25-35 minutes available during transients.

2. RCIC was not credited as an injection source for LOCA events as the steam flow rate was assumed to be greater than its makeup capability.

3. Vapor suppression failure could be countered by depressurization of the reactor through SRVs in accordance with EOPs. SRV operation in this manner directs steam directly from the reactor to the suppression pool, restoring the vapor suppression function.

4. Two vacuum breakers failing open were assumed to defeat the vapor suppression function for a medium or large LOCA. One vacuum breaker failing open was assumed to defeat the vapor suppression function for a small LOCA. These vacuum breaker failures may be tolerable depending on the size and location of the reactor coolant system breach. Further, the potential for vacuum breakers cycling depends on break progression and use of sprays which may affect the vapor suppression availability.

MAAP and the plant training simulator were used to verify vapor suppression assumptions. A SORV in the torus air space simulated approximately a medium LOCA with failed vapor suppression. The results of the run indicated that this situation could have gone indefinitely without failing the containment. Low pressure systems eventually kicked in before containment pressure had reached containment failure pressure. Injection of cold water into the vessel provided sufficient steam condensation to reduce the pressurization of the containment.

A small to medium LOCA with multiple vacuum breakers open was evaluated in the simulator. Similar to the SORV, Containment failure was not reached for this event.

A large LOCA with failed vapor suppression was evaluated in the simulator and with MAAP. Nearly all vacuum breakers were opened before the LOCA, and containment failure was not reached. The differential pressure between the wetwell and the drywell remained sufficiently great the overcome the static head in the downcomers and provide significant vapor suppression. Also, low pressure pumps came on quickly injecting cold water which minimized the steam input to the drywell.

All of the above results suggest that the success criteria for vapor suppression used in the IPE was conservative.

5. ATWS events with a failure of both recirculation pump field breakers to trip were assumed to cause a large LOCA. Simulator verification of this sequence, showed that the recirculation pump drive motor breakers would also have to fail to reach a pressure high enough to cause a RPV overpressure. It was observed that the higher the reactor pressure for this event, the more flow would be reduced from feedwater pumps, limiting the power level of the event.

Important human actions identified were:

1. Failure to depressurize the reactor within ten minutes for a medium LOCA. This action accounted for about 10% of the total LOCA CDF, and applied to high pressure injection failure or vapor suppression failure events.

2. Failure to maintain the reactor depressurized either by aligning bottled N2 or restoring power to solenoids in the N2 supply to SRVs given random failures in the power supplies to the solenoids. This action accounted for 14% of the total LOCA CDF.

3. Failure to depressurize the reactor for small LOCAs with a failure of high pressure injection. This action accounted for about 7% of the total LOCA CDF.

Important component failure events were:

1. Failure of two vacuum breakers to close. This event accounted for 20% of the total LOCA CDF.

2. Common cause failure of RHR and core spray pumps to start or run. This accounted for 19% of the total LOCA CDF.

3. Common cause failure of RHR and core spray injection valves to open. This event accounted for 11% of the total LOCA CDF.

4. Common cause failure of all SRVs to open. This event accounted for 11% of the total LOCA CDF.

3.4.2.7 Class 4

These sequences involved failure to insert negative reactivity into the core, which eventually is assumed to lead to a containment challenge due to high containment pressure. The ultimate capacity of containment in the drywell is assumed to be reached within an hour for an event in which failure to scram from 100% power occurred. Containment failure was then assumed to fail all reactor building injection systems which resulted in core damage. Operator action to inject SLC or control reactor level early in the event would prevent or extend the time to containment failure. The class 4 ATWS events at Monticello accounted for a total frequency of 2.5E-6 per year, which is approximately 13% of the overall CDF.

Assumptions associated with this accident class included the following:

1. The reactor was assumed to be at 100% power when the failure to scram occurred. Lower power levels would present a less severe challenge to some systems, and extend the amount of time available for the operator to take action.

2. All control rods failed to insert, and all subsequent operator actions taken to insert the control rods also failed. Thus, the power level was remained high throughout the event until SLC was effective.

3. The time to pressurize the containment to its ultimate capacity was determined using MAAP assuming reactor power with both recirculation pumps tripped and reactor level near normal. Controlling power level by lowering the vessel water level to the top of the active fuel was given limited credit. This action could extend the amount of time available to the operator to take mitigating actions such as injecting boron. Faced with the symptoms requiring boron injection, it was assumed the operator would elect to initiate SLC before lowering reactor level. Thus, level control is a potentially significant option for events in which it is postulated that mechanical or electrical failure of SLC has occurred.

4. The time available to the operator to initiate SLC was based on the heat capacity of the suppression pool below a bulk temperature of 260°F. This pool temperature conservatively accounts for any potential for incomplete vapor suppression as suppression pool temperature rises. This assumption applied even if boron had been injected, but did not completely shut down the reactor. Assuming adequate vapor suppression capability up to the point of containment failure would provide a somewhat longer period of time for operator action to initiate SLC.

5. Containment venting or RHR system operation were assumed to be inadequate for containment heat removal if the reactor was not shut down.

6. In cases for which boron dilution occurred after the reactor was shut down, no credit was taken for recovery (e.g., terminating dilution or injecting additional boron) over the time frame in which dilution might occur.

7. Overpressurization of the RPV during an ATWS event with failure of RPT was assumed to lead directly to a LOCA. Overpressurization was assumed to occur if a single recirculation pump field breakers failed to trip. In reality, investigation of various components within the primary system might yield less severe consequences such as loss of pump seals within the primary system, permitting additional pressure relief.

   Also, the failure to trip a single pump during ATWS was evaluated using the simulator. Pressure reached 1180 psig with one RPT. All 8 SRVs opened, with reactor power at about 68%. This sequence indicated that a LOCA would not occur with one RPT failure during an ATWS.

   With both pumps running, reactor pressure increased which increased reactor power because of collapsed voids. Feedwater flow could not control level because of increased pressure and increased power. Reactor pressure reached about 1300 to 1400 psig and level dropped to -47 inches. The recirculation pumps tripped at -47 inches from drive motor breaker trip. It appeared from this scenario that a LOCA may be less likely than assumed in the sequence quantification because of the additional redundancy of the drive motor trip and inability of feedwater to maintain level at very high reactor pressure.

8. Failure of the primary containment was assumed to result in a substantial and energetic release of steam to the reactor building. Environmental conditions in the reactor building were assumed to result in loss of inventory makeup and core damage.

9. Alternate boron injection was considered for turbine trip and reactor trip without turbine trip events which involved the additional failure of SLC mechanical or electrical components. Alternate boron injection was not credited for initiating events leading to isolation of the reactor from the main condenser, because of the relatively limited time available to align these systems locally in the reactor building before the containment pressure rose to levels above design.

10. With the exception of alternate boron injection for events in which the main condenser is available, recovery factors were not considered for ATWS.

The most significant initiating events in this damage class were:

1. Turbine trip, which accounted for 51% of the total damage class CDF. This initiating event had a significantly larger frequency than the other initiating events.

2. MSIV closure, which accounted for 23% of the total damage class CDF. This initiating event resulted in the immediate unavailability of the main condenser to handle reactor power.

3. Loss of feedwater, which accounted for 14% of the total damage class CDF. This initiating event resulted in the immediate unavailability of the main condenser because of MSIV closure caused by low reactor level.

Important operator actions identified were:

1. Failure to inject standby liquid control for the various initiating events. This action accounted for 54% of the total CDF of this damage class.

2. Failure to control level after successful boron injection causing recriticality. This action accounted for 17% of the total CDF of this damage class and could be reduced significantly with credit for operator recovery.

Important component failure events were:

1. Mechanical failure of the control rods to insert. This point estimate event accounted for 87% of the total CDF of this damage class. Mechanical CRD failure derives the risk associated with ATWS given the assumptions made in sequence quantification.

2. Electrical failure of the RPS system. This point estimate event accounted for 13% of the total CDF of this damage class. This event is limited in significant due to the diversity provided by ARI.

3. Failure of both recirculation pump field breakers to trip. These events accounted for 18% of the total CDF of this damage class. These events were assumed to lead to a LOCA. As noted earlier, MG set drive motor breaker trip may provide an adequate backup to RPT and even if both pumps continue to run, RPV pressurization may not be significant due to existing SRV capacity.

4. SLC hardware failures accounts for roughly 10% of the core damage frequency associated with ATWS.

3.4.2.8 Class 5

Class 5 consists of LOCAs with a bypassed containment, including interfacing system LOCAs, which occur outside containment. The break location can bypass the source term mitigation features associated with the containment, suppression pool, and containment sprays. This class of sequence represented less than 1% of the total CDF at Monticello. No sequences in this damage class met any screening criteria.

Important and sensitive assumptions included:

1. The initiating event frequency for high pressure piping was based on generic pipe failure rates.

   High pressure piping outside containment considered as a part of this analysis includes that associated with:

   - Main steamline
   - Main feedwater line
   - HPCI steam supply
   - RCIC steam supply

   Smaller high energy lines associated with reactor cleanup system were considered bounded by the breaks listed above and would have limited impact on the environment in the reactor or turbine buildings and the operation of other core cooling equipment.

2. The exposure of low pressure piping outside the containment to primary system pressure was considered to be a result of one or a combination of the following:

   - Interfacing isolation valve failures
   - Human error during surveillance testing

   Low pressure systems considered in the interfacing system LOCA analysis for Monticello included:

   - LPCI injection lines
   - Core spray injection lines
   - Head spray
   - Shutdown cooling suction

On exposure of low pressure piping to reactor pressure, it is recognized that the ultimate rupture strength of the piping is many times the design. While leakage through the interfacing system may occur, there was only limited potential for gross rupture of the piping. A conditional pipe rupture probability of 0.01 was used on exposure of low pressure piping to full RCS pressure.

3. Equipment in RHR corner rooms was assumed to remain operable if a break occurred outside the corner room. In addition to the low initiating event frequency this was a relatively significant assumption which caused the overall class 5 damage class CDF to be low. Verification of the environment in the reactor building was performed for various break locations within existing interfacing systems and is presented in Sections 4.1.3 and 4.2.2-2.

4. A break in the steam tunnel was assumed to disable the division 2 power supplies in the turbine building due to the existence of a blowout panel with an unobstructed pathway to the upper 4KV area.

3.4.2.9 Internal Flooding

The total core damage frequency for internal flooding events is estimated to be less than 7E-6/year. The internal flooding evaluation concluded there were two core damage sequences greater than 1E-6/year. The first sequence involved a service water line break in the reactor building which was assumed to fail all high pressure injection systems. The second event involved a service water break in the 931' east turbine building area which propagated to main access control and station batteries. High pressure injection systems were rendered inoperable by this event.

No flood initiators were identified that could result in inadequate core cooling without the additional random failure of unaffected core cooling equipment. A detailed discussion of potential flood initiators, flooding zones, affected equipment, and a breakdown of accident sequences is presented in Section 3.4.5.

3.4.3 Vulnerability Screening

No vulnerabilities were identified as part of the IPE process for Monticello. The criteria used to determine if any vulnerabilities existed were:

1. Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs?

2. Do the results suggest that the Monticello core damage frequency would not be able to meet the NRC's safety goal for core damage?

Neither of these criteria lead to the identification of potential vulnerabilities for the Monticello plant. The accident classes that contribute to the potential for core damage are similar to those identified in PRAs of comparable facilities such as those evaluated in NUREG-1150 and the IDCOR IPE Methodology. Also, while it does not include the contribution from external events, the overall core damage frequency of 2E-5/year is only a fraction of the NRC's safety goal for core damage of 1E-4/year, leaving ample margin for accommodating risks of other events such as earthquakes or fires.

Another term frequently used in this report is "significant insight". Many insights were generated as part of this study. In general, a significant insight was a system, component, or action which influenced the results of this study more than other events evaluated. A significant insight may involve:

1. A unique safety feature which significantly drove risk either by limiting the potential for or contributing to core damage.

2. A system interaction effect which had a relatively important impact on the overall results of this study.

3. A component failure mode or operator action which had a significant impact on the results of an accident class or the overall results.

4. A failure or operator action worthy of consideration of a recommendation.

5. A critical operator action which had limited procedural guidance.

Detailed discussion of insights derived from the Monticello IPE are presented in Section 6.0.

3.4.4 Loss of Decay Heat Removal

Generic Letter 88-20, section 5, discusses resolution of USI A-45 "Shutdown Decay Heat Removal Requirements." This section outlines the analysis of the Monticello decay heat removal (DHR) capability, as required by the generic letter. The conclusion of this analysis is that Monticello DHR capability does not contribute significantly to the potential for core damage. For the purposes of this discussion, DHR is defined as decay heat removal from the containment.

As part of the Monticello IPE, the following topics related to DHR were analyzed and will be discussed:

1. The issues discussed in USI A-45.

2. Systems available at Monticello for DHR.

3. Analysis of plant response to a total loss of DHR.

4. The results of the IPE and a discussion of what factors have the most influence on reliability.

5. Proposed modifications.

6. Uncertainties.

7. Conclusions.

3.4.4.1 Relevant USI A-45 Issues

The various analyses performed to resolve the DHR issue were based on NUREG 1289. Six specific alternatives were discussed:

1. No corrective action.

2. Perform detailed risk assessment.

3. Install various modifications.

4. Install hardpipe containment vent.

5. Install a dedicated hot shutdown DHR system.

6. Install a dedicated cold shutdown DHR system.

The focus of this study was on item 3 from the list above. Item 1 was not considered because actions to identify and address DHR risk were performed as a part of this evaluation. Item 2 was effectively implemented in the form of a detailed PRA in response to Generic Letter 88-20. Item 4 is currently planned in response to the NRC's requests in GL-89-16. Alternatives 5 and 6 are not cost beneficial based on the very small residual core damage frequency (CDF) with or without the hardpipe vent. Various modifications discussed in recommendation 3, as well as other plant specific recommendations, will each be described separately.

3.4.4.2 Systems Available for DHR

There are four possible methods by which heat can be removed from the reactor vessel and/or containment:

1. Main condenser.

2. RHR in suppression pool cooling, wetwell spray, drywell spray, or shutdown cooling mode.

3. Reactor water cleanup system (RWCU) either through the non-regenerative heat exchanger or in a feed and bleed mode to the main condenser.

4. Containment vent.

3.4.4.2.1 Main Condenser

The main condenser is the preferred method of removing decay heat and depressurizing the reactor until the shutdown cooling system can be placed in service. Fault tree analysis of the main condenser suggests a failure rate of 1.7E-2 over a 24 hour mission time following a reactor trip. A train of MSIVs must be open to permit steam flow from the reactor. Support systems include offsite AC power and instrument air.

Effects of Important Initiating Events on Maintaining the Main Condenser:

Initiating Event	Failure Probability
Transients	1.7E-2
Loss of offsite power	1.0
LOCA	1.0
Stuck open SRV	1.0
Loss of feedwater	0.1
MSIV closure	0.1
Loss of main condenser	0.33
Loss of instrument air	0.5
Loss of service water	2.7E-2

As shown above, the availability of the main condenser is highly dependent on the initiating event. During turbine trip or manual shutdown, the condenser is relatively reliable and its operation is principally dependant on support equipment such as the turbine bypass valves, steam seals, circulating water, etc. For events involving a loss of offsite power, major support equipment, such as circulating water pumps and condensate pumps will not be operable, eliminating the main condenser as a heat sink.

LOCAs and stuck open SRVs are assumed to depressurize the reactor to the point that little steam flow to the main condenser occurs, even if MSIV closure on low steam line pressure is bypassed.

The remaining initiating events listed above also directly or indirectly result in the loss of the main condenser. The main condenser failure probabilities for these events reflect simple early recovery factors:

Loss of feedwater	The main condenser is dependent on the operation of the condensate system for maintaining steam seals and the inter and after condensers. Only a portion of loss of feedwater events also result in loss of the condensate system. Further, review of operating experience indicates that feedwater has been recovered in nearly all loss of feedwater initiators (see Section 3.3 Table 3.3-4).
MSIV closure	Spurious closure of an MSIV results in high steam flow in the other steam lines and closure of the remaining MSIVs. Reopening one of the other MSIV paths in a manner similar to that performed during plant startup is the basis for the recovery factor shown. The action is simple, requires no repair and can be taken from the control room. It is noted that MSIV opening must take place during a loss of DHR prior to the containment reaching 45 psig. The inboard MSIV pneumatic pressure would be insufficient to permit opening the valves at higher containment pressures.
Loss of main condenser	This recovery factor is based on plant operating experience with this type of initiating event. Actions involve placing the main condenser in service, as opposed to repairing failed equipment.

Loss of instrument air	This system supports the operation of both the MSIVs and the containment vent valves. This recovery factor is based on plant operating experience. The only loss of instrument air precursor to occur at Monticello, air receiver tank relief valve failure, was recovered prior to loss of air pressure.
Loss of service water	This initiator indirectly affects the availability of the main condenser and the vent valves through cooling of the plant instrument air compressors. The recovery factor is based on operator actions to supply an alternate cooling source to the air compressors (i.e., from the fire system).

Important Hardware Failures

Failure	Contribution to Failure Probability
Turbine bypass valves	71%
Instrument air single failures	11%
Service Water strainers	4%
Main condenser support equipment random failures	4%
Condensate pumps	2%
Random loss of offsite power	1%
Common cause failure of air compressors	1%

Analysis of the main condenser fault tree indicates a potentially significant contribution to system unavailability is from the turbine bypass valves and hydraulic control equipment, estimated to contribute slightly more than 2/3 of the system failure rate. Operation of the bypass valves is dependent on a single hydraulic oil pump which must start and continue to run in order to maintain steam flow to the main condenser following stop valve closure.

Instrument air system loss could eventually lead to closure of the outboard MSIVs. Random failures leading to loss of the instrument air system contribute approximately 11% of the main condenser system failure rate and include I&C failures to start the compressors or flow diversion through components such as the air receiver tank relief valves. The majority of these failures can be overcome by operator action to start air compressors or to isolate flow diversion paths locally.

Main condenser support equipment associated with the steam seal regulators and circulating water provide the next contribution to main condenser unavailability, approximately 4%. Common cause failure of the condensate pumps provide only a small contribution to the loss of the main condenser as a heat sink, approximately 2%.

The remaining contributors to the main condenser fault tree are significant only in that they are also contributors to containment venting. These events include service water strainer plugging, approximately 4%; air compressor common cause failure; approximately 1%; and loss of offsite power coincident with the transient in progress; approximately 1%. Both the service water strainer plugging and offsite power loss are subject to recovery actions, particularly given the time available to reestablish a heat sink given DHR failure.

Important Operator Actions

For transient events and manual shutdowns, turbine bypass valve operation is automatic and main condenser heat removal requires only monitoring by the operator with possible action to depressurize the reactor through manual control of the bypass valves. The most important operator actions are recovery and reestablishing the main condenser for events in which it is lost as a result of the progression of the transient but is otherwise available. These actions include those associated with the transient initiators noted above.

3.4.4.2.2 RHR

If the main condenser is unavailable, RHR can be used to remove decay heat by using one of four modes.

RHR suppression pool cooling would be used as an indirect decay heat removal system removing heat from the reactor vessel via the SRVs and the suppression pool. Suppression pool cooling is initiated by the operator on reaching a suppression pool temperature of 90°F. Either offsite or onsite emergency power may be used to operate RHR pumps and valves. The RHRSW is the ultimate heat sink for any mode of RHR operation. The plant air system supplies air to the RHRSW heat exchanger discharge valves, but these valves have their own dedicated air systems, they fail open on loss of air, and also have manual operators on them to allow local operation.

Shutdown cooling can remove decay heat if the reactor is at low pressure. Use of this system requires depressurizing of the reactor to less than 40 psig and opening both shutdown cooling suction valves from the RPV in addition to LPCI injection valves. EOPs instruct the operator to depressurize the reactor through use of the main condenser, TBVs, or SRVs early in an event in order to place shutdown cooling in service. Because of commonalities with suppression pool cooling, shutdown cooling provides additional redundancy only for sequences which resulted from torus cooling valve failures. An extended period without DHR may result in increased containment temperature and pressure conditions. In this situation, a group 2 isolation signal resulting from 2 psig containment pressure would preclude shutdown cooling from being placed in service.

On increasing containment pressure, wetwell sprays are to be placed in service as another means of providing RHR operation. Wetwell spray is required by EOPs prior to exceeding 18 psig. Drywell sprays are initiated above 18 psig provided that containment conditions do not exceed the drywell spray initiation limit.

Any mode of RHR would be unable to reduce containment pressure on a long term basis without RHRSW available as a heat sink, since the torus water would become saturated. Spraying the drywell or wetwell with RHRSW could reduce containment pressure and limit the rate of pressurization. A crosstie between RHRSW and RHR piping to the containment is provided at Monticello to accomplish this action.

The random failure probability of the RHR system for the purposes of suppression pool cooling is estimated to be 2.8E-3 per demand. This failure probability assumes a reactor shutdown in which offsite power remains available (i.e., no dependency on diesel generators), a single RHR pump success criteria and a mission time of 24 hours.

Dominant contributors to system reliability will be discussed in this section, as derived from the suppression pool cooling fault tree.

Effects of Significant Initiating Events:

Initiating Event	Torus Cooling Failure Probability
Transients	2.8E-3
Loss of offsite power	3.0E-3

The failure probability of the RHR system (suppression pool cooling mode) does not vary significantly for the spectrum of initiating events. Its support systems include only AC power DC power and RHR Service Water. The increase in failure probability for loss of offsite power events noted above reflects the additional dependence of the suppression pool cooling and RHR Service Water systems on emergency diesel generators.

Important Hardware Failures:

Failure	Contribution to Failure Probability
Common Cause Failure of Torus Cooling Valves	28%
Random Failures in Both Loops of RHR Service Water	18%
Random Failures in One Loop of RHR and the Opposite Loop of RHR Service Water	11%
Common Cause Failure of RHR Service Water Valves	6%

Common Cause Failure of RHR Service Water Pumps	6%
Random Failures in Both RHR Loops	5%
Common Cause Failure of RHR Pumps	2%
Operator Action to Initiate System	1%

The largest single contributor to suppression pool cooling unavailability is the common cause failure of the torus cooling valves which accounts for approximately 28% of system failure probability. It is noted that redundant valves in the wetwell spray, drywell spray and LPCI mode of RHR can potentially limit the significance of torus cooling valve failure. In addition, local manual operation of the valves is possible given the time frame available to initiate the system (i.e., days).

Random failures in both loops of RHR Service Water follow as the next significant contributor to suppression pool cooling unavailability, approximately 18%. These failures include:

RHRSW Loop out for maintenance
Pump or control valve I&C failures
RHRSW heat exchanger discharge valve fails to open
Relief valve flow diversion
Failure to restore following maintenance

The valve portion of the trains contributes more significantly to loop reliability than the pumps because of pump redundancy, there are two pumps in each train, only one is required for success of RHRSW. A number of the failure modes noted for the RHRSW loops may be corrected by local manual operation.

Combinations of one RHR loop and the opposite RHR Service Water loop failures are the next most significant failures, approximately 11%. Contributors to RHR loop failure include:

RHR loop out for maintenance

Torus cooling valve fails to open

Similar to RHRSW, valve failures contribute more significantly to RHR loop unavailability because of RHR pump redundancy. In addition, RHR pumps are crosstied such that any of the four pumps can discharge to either torus cooling loop.

Common cause failure of RHR Service Water control valves and pumps contribute approximately 6% each to the failure of RHR. The RHRSW valve common cause contribution is significantly less than the RHR loop common cause because there are only two discharge control valves in the RHRSW system. Since torus cooling valves provide containment isolation, there are two valves in series in each loop. RHRSW pumps contribute little again because all four pumps must fail before heat removal capability is lost.

Random failures in both RHR loops contribute approximately 5% to suppression pool cooling failure. The makeup of RHR loop random failures was noted above and is less than that for an RHRSW loop because of a smaller failure rate for motor operated valves than control valves and a shorter fraction of time devoted to maintenance.

Common cause failure of RHR pumps is relatively small, approximately 2% due to the redundancy provided by four pumps, all which must fail to disable suppression pool cooling.

Important Operator Actions

Failure to initiate suppression pool cooling

Repair and recovery of RHR

Operator action to initiate RHR is only a limited contributor to failure of suppression pool cooling because of the significant time available to actuate the system (days) and the multiple and diverse indications of the need to actuate the system (torus temperature and containment pressure).

As noted in the description of the various contributors to system failure, recovery of the system may be simple in many cases, often involving local manual operation of system components such as valves or breakers. In addition, repair of failed components may be likely because of the long time available before decay heat pressurizes the containment to its ultimate capability. Repair and recovery is therefore considered in the quantification of accident sequences involving loss of containment heat removal.

3.4.4.2.3 RWCU

The RWCU system can remove a portion of the decay heat generated after an automatic shutdown from full power, but is assumed not to have the capacity to remove all of it. The RWCU system can also remove decay heat in a feed and bleed mode by directing reactor water to the main condenser. The RWCU system requires service water to operate as does the main condenser.

3.4.4.2.4 Containment Vent

The existing containment vent system is a system of last resort to prevent containment pressure from rising above the 56 psig design pressure. All other forms of DHR would need to have failed or be insufficient to remove decay heat before the vent would be required. Required support systems include service water and the plant air system. Use of the vent is initiated by actuating smaller 2" containment atmospheric system valves and progressively opening larger penetrations until containment pressure can be maintained below 56 psig. Venting is into the reactor building through ductwork which would burst open. EOPs instruct maintaining the containment below 56 psig as opposed to depressurizing the containment by means of venting, thereby limiting the rate of steam release to the reactor building. The fault tree for the Monticello containment vent, including both wetwell and drywell venting, estimates the system to have a failure rate of approximately 5E-3 per demand.

The planned hardpipe containment vent is intended to be independent of the various other decay heat removal systems. It will have no support systems other than electrical power and nitrogen. It will direct any steam release outside the reactor building, limiting the environmental conditions in the secondary containment.

Effects of Important Initiating Events

Initiating Event	Containment Vent Failure Probability
Transients/LOCA	5.4E-3
Loss of offsite power	1.0
Loss of instrument air	0.5
Loss of service water	1.5E-2

Operation of the containment vent is independent of whether the initiating event was a transient or a LOCA. It is manually actuated with important support systems including instrument air.

While the air compressors are powered from the essential buses, the containment vent would not be expected to be available for an extended loss of offsite power. The compressors are load shed on a DBA signal, loss of offsite power and high containment pressure. It is noted that action to reload the compressors onto the diesels could be taken given the time frame for decay heat to pressurize the containment. Also the potential for recovery of offsite power over the course of several days is very high.

As described for the main condenser, the reliability of the vent for loss of instrument air and loss of service water initiators involve recovery actions by the operator. The recovery of instrument air (approximately .5) is based on plant operating experience with one event in which the system was recovered prior to complete depressurization. The service water system indirectly supports the vent through cooling of the air compressors. Recovery of the compressors (approximately .01) can be performed by local alignment of the fire system as an alternate cooling source to the compressors.

Important Hardware Failures

Failure	Contribution to Failure Probability
Instrument air single failures	33%
Operator action to initiate venting	18%
Service water strainer	13%
Instrument panel failures	11%
Vent valve I&C failures	9%
Loss of offsite power	3%
Common cause failure of air compressors	2%

From the containment venting fault tree, dependencies on the instrument air system contribute to approximately 1/3 of the system unavailability. As noted in the discussion under the main condenser, many of these failures are capable of being corrected locally by the operator and include:

Manual initiation of air compressors on I&C failure
Isolation of instrument air flow diversion paths.

At 1E-3, the operator action to initiate the vent is estimated to contribute to approximately 18% of the failure rate. The failure rate is based on multiple indications of containment pressure available to the operator in the control room plus significant time to anticipate the need to vent (i.e., more than a day).

Service water strainer plugging over the course of the transient provides an estimated 13% of the vent failure rate. This was also discussed under the main condenser and can be corrected locally by the operator.

Instrument and control failures make up approximately 20% of the vent failure rate. This includes loss of Panel Y70 which supplies power to the vent valve solenoids and failure of a manual bypass switch which permits power to be supplied from 250V batteries or AC MCCs.

Offsite power and common cause failure of the air compressors each make up only a small part of the system reliability. Again, these failures are significant in that they are common with the main condenser.

Important Operator Actions

The action to initiate venting is important since the vent is a manually initiated system. Also actions to recover specific support systems, such as instrument air or service water, are important.

3.4.4.3 DHR Transient Analysis

Detailed analysis of plant response to a total loss of decay heat removal was performed to identify the timing and effects associated with containment heatup. The times at which various important setpoints contained within the EOPs were identified. Conditions that might affect the operability of key core cooling equipment were also noted. Explicit analysis of the environment in which core cooling equipment must operate was also performed and included considerations of the effects of venting the containment as well as failure of the containment on overpressure. Survival of DHR equipment following containment failure is discussed in Section 4.

Reactor and Containment Response

The rate at which containment heatup occurs under the assumption that no decay heat removal systems are effective is dependent on the source of makeup to the reactor. Two potential sources were considered in analyzing the Monticello containment response to a total loss of DHR; those external to the containment (such as the condensate storage tanks) and the suppression pool.

Monticello plant has two condensate storage tanks, each 40 feet in diameter and 25 feet in height. The 75,000 gallons of the water stored in these tanks is reserved for ECCS. Normal level in the tanks is maintained between 12 feet and 15 feet or more than 220,000 gallons. The water in these tanks can be supplemented by makeup from the demineralized water system, radwaste or even indirectly from the Mississippi River through operation of systems such as Service Water to the hotwell. The amount of energy required to heat up and boil just the water normally contained within the tanks is equivalent to nearly three days of decay heat generation.

Systems capable of making up to the reactor directly from the condensate storage tanks include:

Control Rod Drive
HPCI
RCIC
RHR
Core Spray

Indirectly, the condensate storage tanks also provide makeup to the reactor through the hotwell by supplying a source of water to the Condensate, Feedwater and Control Rod Drive systems.

As an alternate to the Condensate storage tanks, makeup to the reactor during a loss of DHR can be provided from the suppression pool. The suppression pool contains more than 500,000 gallons of water. Once makeup is accomplished steam from the reactor is directed back to the pool through the SRVs, maintaining a relatively constant pool inventory as a function of time. Systems capable of making up to the reactor from the suppression pool include:

HPCI
RCIC
RHR
Core Spray

Because the water gradually rises in temperature as decay heat is directed to the suppression pool, less heat capacity is available in a given volume of suppression pool water than from that contained in the condensate storage tanks. For this reason, containment pressurization occurs somewhat sooner if water is taken from the suppression pool than if it is drawn from the condensate storage tanks.

An analysis of a total loss of DHR was performed for Monticello assuming makeup from either the condensate storage tanks or the suppression pool. The containment pressure as a function of time for these sources of makeup are presented in Section 7. Section 7 also discusses operator actions contained within the EOPs and the impact of the rising containment pressure and temperature on key plant equipment as a function of time.

Given that the Condensate/Feedwater and CRD systems are the normal means of reactor inventory control during power operation, they are also the preferred means of providing makeup as the reactor is shutdown. For events in which Condensate/Feedwater are lost, HPCI and RCIC taking suction from the CSTs provide a backup means of high volume high pressure inventory control. Regardless of the source of early inventory makeup, as decay heat levels fall, the CRD system is naturally closest to makeup requirements in terms of flow capacity and is the easiest to use to maintain reactor levels in the ranges specified by the EOPs. With these systems as the most likely makeup sources, both early and late reactor inventory control is most likely to be from sources external to containment. Since, these systems can provide makeup at high RCS pressure, successful inventory control can be performed regardless of the status of reactor or containment pressure.

As noted above, late in a transient in which decay heat removal is unavailable, the CRD system will be the preferred source of makeup to the reactor. Only in the event that this system is unavailable is long term makeup from the suppression pool expected to be initiated. If HPCI and RCIC are used to provide makeup, they will be transferred to the pool from condensate storage on a high suppression pool level. The Monticello EOPs are currently based on Revision 3 of the BWR Owners Group EPGs, which require this transfer. On implementation of Revision 4, the suction of HPCI and RCIC will remain on the condensate storage tank unless a low tank level occurs. While RHR and core spray pumps have the capability of being aligned to the condensate storage tanks, the preferred lineup is from the suppression pool with transfer to condensate storage only as a backup to the pool.

Given that the CRD system must be unavailable in addition to the failures associated with all decay heat removal, only a few initiating events would be expected to reach containment failure pressure with two days. Such initiating events include only a loss of service water, which provides feedwater and CRD pump cooling and an extended loss of offsite power (which, if containment heat removal were unsuccessful, would result in a load shed of the CRD pumps on rising containment pressure). The total loss of service water is a very rare initiating event and the loss of offsite power would require that offsite power not be restored for nearly two days in addition to the failure of RHR and the vent.

If makeup from external sources were not available, it is important to note the response of the remaining key inventory makeup systems to the total loss of DHR event. Makeup from RCIC, HPCI, Core Spray and RHR will each be discussed.

RCIC and HPCI are turbine driven systems independent of all support systems with the exception of DC power. It is unlikely that initiating events leading directly to the loss of any of the principal DHR systems would affect the reliability of these systems. However, the operation of these systems is affected by containment temperature and pressure rise associated with a total loss of DHR. RCIC and HPCI pump lube oil is qualified for temperatures up to 140°F, which occurs approximately 5 hrs into a total loss of DHR. If the suction of pumps are aligned to the suppression pool, pump lubricant breakdown may occur, adversely affecting operation. Implementation of Revision 4 of the BWR EPGs will result in alignment of these systems to condensate storage whenever possible, eliminating this potential failure mode. Were RCIC to continue to operate beyond high suppression pool temperatures, a trip of the pump on turbine exhaust pressure occurs at 50 psig, which occurs approximately 21 hours following loss of DHR. The HPCI turbine exhaust pressure setpoint is 150 psig, which is well above containment pressures anticipated during a loss of DHR and would not limit the operation of the HPCI system.

As suppression pool temperature rises, the operator is instructed to depressurize the reactor to maintain the plant within the Heat Capacity Temperature Limit. Approaching this limit or the unavailability of high pressure systems will result in operator action to depressurize the reactor to the point that low pressure systems such as core spray or LPCI can operate. Since this is a loss of DHR event, LPCI may not be operable for either DHR or injection purposes. As suppression pool temperature rises, the operators would be expected to maintain operation of these pumps within NPSH requirements. If they are the only systems capable of reactor makeup, they are operated irrespective of NPSH conditions. NPSH should be of limited concern for makeup to the reactor during loss of DHR since makeup requirements are extremely low (decay heat levels) resulting either in limited velocity head in the pump suction or only periodic operation of the pump. Although the reactor is required to be depressurized during conditions associated with a loss of DHR, an extended period of containment pressurization can result in closure of the SRVs and repressurization of the reactor. The pneumatic supply to the SRVs is nitrogen or air supplied at approximately 100 psig. A differential pressure of 30 psig between the pneumatic supply and the containment atmosphere is necessary to maintain the SRV depressurization function. As a result, it is assumed that once containment pressure exceeds 70 psig 27 hours into a total loss of DHR, SRVs will close and reactor pressurization will occur. At this late point in the event, 9 hours are necessary to repressurize the reactor to the SRV setpoint of 1100 psig. Steam relief and continued pressurization of containment would resume at that point. Under the assumption that no high or low pressure injection systems were making up to the reactor, depletion of reactor inventory would also occur. Analysis of this type of event indicates that uncovering of the core would occur at about 42 hours, but that containment ultimate pressure would be exceeded prior to vessel penetration at 47 hours.

Given these analyses, the operating procedures, and Monticello operating practices for reactor makeup, it is concluded that events involving a total loss of DHR would most likely require nearly two days to result in containment failure. In the unlikely event that no makeup from external sources was unavailable, pressurization of the containment to the failure point would still require nearly two days.

Reactor Building Response

In addition to investigating the timing of events resulting from the loss of DHR, analysis of the environmental consequences associated with releasing steam from containment into the reactor building was performed. These analyses examined both the consequences of venting as well as containment failure on overpressure.

As noted above, venting is a last resort measure for assuring protection of the containment boundary implemented only when all other means of containment pressure control are ineffective. The current containment vent utilizes the containment atmosphere control system of vent and purge to provide containment overpressure protection. EOPs require jumpering of containment isolation signal to the containment vent valves, preferentially specifying the use of the vent valves from the wetwell airspace. The smallest of the vent lines (2") is suggested first, increasing the vent capacity to the 18" vent line if necessary.

The two wetwell vent valves are located on top of the torus. The vent line becomes duct work immediately downstream of the second vent valve and is assumed to rupture should venting with the 18" valves be initiated.

Initiation of containment venting is specified at the containment design pressure (56 psig). The EOPs instruct the operator to maintain the containment pressure below 56 psig as opposed to depressurizing the containment through the vent line. Maintaining the containment pressure at design minimizes releases from containments and limits the rate of steam release to the reactor building.

Analysis of the effects of venting on the environment in the reactor building was performed to establish assumptions to be made in the PRA. A nine node reactor building model was created with MAAP that incorporated significant reactor building volumes including the torus room, corner rooms containing ECCS equipment, the refueling deck, and other areas.

Injection of steam to the reactor building was assumed at decay heat rates, simulating maintenance of containment pressure below 56 psig in accordance with the EOPs. Releases were assumed to occur through ductwork located in the torus area. Peak temperatures in the corner rooms containing ECCS equipment remained less than 100°F. This temperature is well below the qualification temperature of this equipment and verified that continued operation of this equipment could be expected under venting operation. A detailed description of this analysis is provided in Section 7.

It was recognized that a portion of loss of DHR scenarios may occur without the benefit of the containment vent. In these instances, containment pressure would continue to rise beyond 56 psig to the point that containment failed on overpressure, if DHR systems were not recovered in the interim.

For these events, the containment failure size may be limited to what is necessary to relieve the steam being injected to the containment. In this instance, the rate of steam release to the reactor building would be at decay heat rates, similar to venting operation in accordance with the EOPs. The environment would be similar to or less severe than that expected from venting depending on the containment failure location.

If the containment failure size is greater than that required to relieve decay heat, the containment may depressurize as a result of the failure and the steam release to the reactor building could be large. An analysis of the reactor building environment assuming a containment rupture area as large as 20 ft^2 was performed to simulate this situation.

On reaching the containment ultimate pressure, estimated to be 103 psig, steam release to the reactor building from the drywell head or the wetwell airspace occurs and pressurization of the reactor building results. If the drywell head fails the steam release is near the refueling floor in the upper portion of the reactor building. The corrugated steel structure surrounding the refueling floor is capable of withstanding only about 0.5 psig. Release of steam to the environment through the refueling floor area would result. Using the same nine node reactor building model referenced in the venting analysis, peak temperatures in the lower portions of secondary containment such as the corner rooms do not exceed 150°F, again well within qualification temperatures. A similar analysis for the wetwell airspace produced a peak temperature of about 170°F in the corner rooms. Systems in these areas of the reactor building are likely to remain operable after containment failure. It is therefore possible to use these systems for continued injection to the reactor following total loss of DHR and containment failure.

3.4.4.4 Summary of Class 2 Sequence Results

The overall class 2 core damage frequency was very low, less than 1E-7/yr. The probability of containment failure due to loss of DHR alone was estimated to be 8.3E-6/year. Considering the hardpipe containment vent in a similar fashion to NUREG/CR-5225 reduces the loss of DHR containment failure probability to less than 1E-7/year.

Dominant cutsets of different top events were analyzed to obtain significant insights and to determine potential recommendations which could improve function/system reliability. The reasons for the low potential for core damage from DHR failure are as follows:

> Multiple and reliable systems exist to remove decay heat from the reactor and containment in the form of the main condenser, RHR, and containment vent.
>
> Venting or containment failure would have limited effects on the environment in which core cooling systems must operate.

Issues associated with DHR failure which do not contribute significantly to the reliability of DHR at Monticello include the following:

> Loss of offsite power is not a significant contributor to DHR loss because a significant time without power and RHR is required to reach containment failure. The ability to recover power within the first 24 hours of a loss of offsite power limits the significance of this initiator.
>
> Closure of SRVs on high containment pressure can result in loss of low pressure injection. The significance of this condition is limited to only a few sequences in which high pressure systems are also unavailable. The high pressure systems include both CRD and feedwater. On implementation of Rev 4 of the BWR Owners Group EPGs, the HPCI system will also remain available for long term makeup to the reactor from the CST.

Loss of feedwater does not imply a permanent loss of the main condenser as a heat sink. The feedwater pumps are motor driven and main condenser operation is not dependent on them. Further, Monticello operating experience suggests that feedwater has generally been recovered early following its loss.

The containment decay heat removal function (W) itself was dominated by the following types of failures:

1. Loss of the service water or air system. These systems affect both the operation of the main condenser and the containment vent.

2. Turbine trip events or manual shutdowns. These are the most frequent initiators experienced at Monticello.

From a component perspective, the following events had the largest contribution to overall loss of decay heat removal and would be the primary items investigated for potential corrective actions:

1. RHRSW loop corrective maintenance.

2. Failure of RHRSW heat exchanger discharge valves to open.

3. Common cause failure of torus cooling valves to open.

4. Air system receiver tank relief valves stuck open or pressure switch filter plugged.

5. Service water strainer plugging.

Many of the above items were previously addressed by the Monticello RHR Reliability Study. Others have been addressed by the IPE. In general, no modifications were apparent that would both be cost effective and result in a significant reduction in risk. Recommendations were made in NUREG/CR-4448 with regard to RHRSW corrective maintenance, RHRSW heat exchanger discharge valves manual operation, and torus cooling valve reliability. The IPE results contain recommendations regarding air receiver tank relief valves. Service water strainer plug procedures are already in the operating manuals. It appears that most, if not all, of the important event failures of DHR are already addressed or can be handled by currently prescribed operator recovery actions.

3.4.4.5 Proposed and/or Completed Recommendations

This section summarizes proposed and/or completed recommendations which have been identified as part of the DHR analysis at Monticello. In 1987, a reliability study was conducted on the RHR system at Monticello and various recommendations were considered in order to enhance the reliability of the RHR system. Since then, the Monticello IPE has contributed further insights regarding RHR system reliability. The IPE also provided further information on main condenser and containment vent reliability. A summary of important recommendations from both analyses follows:

1. The RHRSW heat exchanger discharge valves were replaced with an improved model. (New valves and Valtek operators)

2. The torus cooling valves were replaced with an improved model. (New valves and Rotork operators)

3. Procedures were developed to manually operate the RHRSW heat exchanger discharge valves, initiate torus cooling with a LPCI signal present, and prevent draining the RHR system.

3.4.4.6 Uncertainties

This section identifies uncertainties associated with the DHR study done as part of the Monticello IPE:

1. The role of the operator as a common cause failure mechanism for the various systems is an uncertainty. Detailed human error analysis produced a human error rate of 1.6E-5 for initiating torus cooling. If the operator has failed to initiate torus cooling, the conditional probability of failure to initiate other systems could be one. This could be the dominant failure mechanism of loss of DHR since other systems such as shutdown cooling, drywell sprays, and the containment vent are manually

initiated. The Monticello IPE assumed there was no common operator failure mode between the various DHR systems. This is because of the amount of time available to cope with loss of DHR and the multiple and diverse indications associated with DHR failure.

2. The actual containment failure pressure and failure location are uncertain. The Monticello IPE used the best available plant and generic information to determine what the failure pressure and location should be.

3. The consequences of containment failure are uncertain. Explicit analysis of the reactor building response was performed to establish the expected environment following venting or containment failure.

4. Equipment recovery over several days is likely. Use of this recovery factor contributed significantly to the conclusion that Monticello has adequate DHR capability, even without a hardpipe containment vent or considering injection systems after containment failure.

5. The Monticello IPE did not consider external events.

6. The impact of prolonging the event by using external water spray sources or RWCU was not considered.

3.4.4.7 Conclusions

The conclusions of the Monticello IPE with regard to DHR reliability were:

1. The current DHR reliability is high as a result of multiple means of DHR including the main condenser, shutdown cooling, suppression pool cooling, drywell and wetwell sprays, and the current containment vent.

2. There are approximately 32 hours available to recover failed DHR systems before containment design pressure is reached.

3. Condensate storage capacity is not a significant contributor to risk during a postulated loss of DHR event. This is due to its large capacity (several days of decay heat) and multiple means of make up to the tanks.

4. If all DHR systems were to fail and lead to containment failure, the potential for core damage is low. This is because steam addition to the reactor building at decay heat rates from venting or containment failure does not significantly affect the equipment in the reactor building. Systems or equipment, like condensate/feedwater, are also available outside the reactor building that would provide adequate makeup.

3.4.5 Internal Flooding Evaluation

Generic Letter 88-20 requires an internal flooding analysis as part of the IPE process. A number of internal flooding PRAs to date have been scoping analyses which have concluded that internal flooding will not lead to core damage. The Oconee 3 PRA, however, concluded flooding was a dominant contributor to the total core damage frequency and subsequently made plant modifications as a result. Other plants have experienced maintenance events which have resulted in flooding of equipment. All of these factors provide the basis for performing the Monticello internal flooding analysis.

The purpose of the internal flooding analysis was to determine potential vulnerabilities due to flooding from sources such as tank overfilling, hose and pipe ruptures, and pump seal leaks. The analysis used bounding, frequently conservative assumptions while still demonstrating a low potential for core damage. Attention was focused on the major flood sources in the plant which could affect multiple systems and propagate to other areas. Low capacity systems which had limited or no impact on multiple systems and flood initiators which were bounded by other flooding events were given less consideration.

The total core damage frequency for internal flooding events is conservatively estimated to be 6.8E-6/year. The study concludes there were only two flooding sequences having a frequency greater than 1E-6/yr. By themselves these account for more than half of the total flooding core damage frequency. The first flooding event involves a service water line break in the reactor building which was assumed to disable all high pressure injection systems; HPCI and RCIC due to flooding, and feedwater due to its dependence on service water. The second event involves a service water break or feedwater break in the 931' east turbine building area. High pressure injection systems are also assumed to be unavailable for this event; HPCI and RCIC due to loss of DC power supplies, feedwater due to the fact that it is the initiator or due to dependence on service water. No flooding initiators were identified that by themselves disabled core cooling.

The assumptions, methodology, mitigative factors, and results of the Monticello internal flooding IPE are discussed in this section.

3.4.5.1 Background

Considerable review of the Monticello plant design and operating procedures has been performed in the past with respect to the potential and effects of internal flooding.

The USAR HELB analysis discusses explicit flooding sources. The feedwater break as an example, may result in the discharge of water to various areas of the plant depending on the break location. The HELB analysis discussed the feedwater line break in the steam tunnel identifying that HPCI, RCIC, and CRD would be unavailable but that other equipment would remain operational. Other HELB sources such as, the main steam line break would be bounded by the feedwater line break.

SOER-85-05 issued by INPO required an assessment of the vulnerability of operating facilities to the loss of safe shutdown functions due to internal plant flooding. Analyses performed in response to this SOER identified no safe shutdown functions which could be compromised as a result of various flood initiators. The response to SOER 85-05 identified that maintenance events were the primary cause of flooding based on industry experience. An extensive part of the plant evaluation of the SOER involved reviewing procedures to see if they adequately addressed flooding and to identify the need for training in this area. This review found that procedural controls are in place assuring safety related isolations require independent verification. Emergency procedures were also reviewed as a part of the SOER and were found to be adequate in this respect. Administrative procedures specifically address flooding as a consideration in the plant modification process. The response to the SOER also involved training plant personnel on internal flooding and the need to ensure adequate isolation of equipment. A review of the training needs identified in the original SOER evaluation was performed in 1989 to verify that they were in place. The response to SOER 85-05 was a nonprobabilistic assessment of the potential for flooding and its consequences, the results of the PRA were found to be consistent with the previous evaluation.

Insights relevant to the potential causes and consequences of internal flooding that were provided as a part of the IDCOR IPE Methodology were also reviewed. The focus of this review was on flooding locations in the lower turbine building and the reactor building ECCS corner rooms. It was concluded that no special flooding vulnerabilities are expected at Monticello consistent with previous reviews.

3.4.5.2 Process

For the purpose of performing the Monticello IPE flooding analysis, flood zones within various buildings of the plant were determined. A flood zone was defined as an area in which systems and equipment included in the level 1 PRA were located that could be potentially affected by flooding from one or more sources. Table 3.4-2 presents the definition of the flood locations defined for the IPE flooding analysis as well as the flooding sources which could impact the operability of equipment located in each of the flood zones. Table 3.4-4 provides a summary of the systems and components that would be unavailable if flooding of a particular zone occurred. Internal flood initiating event frequencies were calculated by zone and were based on the combined frequency of each relevant flooding source's contribution to the zone.

Plant walkdowns were conducted for each zone and each potential flooding source to obtain various factors such as the length and diameter of water piping system, number of valves, tanks, room drains, room sumps, presence of equipment for systems considered in the PRA, propagation to and from other areas, door arrangement, curbs, and more. Generic pipe, valve, and tank rupture frequencies were used to estimate the initiating event frequency due to pipe break. Realizing there was a great deal of uncertainty in the pipe and valve rupture frequencies, a detailed analysis to account for every foot of pipe in the plant was unnecessary because important insights would be apparent regardless of the exact initiating event frequency. The primary objective of the walkdown was to determine potential flooding sources and equipment affected, with a secondary objective to account for the amount of equipment to be considered in the initiating event frequency. Table 3.4-2 contains the estimated frequency for passive component failures leading to the flooding of each of the flood zones. The total estimated frequency of flooding from passive hardware failures is approximately 8E-3/yr. In deriving the initiating event frequency only normally running systems, systems with auto start features, or systems that could drain by gravity were included. Systems which are normally in standby and do not have automatic start capability were not included as a potential flooding source. This eliminates systems such as EDG-ESW, RHRSW, core spray, RHR, HPCI, and RCIC. Further, low pressure piping is assumed to contribute to the potential for flooding at the same rate as pressurized piping. No credit for leak-before-break concepts was taken in this evaluation.

An estimate of the potential for maintenance or surveillance activities to contribute to flooding in each zone was also made. Maintenance induced floods were considered despite administrative controls to prevent flooding and the plant history of no internal floods. No particular maintenance or surveillance activities dominated the risk associated with internal flooding. The potential for the flooding initiator was estimated simply by assuming that there was a 50% chance that a flooding event should have occurred over the life of the plant so far (or 2.5E-2/yr) and that the difference between this frequency and that associated with passive equipment failure is that attributable to maintenance and surveillance activities (or 1.7E-2/yr). The frequency was distributed based on the relative amount of potential flooding sources located in each zone. This approach is considered to provide an upper bound to flood initiating events frequencies due to maintenance and surveillance activities.

For each flood zone for which drainage was credited, analyses were performed to estimate the flooding rate an area could tolerate considering factors such as floor drains, sump capacity, and door leakage. Calculations were performed to establish minimum pipe size which would supply water faster than drains could accommodate. Smaller pipes were screened out. For flood sources associated with a limited volume of water, such as that contained within specific tanks, derivation of room volumes was performed. These were used to determine what level the room would reach for a given volume of water. Where multiple systems contribute to the potential for flooding of a particular zone, the system with the highest flooding rate or most significant effects was considered in the analysis of the zone; for example, the lower capacity fire system was often considered to be bounded by the service water system. Once the low or limited capacity systems were identified, attention was focused on the higher capacity systems, particularly those which would affect multiple systems.

In performing the sequence quantification for each zone, the level 1 PRA results were modified to reflect the effects of flooding on the equipment contained within the zone. Results of this quantification are summarized in Table 3.4-3 categorized by the same accident classes defined in Section 3.1. Discussion of the dominant sequences is provided in Section 3.4.5.4.

3.4.5.3 Assumptions

A number of assumptions were made about the effects of each of the postulated floods. Assumptions regarding systems and equipment that may be disabled as a result of the flood are presented in Table 3.4-4. In this section, other assumptions regarding the magnitude and effects each of the flood areas are presented. Assumptions which were applied generically to all flood areas are presented first followed by those associated with each specific flooding zone. Conservative or bounding assumptions were made in many cases to minimize detailed evaluations of minor factors which may not provide significant insights regarding potential vulnerabilities associated with internal flooding.

Generic Assumptions

- Doors which open away from the flood are assumed to fail before the water rises two feet above the floor, allowing water to flow into and affect equipment in adjacent areas.

  Doors which open into the region of the flood are assumed to remain closed. Leakage around the door into adjacent areas and consideration of its effect were estimated.

- Pump run out or pump overcurrent breaker trips are conservatively assumed not to occur when a pipe break occurs.

- The operating crew was credited for isolating the flood source in some instances. These were when indication of a flood was available to be operators and corrective action to close valves or trip pumps was possible from the control room. This was assumed to be done no earlier than twenty minutes after the onset of the flood with a failure probability of 0.01. This method was used to eliminate a flooding source from consideration only if the same zone had another source which was unisolable and therefore encompassed any insights which might have been derived from the eliminated flood.

- Flooding via drains did not provide significant flow to adjacent areas in the reactor building. This assumption is reinforced by the fact that ECCS room drains are orificed to prevent backflow from the torus area.

- Motor control centers and electrical buses were assumed to fail at 6 inches of water.

- Cable insulation was assumed to be water resistant. A visual check was performed of several cable trays and no notable degradation of cable insulation was observed.

Flood Zone 1 (Reactor Building - Torus Ring Header)

- A large failure of the torus significantly below the pool surface could render the vapor suppression function inoperable if the SRV tailpipes became uncovered. The dominant contributor to the torus break initiating event was the torus ring header. The potential for a loss of the integrity of the torus itself is considered to be extremely small. Furthermore, the EOPs cover this situation by providing instructions to depressurize the reactor on reaching the Suppression Pool Level Limit. A simple calculation of the flow rate due to gravity through the ring header demonstrates that at least 10 minutes would be available for the operator to depressurize the reactor vessel for a torus ring header break. Credit was taken for this operator action within a 10 minute time frame as directed by EOPs.

- A torus ring header break is also assumed to cause RHR, core spray, HPCI, and RCIC to be inoperable due to flooding.

Flood Zone 2 - CSW Piping in the Reactor Building

- A flood of the torus area due to a break in the CST piping at the 896' level is assumed to fail HPCI and RCIC systems.

- In addition to loss of HPCI, RCIC and CRD, Feedwater and Condensate are assumed to be unavailable once the CST is drained. No credit for service water makeup to the hotwell was taken in this analysis.

Flood Zone 4 - Service Water Break in the Reactor Building

- Flooding in the reactor building cascades down all floors, eventually affecting the CRD, HPCI, and RCIC systems but not necessarily the RHR and core spray systems. The RHR rooms have 6 inch curbs outside the room door. The RHR room doors open into the reactor building thereby limiting leakage into the rooms. Should water leak through the door, the RHR and core spray pumps are not located directly under the stairs. Room sumps are also present. Furthermore, there are other more likely places for water to go at the 935' elevation besides the RHR rooms; for example, the CRD room, the door of the CRD room once the room fills with water, and the doors leading to the RCIC room which open away from the flood.

- Besides CRD, HPCI and RCIC, Feedwater is assumed to be unavailable due to loss of Service Water cooling. Condensate has been demonstrated to run without Services Water cooling and is expected to remain available.

Flood Zone 5 (7) - Service Water Piping Failure in SE (SW) RHR Rooms

- Service water lines above 1" contribute to the flooding of this room and lead to the loss of one train of core spray and RHR. Corner room sumps are capable of preventing flooding for piping failures less than this size.

- Feedwater and CRD operation are assumed to be affected by the loss of service water. Flow diversion is assumed to be sufficiently great to prevent seal cooling.

Flood Zone 6 (8) - CSW Piping Failure in the SE (SW) RHR Rooms

- One train of RHR and core spray are assumed to be affected due to flooding in the respective room. The Feedwater, Condensate and CRD systems are considered unavailable once the CSTs are drained.

Flood Zone 9 - Service Water Failure in the Turbine Building Leading to Flooding of the 911' Elevation

- Flooding of the 911' elevation in the turbine building is assumed to disable the Division 1 4KV room, air compressors, feedwater pumps, condensate pumps, main condenser, and MCC 31 which results in loss of 2R transformer.

Flood Zone 10 - West Diesel Generator Room

- Flooding effects were considered in the west diesel room. The west room was considered bounding over the east room since it was assumed to fail both diesel generators. The door between the diesel generators opens from the west room to the east room. The magnitude of flooding is limited as there are doors leading to the outside and there is a curb between the rooms which would minimize propagation of the flood. The fire system was the only possible flood source in the diesel room. EDG-ESW is a standby system with no auto start capabilities and therefore was not considered as a flood source. The flood was conservatively assumed to propagate to the other diesel room and then to the 911' elevation in the turbine building.

Flood Zone 11 - Fire System Break in the West Turbine Building

- A flood in the turbine building elevation 931' West is assumed to affect the Division II 4KV room but will not affect the 911' elevation. The basis for this assumption is the way the doors are arranged. A flood in this area will cause a loss of MCC 21 and later MCC 31 which in effect cause a loss of feedwater and condensate due to failure of condensate demin valves. The fire system was identified as the significant flooding source in this area.

- When repair and recovery was considered, credit for one CRD pump was taken for the fire system breaks in the turbine building. An immediate loss of feedwater and/or RCIC at the beginning of the flood initiator was determined to be highly unlikely. If a scram occurs and water is made up to the vessel by other systems for an estimated half hour to 45 minutes, core damage will not result if a CRD pump is used in the scram mode. In addition, this flood was assumed to fail HPCI because of a loss of the battery chargers. In fact, HPCI would operate for at least four hours without battery charging in which case CRD would be more than adequate to provide makeup.

Flood Zone 12 - Service Water Piping in the East 931′ Elevation of the Turbine Building

- A flood in the east 931′ elevation of the turbine building was assumed to cause a loss of MCC 42, MCC 43, and the battery systems present at main access control (MAC). The door to MAC opens away from the flood and is assumed to fail open for a flood in this area. In fact, the door to the machine shop will probably open first because it is larger and opens away from the flood. The two 125V battery trains and one of the 250V batteries are located in the MAC area.

- A pipe break in the service water system would not effect ESW or EDG-ESW because of the presence of check valves preventing backflow to the service water system.

Miscellaneous Flood Zones

- Intake Structure

The circulating water pumps have automatic trip devices in the circ water bay. The pump will trip if a break occurs in the circ water bay. This break was not considered significant because of this additional trip logic feature.

The circulating water pumps also have automatic trip devices in the condenser pit. The circulating water pumps will trip if a break occurs in the condenser room. Trip logic would have to fail for this break to have a significant impact.

A flood from other systems in the intake structure will not result in the trip of anything other than the circ water pumps and is assumed to propagate to the 911' elevation in the turbine building. The door to the intake tunnel opens out of the room and is in a lower portion of the room than the pumps in the intake structure. These pumps are elevated on pedestals in the room and are assumed to remain operational given a flood in the intake structure.

- CRD Room

CRD pipe break in the CRD room will cause the CRD pumps to trip. Since the doors open into the room, only controlled flooding at relatively low flow rates will occur around the doors into other areas. CRD pipe break was considered a passive failure of the CRD system having limited impact on its reliability as compared to random failures.

- Containment

LOCA analyses was assumed to bound all flooding which could occur within the primary containment.

- Radwaste

Flooding in the Radwaste Building was not evaluated because there is no vital equipment located there which was considered in the IPE. Flooding was assumed not to propagate to the reactor building because of the airlock door arrangement.

- Recombiner Room

A condensate line break in the recombiner building was not considered because it would have the same affect as a loss of the condensate system. Pipe break was considered a passive pipe failure of the condensate system alone and was not considered significant as compared to random failures.

- Recirculation Pump MG Set Room

The Recirculation MG room was not considered to be a significant flood zone. Flooding from Fire piping and Service Water piping in this room should be contained within the room as the double doors to the reactor building open into the room. There is an elevated door which opens into the Administration building which is most likely where most of the water would go. There is a chance of flooding in the administration building at this point, but this was considered remote since there are usually personnel in the area including the Shift Supervisor's office one floor below to detect water flow. Two holes at the base of the north wall were observed and were assumed not to cause any other systems in the IPE to fail if water flowed through them to the turbine operating floor.

- Administration Building

Flooding in the administration building was considered to result in limited risk because any effects resulting from a water line break would be bounded by other flooding sources such as service water line breaks in the turbine building. In addition, there are usually personnel somewhere in the vicinity who would most likely provide early detection and isolation.

- Turbine Building Operating Floor

  Fire piping is present in this area but is not of sufficient capacity to flood equipment other than that considered at lower elevations. That equipment is bounded by flood zones 9, 11, and 12.

  The feedwater lines in this area are also not considered to result in significant risk because the HELB study concluded all water would leak back into the condenser area.

3.4.5.4 Transient Analysis

In quantifying each of the flood initiated accident sequences, examination of reactor response established key timing for reactor conditions and operator response. Each of the accident sequences were assigned to an accident class:

1A - Flood initiated core damage event with reactor at high pressure

1D - Flood initiated core damage event with reactor at low pressure

3 - Flood induced LOCA due to SRVs failing to open.

No new transient analyses were considered necessary to quantify flooding initiators. The timing of each of these accident classes was essentially the same as their counterparts in the internal events PRA.

3.4.5.5 Results

The total CDF for internal floods was estimated to be less than 7E-6/yr. Table 3.4-3 contains a detailed breakdown of the flood initiators by accident class.

The overall conclusions of this evaluation are that flood initiators do not contribute significantly to the risk of core damage. No internal flood events could be identified that do not also require additional, non-flood related random failures for inadequate core cooling to occur.

Four sequences essentially dominate the results, making up over 80% of the internal flooding CDF. These are discussed below.

Class 1A - Flood Initiated Core Damage at High Pressure

Sequence F4QUX - 2.1E-6/yr

Flood Zone 4 involves a service water line break in the reactor building. Systems assumed to be flooded for this initiator include CRD, HPCI and RCIC. The feedwater system is assumed to be lost eventually as a result of flow diversion from the service water system and failure of pump seal cooling. Having lost all high pressure injection systems, emergency depressurization of the reactor is required to enable low pressure systems. The condensate system remains as an injection system from outside the reactor building. RHR and Core Spray remain as injection systems because the flood can not propagate into the corner rooms in which this equipment is located.

The dominant cutsets in this sequence consist of the flood initiator and failure to depressurize the reactor within 1/2 hour of reactor trip.

F12QUX - 1.9E-6/yr

This flood initiator consists of a service water line or feedwater line break in the east 931' elevation of the reactor building. Equipment affected by this flood includes MCC42 and MCC43, which provide power to one division of core spray and LPCI valves, one 250V DC battery, supplying control power to RCIC and both 125V DC battery divisions supplying power to ADS. The Division II 125V DC also supplies the HPCI control system, therefore HPCI is made inoperable.

The remaining Division II 250V DC will supply power to four of the eight SRV solenoids from the alternate shutdown panel. Condensate is available for makeup as a low pressure injection system. One train of LPCI and core spray are also available, with local breaker operation.

F9QUX - 9E-7/yr

This flood involves a service water line break at the 911' elevation of the turbine building.

One division of AC power is assumed to be lost for this flood initiator. Feedwater and CRD are assumed to fail eventually due to their dependence on service water for cooling. RCIC is assumed to fail in the long term as a result of battery depletion.

HPCI, ADS, and one train of LPCI and core spray all remain available for this event. Dominant cutsets for this initiator involve failure to depressurize the reactor.

<u>Class 1D - Flood Initiated Core Damage at Low Pressure</u>

F9QUV - 9E-7/yr

This flood initiator is the same as that discussed above for F9QUX except that depressurization is successful. Operation of low pressure systems is unsuccessful. As noted above, one train of LPCI and one train of core spray are available for core cooling. Random independent failures, instrument and control, and battery failure contribute to the sequence cutsets.

Table 3.4-1

Summary of CDF Results

ACCIDENT CLASS	DESCRIPTION	CORE DAMAGE FREQUENCY
1A	Loss of Coolant Make-up, High Pressure Core Melt.	3.0E-6/yr
1B	Loss of All AC Power.	1.2E-5/yr
1D	Loss of Coolant Make-up, Low Pressure Core Melt.	3.6E-7/yr
2	Loss of Decay Heat Removal.	7.1E-8/yr
3A	Failure of SRVs to Open Causes RPV Rupture.	1.1E-7/yr
3B	LOCA, High Pressure Core Melt.	4.7E-7/yr
3C	LOCA, Low Pressure Core Melt.	3.9E-7/yr
3D	LOCA, Failure of Vapor Suppression.	2.9E-7/yr
4	ATWS.	2.5E-6/yr
5	Unisolated LOCA Outside Containment.	3.2E-10/yr
--	TOTAL (internal events)	1.92E-5/yr
6	Core Damaging Accidents Initiated by Internal Floods.	6.8E-6/yr
--	TOTAL (with flooding)	2.60E-5/yr

Table 3.4-2

Flood Initiator Area Definition

FLOOD DESIGNER	AREA	FLOOD FREQUENCY DUE TO PASSIVE FAILURES
F1	Torus Ring Header	4.8E-4/yr
F2	Condensate Service Water Break at Reactor Building 896'	5.1E-4/yr
F4	Service Water Break in Reactor Building > 896'	6.5E-4/yr
F5	Service Water Break in SE RHR Room	1.9E-3/yr
F6	Condensate Service Water Suction Line in SE RHR Room	2.1E-5/yr
F7	Service Water Break in SW RHR Room	1.7E-3/yr
F8	Condensate Service Water Suction Line in SW RHR Room	1.6E-4/yr
F9	Service Water Break in TB 911'	2.3E-3/yr
F10	W Diesel Generator Room	1.4E-4/yr
F11	Fire in TB 931' W	3.7E-5/yr
F12	Service Water Break in TB 931' East	2.2E-4/yr

Table 3.4-3

Flood Sequence Frequency

SEQUENCE	FREQUENCY
Class 1A	
F1QUX	1.8E-9
F2QUX	8.2E-8
F4QUX	2.1E-6
F5QUX	8.0E-8
F6QUX	1.7E-10
FYQUX	7.1E-8
F8QUX	1.6E-9
F9QUX	8.6E-7
F10QUX	6.8E-8
F11QUX	5.7E-8
F12QUX	1.9E-6
TOTAL	5.3E-6
Class 1D	
F1QUV	8.6E-8
F2QUV	1.2E-9
F4QUV	1.4E-9
F5QUV	1.3E-9
F6QUV	4.6E-9
FYQUV	1.2E-9
F8QUV	9.0E-11
F9QUV	9.1E-7
F10QUV	6.7E-8
F11QUV	8.3E-11
F12QUV	2.4E-7
SUB TOTAL	1.3E-6

Table 3.4-3 (continued)

Flood Sequence Frequency

SEQUENCE	FREQUENCY
Class ID with SORV	
F1PV	1.2E-12
F2PV	6.5E-12
F4PV	2.5E-11
F5PV	1.7E-10
F6PV	/OMEGA
F7PV	1.5E-10
F8PV	2.5E-11
F9PV	4.0E-8
F10PV	3.9E-9
F11PV	2.4E-10
F12PV	3.6E-9
SUB TOTAL	4.8E-8
LOCA from SRV Failure to Open	
FLOCA	1.1E-7
Total of All Flood Sequences	6.8E-6

Table 3.4-4

Effects of Flood Initiators

FLOOD	ZONE	FREQUENCY	AFFECTED SYSTEMS	REMAINING SYSTEMS	FLOOD RELATED OP ACTIONS CREDITED
F1	Torus Ring Header	4.8E-4/yr	HPCI, RCIC, RHR, CS, ADS (after torus drain)	FW/COND, RHRSW, CRD	Manual depressurization on torus level
F2	CSW @ RB 896	1.7E-3	HPCI, RCIC, FW/COND	RHR, CS	
F4	SW @ RB 896	2.1E-3	HPCI, RCIC, FW, CRD	COND, RHR, CS	
F5	SW @ SE RHR	6.3E-3	CRD, FW, 1/2 RHR, 1/2 CS	1/2 RHR, 1/2 CS, HPCI, RCIC	
F6	CSW @ SE RHR	6.9E-5	CRD 1/2, RHR, FW, 1/2 CS	1/2 RHR, 1/2 CS, HPCI, RCIC	
F7	SW @ SW RHR	5.6E-3	1/2 CS, FW, CRD, 1/2 RHR	RCIC, HPCI, 1/2 RHR, 1/2 CS	
F8	CSW @ SW RHR	5.3E-4	1/2 RHR, FW, 1/2 CS, CRD	HPCI, RCIC, 1/2 RHR, 1/2 CS	
F9	SW @ TB 911	7.6E-3	FW, COND, CRD, 1/2 CS, 1/2 LPCI	HPCI, RCIC (4 hours), 1/2 CS, 1/2 LPCI (Manual)	Locally open LPCI Injection Valve
F10	W DG Room	4.6E-4	FW, COND, 1/2 CS, 1/2 LPCI	HPCI, RCIC, 1/2 CS, 1/2 LPCI	
F11	Fire @ TB 931 W	3.7E-5	1/2 CS, 1/2 LPCI, RCIC, HPCI (for 4 hours), 1/2 CRD	FW, COND, 1/2 CS, 1/2 LPCI	
F12	SW @ TB 931 E	2.2E-4	FW, 1/2 CS (valves), 1/2 LPCI, RCIC, HPCI, CRD	COND, 1/2 CS, 1/2 RHR	Operation of SRVs from the alternate shutdown panel

4. BACK END ANALYSIS

The purpose of the back end analysis was to obtain an understanding and appreciation of potential containment challenges, the impact of phenomena and plant features on prevention and mitigation of the challenges in assuring containment integrity and limiting offsite releases, and the role of operator actions in dealing with containment challenges. This included possible recommendations for training, procedure revisions and modifications. A secondary objective was to allow for the evolution of an accident management program.

4.1 Plant Data and Plant Description

This section describes component, system, and structure data important in assessing severe accident progression. A discussion of equipment whose operability is desired in harsh environments is included as well as a description of the containment geometry.

4.1.1 Monticello Containment

Monticello has a Mark I containment as shown in Figure 4.1-1. The drywell is a steel pressure vessel enclosed in reinforced concrete. The drywell has a removable head which is held in place by bolts and sealed with a double gasket. The gaskets are made of a silicone rubber compound. One double door personnel air lock and two bolted hatches are provided for drywell access. Drywell internal design conditions are 56 psig and 281 degrees. Drywell external design conditions are 2 psig and 281 degrees. The free air volume of the drywell is about 134,000 cubic feet.

The suppression chamber is a steel pressure vessel in the shape of a torus. The torus is supported by the concrete foundation slab in the reactor building. The normal water volume is approximately 70,000 cubic feet, and the normal air volume is about 106,000 cubic feet.

Eight vent pipes connect the drywell to the suppression chamber vent header. The vent header is shaped like a torus and is contained within the torus air space. Projecting from the header are 96 downcomer pipes which terminate approximately

3 feet below the water surface of the torus. Eight 18 inch vacuum breakers are provided to equalize the pressure between the torus air space and the drywell. The torus also has pipes from the SRVs, HPCI steam exhaust line, and RCIC steam exhaust line terminating below the water line.

Containment nitrogen purge and vent lines are connected to the drywell and torus air spaces. The torus purge line has two reactor building to torus vacuum breaker check valves with air operated butterfly valves in series to protect the containment from exceeding its 2 psig external pressure limit.

4.1.2 Containment Systems

The only system included in the level 2 analysis that was not included in the level 1 analysis was containment spray, either using RHR pumps or division I RHRSW pumps. The system was used only as an injection source to the containment for debris cooling when RHRSW or RHR was unavailable as a heat sink. The fault tree quantification process was identical to the level 1 process, described in section 3.3.7.

Slight modifications were made to the SRV system by requiring only one valve instead of two and allowing 100 minutes for depressurization. The basis for these changes is the longer time available to depressurize prior to vessel penetration compared to the time available to depressurize to prevent core damage. The longer time permits a lower human error probability and allows a single SRV to successfully depressurize the reactor at lower decay power levels.

Similarly, the human error failure rate to manually align RHRSW to the RHR system is lower for level 2 than for level 1. In level 1, the need to align RHRSW within 30 minutes to prevent core damage created a large failure probability of 0.75. If core damage occurs, the goal becomes avoiding containment failure. Approximately 90 additional minutes are available after core damage, so the probability of not aligning RHRSW to RHR becomes 0.25.

If a system or component is disabled by the progression of events in level 1, then it is generally not credited in level 2. Exceptions are recovery of failed systems, such as offsite power, or recovery of human errors as described for RHRSW and ADS, above. Systems expected to be available after core damage are:

1. Core Spray. Core spray equipment was assumed to be available for recovery in-vessel or debris cooling. Injection from core spray could either be into the vessel or through a hole in the bottom of the vessel onto the drywell floor. One core spray pump was found to be sufficient. A LOCA outside containment or flooding in a corner room was assumed to fail the equipment in that room. Equipment in the other corner room was unaffected and therefore still available. If AC power could be recovered in a station blackout sequence prior to containment failure, then core spray was assumed to be available.

2. RHR (Containment sprays, LPCI, shutdown cooling, and torus cooling). Assumptions regarding RHR availability were similar to those for core spray. A LOCA outside containment in either loop of RHR was assumed to render the entire RHR system inoperable. Containment sprays were credited as a debris cooling system without RHRSW available. Torus cooling was credited as a decay heat removal system with the same success criteria as the level 1 analysis, i.e. one RHR pump and one RHRSW pump in the same loop were required. LPCI was used as an injection system similar to core spray. One RHR pump was found to be sufficient. Shutdown cooling was available as a DHR system for sequences with an intact reactor vessel at low pressure.

3. HPCI. HPCI was assumed as an injection system for ATWS until containment failure. HPCI was considered operable during ATWS whether the suction source was the torus or the condensate storage tank irrespective of environmental conditions. In this scenario, some environmental conditions will cause a loss of HPCI prior to containment failure. The loss of HPCI will tend to shut down the reactor and limit heat addition to the containment. Therefore, assuming HPCI runs right up to containment failure is a conservative assumption.

   The system was exposed to high torus water temperature and was not credited during loss of DHR (Class 2) sequences.

   HPCI was assumed to be unavailable following containment failure because local temperature switches could isolate the HPCI steam line. These scenarios included: ATWS, containment failure due to loss of DHR, and LOCA outside containment.

4. Systems in the turbine building. Feedwater, condensate, service air system, service water, RHRSW, RHRSW crosstie to RHR, and the AC power systems were potentially available for any sequence post core damage unless these systems failed in order to lead to core damage. The division II AC power system was assumed to be unavailable for a LOCA in the main steam tunnel, since there is a unobstructed passage between the turbine operating floor where steam could be released, and the division II switchgear room. This assumption is probably conservative. Feedwater and condensate were assumed to be ineffective for in-vessel recovery following a large LOCA sequence because of the location of the feedwater sparger outside the core shroud.

5. SRVs. RPV depressurization through SRVs was assumed to be potentially available for any sequence unless hardware failures in this system led to core damage. No sequences were identified in which the containment conditions exceeded the EQ limits of the valves before vessel penetration. SRVs are not required after vessel penetration. Loss of the depressurization function was assumed for sequences in which containment pressure rose well above 60 psig. This is because the pneumatic supply pressure to the SRV actuators is insufficient to overcome elevated containment pressures in excess of design. SRVs were assumed to resume functioning after containment failure caused by containment overpressure.

6. Containment vent. The containment vent was credited as a potential heat removal system during core damage sequences with the debris either in-vessel or ex-vessel. The valves were assumed to remain operational after venting at high containment pressure.

The reactor building was assumed to be inaccessible for at least several days following core damage. Recovery actions in the reactor building were considered impossible following core damage.

4.1.3 Systems Credited After Containment Failure

The Core Spray and RHR systems were generally credited with continued operability following containment failure. The availability of these systems is supported by MAAP analysis which shows that the various containment failure scenarios do not produce conditions which exceed the environmentally qualified limits for the pump motors. Other equipment of the core spray and RHR systems is either already in its actuated position or moves to its actuated position shortly after containment failure. It is assumed that any harsh environmental conditions would not degrade these components during the short time between containment failure and actuation. Table 4.1-1 lists expected conditions and environmentally qualified limits for the RHR and core spray motors.

The suction and injection lines of these systems are expected to remain intact upon containment failure because the anticipated containment failure location is either the drywell head or the expansion bellows of the drywell-to-wetwell vent lines. Both of these locations are physically distant from the suction and injection lines for core spray and RHR. Moreover, these lines are designed to withstand dynamic loading from vessel blowdown and seismic forces, so any reaction loads introduced by containment failure should not harm these lines.

The manufacturer's stated NPSH requirements for RHR and core spray pumps may not be met following containment failure. While being below NPSH requirements may cause some cavitation or degradation of pump performance, these pumps are expected to continue pumping an adequate water flowrate to meet decay heat requirements.

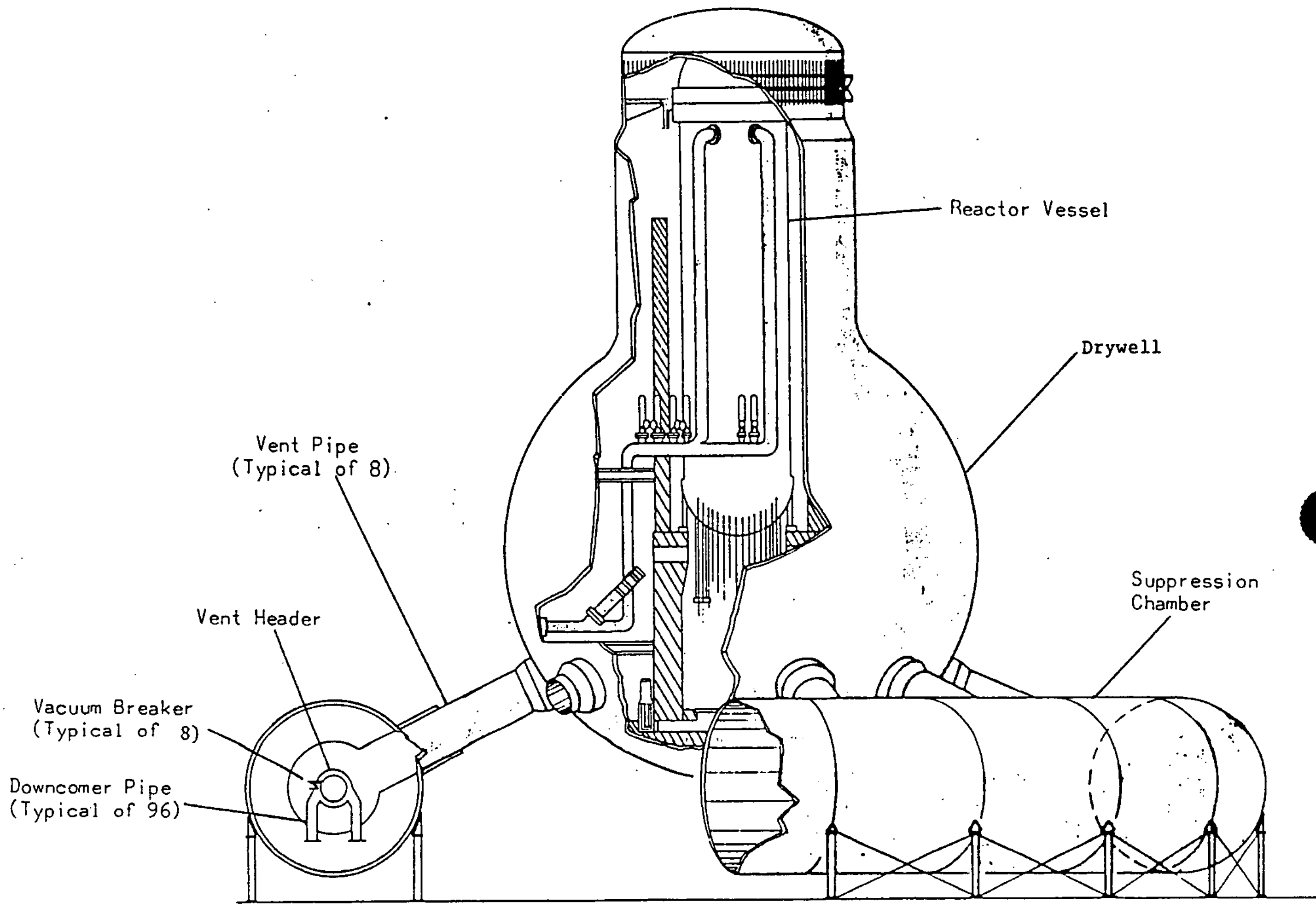
Table 4.1-1
EQUIPMENT SURVIVAL AFTER CONTAINMENT FAILURE

Containment Failure Mechanism	Parameter	Pump Motor * Environmentally Qualified Limit	RHR Room MAAP Analysis Peak Sustained Value
ATWS	Temperature	212° F	200° F
	Pressure	80.7 psia	≤16 psia
	Humidity	100%	≤100%
INTERMITTENT CONTAINMENT VENT	Temperature	212° F	90.4° F
	Pressure	80.7 psia	≤16 psia
	Humidity	100%	≤100%
LOSS OF DHR	Temperature	212° F	143° F
	Pressure	80.7 psia	≤16 psia
	Humidity	100%	≤100%
LOCA OUTSIDE CONTAINMENT**	Temperature	212° F	107° F
	Pressure	80.7 psia	≤16 psia
	Humidity	100%	≤100%
CONTAINMENT OVERPRESSURE	Temperature	212° F	170° F
	Pressure	80.7 psia	≤16 psia
	Humidity	100%	≤100%

* Reference: NEQ Central File pp C.12B.3.1 - C.12B.3.4.

** Modeled as steam line rupture directly into the reactor building at ground level.

Figure 4.1-1
PRIMARY CONTAINMENT



4.2 Plant Models and Methods for Physical Processes

This section contains documentation of all analytical models used in the accident progression analysis. General assumptions used in the modelling of phenomenology are described.

4.2.1 Plant Models

MAAP was the primary code used for the back end analysis. Plant data, including reactor building parameters, were used whenever possible as input to the code to allow analysis of sequences, source term, and building conditions. The results of other analyses were considered in developing the positions for the various containment failure modes.

4.2.2 Assumptions

This section contains important assumptions used in the level 2 analysis. The assumptions are:

1. The model accounts for the use of various means to makeup water to the hotwell during accident sequences where extended coolant makeup to the reactor is needed. The condensate storage tanks hold enough water to meet the decay heat requirements for 24 hours, so the plant staff would have a day or so to open the service water cross-tie or use other means to fill the hotwell with water. A failure rate of 0.10 was used for this activity based on engineering judgement.

2. The Core Spray and RHR systems were generally credited with continued operability following containment failure. The availability of these systems is supported by MAAP analysis which shows that the various containment failure scenarios do not produce conditions which exceed the environmentally qualified limits for the pump motors. Other equipment of the core spray and RHR systems is either already in its actuated position or moves to its actuated position shortly after containment failure. It is assumed that any harsh environmental conditions would not degrade these components during the short time between containment failure and actuation. See Section 4.1.3 for more details.

3. The EOPs currently prohibit spraying the containment under the conditions expected before vessel failure in Accident Classes 1A, 1B, and 1D (i.e., conditions outside the DSIL). Therefore it was assumed that should these sequences proceed to the point of vessel penetration, debris would exit the vessel into a dry pedestal.

4. It was assumed that hydrogen combustion was highly likely whenever the core is damaged while the containment in not inerted, due to the presence of numerous electrical components. Furthermore, it is assumed that hydrogen combustion always produces a release of radioactivity. No credit was taken for frequent periodic burning of small amounts of combustible gases to limit the pressure rise.

5. No credit was taken for operator actions from the ASDS panel during a complete failure of 125 VDC. This action could be important for sequences involving total loss of 125 VDC or sequences initiated by an internal flood in zone 12 (F12) which is assumed to flood both 125 volt battery rooms.

6. It was assumed that service water was not required for operation of the Condensate System. The first year of operation at Monticello was successfully conducted without Service Water cooling of the Condensate pumps. However, service water failure induces failure of the instrument air compressors which means the normal hotwell makeup valves will fail closed. The failed valves can be bypassed readily by opening a manual valve in the turbine building.

7. Containment flooding was assumed to be attempted for all core damage events that lead to vessel penetration. This action was further assumed to lead to drywell venting as non-condensible gases are compressed into the upper drywell. If the drywell vent was unavailable, the operator is assumed to terminate containment flooding in accordance with the EOPs before reaching the containment ultimate pressure.

8. Liner meltthrough is not expected to occur at Monticello. Best estimate calculations show that insufficient core debris exits the reactor vessel to overflow the containment sumps and contact the drywell shell.

9. If no debris cooling occurred, the containment was assumed to eventually fail. Currently, the most likely failure modes are considered to be overpressurization or movement of the reactor vessel and steamlines due to weakening of the pedestal, causing a tear in a containment penetration.

10. Drywell failure was assumed to occur 50% of the time that the containment ultimate pressure was reached. The other 50% of the time is wetwell airspace failure.

11. MAAP computer runs and work in the control room simulator showed that the vapor suppression function was not needed under most conditions. In spite of this, vapor suppression was included in the level 1 and 2 analyses.

12. The southeast RHR room would not exceed conditions where the RHR and Core Spray pumps would not operate in the event of containment failure, ATWS, or LOCA outside containment not located in this room.

13. Drywell spray valves would not be disabled by the environmental conditions for any sequence.

14. LPCI injection valves would not be disabled by the environmental conditions for any sequence.

15. Core spray injection valves would not be disabled by the environmental conditions for any sequence.

16. Except as noted earlier for pipe breaks in the steam chase, the environmental conditions in the turbine building would not disable important equipment for any sequence.

17. After core damage, access to the reactor building would not be possible. Access to the turbine building was still possible for sequences with core damage or a release in the reactor building.

18. After 56 psig is reached in the primary containment, personnel would not be allowed in the reactor building.

19. The containment vent valves remain operational after containment venting.

20. Loss of room cooling would not affect the operation of RHR, Core Spray, RCIC, and HPCI. Tests conducted by the Monticello plant staff support this assumption.

21. The containment failure pressure is assumed to have no dependence on temperature.

22. It is assumed that initiating ADS within 100 minutes of the initiating event will allow low pressure systems to arrest core damage prior to vessel penetration.

4.3 Bins and Damage States

This section covers the methodology and results of binning sequences from the front and analysis for evaluation in the back end analysis, and binning to the results of the level 2 sequence quantification. The bins are organized by factors such as timing, reactor conditions and containment conditions. A discussion of the binning process is presented for the following level 1 and level 2 results:

- Accident Classes
- Containment Failure Modes
- Release Modes

4.3.1 Front-to-Back End Interfaces

As noted in Section 3.1.5, five major classes of accidents were used to categorize the level 1 accident sequence results. These categories were in turn subdivided into subclasses. The accident classes are presented in Table 4.3-1. The delineation of accident classes and subclasses is dependent on the functional failures that occur in the level 1 sequences that are assumed to lead to core damage. These functional categories are convenient in characterizing the level 1 results and identifying the plant design and operating characteristics that drive the potential for core damage.

These same categories are also useful in transferring the results of the level 1 PRA to the level 2 containment event trees. This transfer is accomplished simply by using the cutsets from the level 1 sequences as the initiating events in the containment event trees. Fault tree linking allows dependencies and failures important to the level 1 results to be carried directly into the level 2 sequence analysis. Fault trees developed for the level 2 event tree headings include frontline and support systems similar to the level 1 allowing for these dependencies to be counted in the level 2 sequence analysis. Because of the fault tree linking approach, an explicit check list accounting for the transfer of dependencies between the level 1 and level 2 Sequences is not necessary. Nevertheless, Table 4.3-2 is provided to show where these dependencies occur.

4.3.2 Damage States

Damage states are identified for each sequence of the level 2 CETs. A four letter code was used to identify the damage state, A BB C. These codes are defined in Table 4.3-3.

The first letter (A) defines the state of the reactor at the time of vessel penetration, whether the event was recovered within the vessel or vessel penetration was assumed to occur at either high or low pressure.

The second two letters (BB) are used to define the status of the containment at the end of each of the containment event tree sequences. Whether the containment is intact or failed as a result of any of a number of severe accident phenomena is identified. The containment failure modes identified by this two letter code are patterned after the phenomenological challenges identified in NUREG-2300 and discussed in Section 4.4. In this manner the CET sequences are categorized into functional causes for containment failure much in the way the level 1 sequences were classified with respect to functional challenges to core cooling.

The last letter in the plant damage state identifier represents the timing of the event. It is noted that the timing specified in this identifier is relative to the initiating event. The timing of the potential for containment failure with respect to core damage is also important but is specified as a part of the release mode, which is covered in the next paragraph.

4.3.3 Release Mode

The release mode describes the type of releases for source term binning, and is shown for each sequence on the CETs. The release mode codes are:

1. A- Containment or reactor vessel is intact at accident termination. An example is the containment was intact at the time of core damage and the accident was terminated with the reactor vessel intact.

2. B- Containment failure occurs with release scrubbed through the suppression pool. An example is loss of decay heat removal with containment overpressure failure in the wetwell airspace. Scrubbing of the releases through the suppression pool occurs for this release mode.

3. C- Containment failure precedes or is coincident with reactor vessel failure. The suppression pool is bypassed. Examples are liner meltthrough, overpressure failure due to ATWS, or loss of decay heat removal with containment failure in the drywell.

4. D- Containment failure is delayed after reactor vessel failure. The suppression pool is bypassed. An example is vessel failure into an intact containment, but containment failure occurs later because of non-condensible gas generation from core concrete attack.

5. E- Radionuclides exit the primary coolant system directly to the reactor building through an unisolated LOCA outside the containment. An example of this is a Class 5 sequence or interfacing system LOCA.

Subcategories of the release modes are shown in Table 4.3-4. Generally speaking, odd numbers indicate a small containment failure and even numbers indicate a large containment failure. A spectrum of methods for scrubbing fission products prior to release are distinguished in this table (i.e. sprays, pool of water overlying debris, none).

Table 4.3-1:
Front-to-Back-End Interface

Containment Event Tree	Inputs
1A	- Class 1A, (TQUX). - Flood sequences with Reactor at High Pressure. - Small LOCAs with Reactor at High Pressure.
1B	- Class 1B, (Station Blackout).
1D	- Class 1D, (TQUV). - Flood sequences with Reactor at Low Pressure. - Small LOCAs with Reactor at Low Pressure.
2	- Class 2 (TW).
3	- Class 3 (LOCA) - Large and Medium LOCAs where Vapor Suppression Succeeds. - Sequences where SRVs do not open causing Reactor Overpressure Failure.
4	- Class 4 (ATWS). - LOCAs where Vapor Suppression Fails.
--	- No CET for Class 5 (LOCA outside containment, or internal flood due to Torus Ring Header break) because both have containment failure as part of the initiating event.

Table 4.3-2
MONTICELLO LEVEL 1 TO LEVEL 2 DEPENDENCIES

	DEBRIS COOLING									REACTOR DEPRESS	CONTAINMENT HEAT REMOVAL		
ACCIDENT CLASS	FEED-WATER	HPCI	RCIC	CRD[1]	CONDEN-SATE	LPCI	CS	RHRSW	CONT. SPRAY[2]	SRVs	MAIN COND[3]	RHR	VENT
1A	--	--	--		✓	✓	✓	✓		✓[4]		✓	✓
1B	✓[5]	--	--		✓[5]	✓[5]	✓[5]	✓[5]		✓[5]		✓[5]	✓[5]
1D	--	--	--		--	--	--	✓[6]		N/A		✓[7]	✓
2	✓	--[8]	--[9]		✓[10]	✓[10]	✓[10]	✓[10]		✓[10]	N/A	N/A	N/A
3B	✓[11]	✓[12]	✓[12]		✓	✓	✓	✓		N/A[12]		✓	✓
3C	✓[11]	--	--		✓[11]	--	--	✓[6]		N/A		✓[7]	✓
4	✓[13]	--[14]	--[14]		✓[13]	--[14]	--[14]	✓[13]		✓	N/A	N/A	N/A

✓ Credited in level 2.
-- Failed as part of level 1.
N/A Not relevant to outcome of sequence.
(1) Not credited due to limited flow capacity.
(2) Use prohibited by DSIL.
(3) Not credited due to high steamline radiation isolation or low reactor pressure.
(4) Credited recovery of operator failure to depressurize.
(5) Credited only on recovery of AC power.
(6) Potentially available given additional time to align.
(7) RHR highly dependent on LPCI.
(8) Failed due to suppression pool temperature.
(9) Failed due to suppression pool temperature or high turbine exhaust pressure.
(10) Provided containment pressure < 70 psig (SRV closure).
(11) Sufficient for debris cooling once blowdown is over.
(12) Vessel depressurization for medium to large LOCAs.
(13) Sufficient for debris cooling once subcritical.
(14) Failed due to environment in reactor building.

Table 4.3-3:
LEVEL 2 DAMAGE STATES

CODE = A BB C

A: REACTOR STATE

R = Arrest in vessel. Accident terminated prior to vessel penetration.

H = Vessel penetration at high pressure.

L = Vessel penetration at low pressure.

BB: CONTAINMENT STATE

XX = Intact. No containment failure.

VS = Containment vent used - scrubbed release.

VB = Containment vent used - unscrubbed release.

OD = Overpressure failure due to decay heat or non-condensible gas generation.

OA = Overpressure failure due to ATWS.

OH = Overpressure failure due to hydrogen combustion.

LM = Liner meltthrough.

OT = Overtemperature failure.

CI = Containment isolation failure.

CC = Core-concrete interaction.

C: RELEASE TIMING

X = No release (other than normal leakage).

L = Late release (~24 hours after initiating event).

I = Intermediate release (~4-24 hours after initiating event).

E = Early release (~0-4 hours after initiating event).

TABLE 4.3-4:
RELEASE MODES

		RADIONUCLIDE LOCATIONS PRIOR TO VESSEL FAILURE [7]	SMALL FAILURE		LARGE FAILURE	
			Before or as a result of VF	Delayed after VF	Before or as a result of VF	Delayed after VF
Through Suppression Pool	CS [6]	WWA/DW	X	X	X	X
		WWW	X	X	X	X
	No CS + No RPV Inj.	WWA/DW	X	X	X	X
		WWW	X	X	X	X
	No CS + No RPV Inj.	WWA/DW	X	X	X	X
		WWW	✓[1] B1	X[2]	✓[1] B2	X[2]
Bypass Suppression Pool	CS	WWA/DW	✓[3] C1	X	✓ C2	X
		WWW	✓[3] C3	✓[4] D1	✓ C4	✓[4] D2
	No CS + RPV Inj.	WWA/DW	✓[3] C5	X	✓ C6	X
		WWW	✓[3] C7	✓[4] D3	✓ C8	✓[4] D4
	No CS + No RPV Inj.	WWA/DW	✓[3] C9	X	✓ C10	X
		WWW	✓[3] C11	✓[4] D5	✓ C12	✓[4] D6
Bypass Containment	CS	WWA/DW	X	X	X	X
		WWW	X	X	X	X
	No CS + RPV Inj.	WWA/DW	X	X	X	X
		WWW	X	X	X	X
	No CS + No RPV Inj.	WWA/DW	✓[5] E1	X	X	X
		WWW	X	X	X	X

(1) Debris cooling & RPV release location before vessel failure are not significant to release category definition due to large suppression pool decontamination factor.
(2) Timing of release vs. vessel failure is not important to release category definition as releases are principally limited to noble gases.
(3) Distinction between WWA/WW and WWW important only for sequences with failure to isolate because liner meltthrough is only other small, early failure mechanism.
(4) Distinction between WWA/SW and WWW is not significant to Release Mode definition because of long gravitational settling time for aerosols.
(5) Marked bex indicates expected dominant sequence.
(6) CS = Containment spray.
(7) DW = drywell, WWW = wetwell below waterline, WWA = wetwell airspace.

4.4 Containment Failure Characterization

This section presents discussions of the various potential Mark I containment failure mechanisms and summaries of the calculations which were performed to determine their applicability to Monticello.

4.4.1 Direct Containment Bypass

The possibility of a direct containment bypass event was considered explicitly in the level 1 sequence quantification. The initiating event, unisolated LOCA outside containment, was subdivided by area and system because different systems would be affected by different pipe break locations. Breaks with an initiating event probability less than 1E-7 per year were screened out because they could not lead to accident sequences with significant risk. The break locations considered were all locations in the reactor building containing high energy lines as well as the feedwater and steam lines in the main steam tunnel. The initiating event frequencies were calculated from generic pipe rupture failure rates from WASH-1400. Plant specific valve and pipe arrangements were considered in the calculations.

4.4.2 Vessel Blowdown

Blowdown forces from penetration of the vessel by core debris at high pressure were considered. An analysis was performed to show the vessel could withstand the upper bound jet thrust that might be expected at the time of vessel breach.

The upper bound jet thrust from the vessel was estimated at 562 kips. The vessel is supported such that it can withstand a force of 562 kips acting on it without moving.

Jet reaction force due to break of recirculation outlet nozzle have a design load of 658 kips. This was greater than the estimated upper bound (562 kips) and shows the vessel supports are designed to withstand a lateral force equal to the upper bound estimate.

The vertical jet reaction force was estimated from the weight of the RPV and some of its internal components. The component weights added up to 882 kips. This value alone was greater than the upper bound estimate for jet reaction of 562 kips. The weight of the vessel is carried by the RPV support skirt. The support skirt is attached to the interior concrete wall inside the drywell that surrounds the skirt and is securely embedded, about 13 feet deep, in the concrete floor below it. The primary containment loading drawing states that the total vertical load (normal operation - static) at the base of the concrete cylinder surrounding the skirt is 6465 kips. That implies that the total weight of the vessel and its internals, supports and shielding is less than 6465 kips. It can be safely assumed that the amount of thrust in the vertical direction required to move the vessel is somewhere between 882 and 6465 kips. The upper bound estimate for vessel blowdown jet reaction is less than this and therefore the vessel will not move in the event of a vessel blowdown.

4.4.3 Steam Explosions

Steam explosion phenomena were evaluated for both in-vessel and ex-vessel steam explosions as potential mechanisms for containment failure under accident conditions and, therefore, as potential causes for radioactive releases to the environment.

In-vessel

The issue for in-vessel steam explosions is whether an explosion of sufficient magnitude to fail the reactor vessel, with consequential failure of the containment, could occur. This was addressed by evaluating the fundamental physical processes required to create an explosion of such magnitude. The analysis closely follows the IDCOR assessment of this phenomenon [4.4-1] and indicates that explosions of this magnitude are not likely to occur within the Monticello reactor vessel. This is in agreement with the findings of the NRC sponsored Steam Explosion Review Group (SERG) [4.4-2] which concluded that the likelihood of an in-vessel steam explosion leading to containment failure, alpha mode failure, was very unlikely.

Experimental evidence [4.4-3] has demonstrated that a relatively high reactor coolant system pressure prevents explosions altogether. For conditions in which reactor pressure exceeds 150 psia, steam explosions are not considered possible.

For events in which reactor pressure is likely to be low, a number of conditions must be met in order to produce an energetic fuel-coolant interaction that might jeopardize the integrity of the reactor vessel:

- Large amount of core debris entering the lower plenum at once.
- Fragmentation of the hot material within the water in the lower plenum.
- A trigger to initiate the explosion.
- Efficient energy transfer from the debris to the coolant.
- An overlying slug of water to transmit the energy in a coherent fashion.
- The ability of the slug to be transmitted through the upper structures within the reactor pressure vessel.

It is recognized that each of these conditions must be achieved to create an explosion of sufficient magnitude to rupture the reactor pressure vessel; the failure of a single element is sufficient to preclude an explosion of such magnitude.

Given a core damage event, as the melt progresses to the bottom of the vessel, the debris is expected to flow into the lower plenum as opposed to dropping as a large mass. This limits the rate of energy transfer to the coolant. Further, there is no means of finely dispersing large amounts of the debris within the coolant, particularly given the limited free space provided by control rod guide and instrument tubes. Also, the inherent capability of the vessel to withstand internal forces makes it unlikely that the limited fuel-coolant interactions that may occur will compromise the integrity of the vessel.

Given the limited potential for conditions necessary to generate a large scale in-vessel steam explosion and the capability of the vessel to withstand internal forces, in-vessel steam explosions was not explicitly modeled. However, other potential early containment failure modes are modeled, such as hydrogen combustion. In-vessel steam explosions would have effects similar to the hydrogen combustion models.

Ex-vessel

Ex-vessel steam explosions also may occur in the progression of a severe accident should molten debris be discharged from the reactor vessel into a pool of water. Within the containment, the occurrence of a steam explosion could exert pressure spikes on submerged surfaces or pressurized subcompartments. Significant pressure peaks could generate missiles or impair load-carrying capabilities of walls, either of which could result in containment failure.

Two aspects of ex-vessel steam explosions were addressed in the Monticello IPE.

1. Containment Overpressure - The containment pressure increase due to the rapid generation of steam was found to be < 2 psi. This value is small compared to the expected containment failure pressure of 103 psig. The calculated pressure rise was based on the assumptions:

    - The drywell floor was flooded at the time of vessel penetration. In fact, many of the core damage sequences which dominate the IPE results occur without release of steam or water to the drywell until after vessel penetration.
    - The debris is codispersed in the water on the drywell floor, assuring a large heat transfer rate (30 Mw/m^2) to the water.
    - The heat transfer time frame is as long as 1 sec based on experimental data [4.4-4].

2. Shock Waves - The maximum temporary pressure increase at the containment boundary due to shock waves was calculated to be about 15 psi. Again, this is relatively small compared to the expected containment failure pressure of 103 psig. The pressure rise due to shock waves was estimated using the following assumptions:

    - The peak pressure at the source of the explosion is 10 MPa [4.4-5]. This corresponds to a condition of critical size bubble growth. Above this pressure vapor cannot be produced at a higher pressure than the local pressure.
    - Transmission of the force to the containment wall through an air-steam-hydrogen mixture as the drywell is only partially full of water up to the drywell-to-wetwell vent pipes.

- Extrapolation of the pressure from the source of the assumed explosion, roughly 0.5m or less diameter based on vessel hole size, inversely proportional to the size of the containment (13 m diameter).

Like the in-vessel steam explosion challenges, for ex-vessel explosions have not been modeled because of the very low potential for containment failure from this challenge. Again, other early containment failure modes have been included in the CET which bound, both in terms of probability and consequences, the effects of ex-vessel steam explosions.

4.4.4 Penetration Thermal Attack

Containment penetration thermal attack is a postulated condition where non-metallic seal materials in containment penetrations are exposed to elevated drywell atmosphere temperatures for prolonged time periods during a severe accident. Following vessel failure, drywell gas temperatures may reach sufficient levels to reduce penetration seal performance to the extent that a containment breach effectively occurs earlier than the failure times of other containment failure mechanisms [4.4-6, 4.4-7, 4.4-8]; e.g., overpressurization or concrete erosion. The impact on containment failure timing thus depends on the gas temperatures achieved, the exposure time at elevated temperatures, and the characteristics of the materials involved.

The issues important to thermal attack are the severe accident thermal loadings for non-metallic penetration seal materials; and the potential for accelerated adverse thermal effects on material properties that influence sealing performance. At drywell gas temperatures above approximately 700° F, previous studies indicate that mechanical failure of the drywell shell is likely to supersede concerns about non-metallic materials performance. Consequently the time frame that was addressed for thermal loading purposes is the period leading up to the point where 700° F is exceeded.

The potential for penetration thermal attack included an identification of non-metallic pressure retaining parts of the containment boundary. These penetrations and the materials identified in this investigation are summarized in Table 4.4-1.

Representative drywell severe accident temperature profiles were also reviewed as a part of the investigation. Four accident sequence types were selected to define the thermal conditions to which the penetrations might be exposed. Temperature profiles used in the thermal analysis of Monticello containment penetrations are shown in Figures 4.4-1 through 4.4-4.

Figure	Accident Class	Description
4.4-1	1D	Transient-initiated core melt at low reactor pressure. No low pressure injection or debris cooling after vessel penetration. (Dry/Dry)
4.4-2	3C	LOCA-initiated core melt at low reactor pressure. Debris cooling occurs only by water which was spilled on the drywell floor as a result of the LOCA. (Wet/Dry)
4.4-3	1A	Transient-initiated core melt at high reactor pressure. Operation of low pressure coolant makeup systems was assumed for debris cooling after vessel penetration and depressurization. (Dry/Wet)
4.4-4	3B	LOCA-initiated core melt at high reactor pressure. Operation of low pressure injection systems was assumed for debris cooling after vessel penetration and depressurization. (Wet/Wet)

Other accident classes important to the level 1 Analysis are either similar to these four accident classes or lead to containment failure for reasons other than thermal attack.

An aging calculation was performed for each of the non-metallic materials for each of the four relevant accident classes. In addition, a failure modes and effects analysis of each of the penetrations was performed to establish the possible impact of its failure on the magnitude of release from containment. Results are presented in Table 4.4-1 and are summarized below.

- Thermal attack of penetrations is not a significant failure mode in accident sequences where debris cooling is successful in preventing a significant rise in the containment temperature. Aging of non-metallic components is not sufficient to compromise the containment pressure boundary.

- For accident scenarios where no debris cooling occurs, drywell integrity may be a concern. For these sequences, the significance of thermal attack depends on whether failure of the penetrations occurs prior to other failure modes and whether the size of the release path through the penetration is significant.

- Of the penetrations examined at Monticello, only the vent valve boot seals are expected to fail due to temperature prior to other containment failure mechanisms. The vent valve boot seal is not important to the leak tightness of the valves because an upper bound MAAP analysis has shown that only a minimal release of radionuclides would occur if these seals fail.

Thermal attack of the containment boundary was incorporated into the Monticello CETs by assigning an overtemperature failure mode to CET branches where the core melt proceeded to the point of vessel penetration and no systems were available to provide debris cooling. The timing of overtemperature failure is that time required to achieve more than 700° F in the drywell.

4.4.5 Containment Isolation

Isolation valves are provided on lines penetrating the drywell and suppression chamber to assure integrity of the containment under accident conditions. Those isolation valves which must be closed to assure containment integrity immediately after a major accident are automatically controlled by the plant protection system.

Many different types of penetrations were considered during the containment isolation evaluation. The following piping and hatch penetration groups were examined:

1. Feedwater, main steam lines, and associated main steam drain lines,

2. HPCI steam line,

3. RCIC steam line,

4. CRD lines,

5. Low pressure ECCS lines (for interfacing systems LOCA considerations),

6. Instrument lines,

7. Personnel locks, hatches, and drywell head,

8. Cable penetrations,

9. Instrument air lines,

10. Reactor building closed cooling water lines, and

11. Purge and vent lines.

The penetrations listed in numbers 1 through 6 above are important primarily during analysis of potential containment bypass or interfacing systems LOCAs. Breaks or leaks in those lines could result in direct release of fluids and radioactivity from within the reactor vessel to the reactor building.

For penetration/piping groups 7 through 11 above, isolation is necessary to prevent flow of containment atmosphere into the reactor building or outdoors. Radionuclide release into the containment or containment pressurization may occur as a result of an accident. Containment isolation minimizes releases to the outside atmosphere and avoids potential adverse impacts on accident mitigating systems.

The following criteria were used to focus the analysis on those penetrations that contribute most significantly to a release:

- Penetrations of closed piping systems: If the system is not open to the containment atmosphere the probability of simultaneous failure of the isolation valve(s) in the system and a pipe break is negligibly small. This criterion was used to screen out RBCCW piping.

- Hatches, personnel locks, drywell head: These items are considered part of the continuous liner of the containment and therefore are factored into the analyses performed for containment overpressure and overtemperature failures. They are not considered here.

- Pipes with diameters less than 2": These pipes, such as instrument and sample lines, are not considered important for two reasons. Aerosol plugging is likely to reduce the amount of leakage which could occur through these pipes; and they are not large enough to relieve containment pressure fast enough to prevent eventual containment overpressurization failure.

Table 4.4-2 shows the containment penetrations which were not screened out using the criteria given above. The table shows the configuration of the containment isolation valves, normal position, signals to close the valves, and dependencies of the valves on support systems for motive and control power.

Table 4.4-3 gives the resulting containment isolation failure probabilities, shown by availability of support systems. All level 2 sequences analysis used the value 3E-4 as an upper bound failure probability.

4.4.6 Containment Over-pressure

Explicit consideration was given to the potential for containment pressurization from a variety of sources depending on the characteristics of the accident sequences in question. Short term challenges include:

- Reactor blowdown
- Steam generation from ATWS
- Ex-vessel steam explosion
- High pressure vessel penetration
- Hydrogen combustion

Long term challenges include:

- Non-condensible gases produced from molten core concrete interaction
- Steam generation from decay heat

Two factors are important in determining the potential for release and establishing the effects of containment failure on accident sequence progression: the capability of the containment to withstand these challenges; and the timing and location of containment failure should it occur. An examination of the Monticello containment to determine the probability of failure under various accident loads was performed as a part of the IPE.

Table 4.4-4 identifies the components considered as a part of this analysis. A series of analyses and tests were examined to determine the expected failure pressures these components. Among these analyses was a Monticello specific analysis performed to establish the most likely containment failure locations on overpressure [4.4-9]. In addition to the studies listed in Table 4.4-4, a Chicago Bridge and Iron analysis of Mark I containments performed for the BWR Owners Group [4.4-10] and the structural results for available PRAs from Browns Ferry 1, Cooper, Peach Bottom and Quad Cities [4.4-11] were used as input to this analysis.

Probability distribution functions were derived for three locations in the containment; the drywell head, the wetwell-drywell vent line bellows, and the wetwell airspace. These three locations were selected because they bound the capacity of other containment components. The three locations also adequately represent the effects of containment failure in any of the other locations with respect to source term or environmental effects in the reactor building. The variances associated with the torus shell and the wetwell-drywell vent bellows were based on carbon steel material properties. The drywell head closure was assumed to have larger variance than simply material properties because it is a function of bolt preload, gasket spring back and other factors.

The probability distribution function used in the Monticello IPE is presented in Figure 4.4-5. The median containment failure pressure is estimated to be 118 psia consistent with the Monticello containment capacity analysis. The vent pipe bellows dominate the failure probability above this pressure. Below the median the drywell head dominates the probability of containment failure due to uncertainties associated with bolt preloading, gasket response, etc.

MAAP analyses of each accident sequence type were performed to establish containment pressure as a function of time. Given these pressure profiles, application of this curve to the accident sequence quantification occurred in one of two ways.

A number of challenges are limited with respect to the maximum pressure which can be attained in the containment. These challenges include:

- Reactor Blowdown (LOCA)
- Hydrogen combustion
- Ex-vessel steam explosion
- High pressure vessel penetration

The peak pressure expected as a result of the challenge was derived for these events, and the probability of containment failure at this pressure was obtained from the probability distribution function curve. None of these events produced any significant challenge to the containment. However, the method MAAP uses for modeling hydrogen combustion does not produce the limiting challenge to containment integrity. Since hydrogen combustion has a very small probability of occurring, detailed analysis was not warranted. The containment was assumed to fail any time hydrogen combusted within it.

A few challenges are assumed to pressurize the containment to its ultimate capacity. For example:

- Steam generation due to ATWS
- Steam generation due to decay heat (loss of DHR)
- Unquenched molten core concrete interaction generating non-condensible gas

For these types of events, the pressure at which containment failure is assumed to occur is the median point on the probability distribution curve, 118 psia. MAAP analysis of the time to reach this pressure for each of the various accident scenarios was performed. This time was used as input to the derivation of human error probabilities or repair and recovery factors to prevent the challenge from failing the containment.

4.4.7 Liner Meltthrough

Liner meltthrough is a potential early containment failure mode that could occur once a core melt has progressed to the point of lower head penetration. Molten core debris ejected from a failed reactor vessel could flow across the drywell floor, come into contact with the containment shell and ablate an opening through it. The largest potential for the occurrence of liner meltthrough would be during core melt accidents that have a fairly coherent pour of molten material at vessel failure. The localized accumulation of a significant debris mass in contact with the drywell shell is considered much more likely during such sequences. This implies that liner meltthrough is more likely for core melt sequences that fail the reactor vessel at low pressure. During high pressure core melt sequences, the exiting gas stream would be expected to have sufficient energy to disperse the debris leaving the vessel in a relatively incoherent manner. Also the amount of equipment under the vessel would help to break up any coherent flow. This dispersal of the debris substantially minimizes the probability of a coherent stream of molten debris contacting the drywell wall for a substantial period of time.

After reactor pressure vessel failure, a core debris pool would form on the pedestal floor and fill the containment sumps. Corium could flow through the pedestal doorway and reach the drywell shell. Since the distance from the doorway to the shell is only a few meters, and the viscosity of molten core debris might be low, molten material could reach the drywell shell if heat losses to concrete or overlying water are not sufficient to freeze the material. If the melt temperature is above the melting point of steel (1750° K) after the melt reaches the shell, the potential for local meltthrough may exist.

Two aspects of Monticello plant response affect the potential for liner meltthrough. The most important is the volume of core debris available at the time of vessel penetration as compared to the containment sump capacity.

A summary of the material available to contribute to the volume of the core debris follows:

	Mass	Volume
UO_2	8.6E4 kg	7.8m^3
Zr (including channels)	3.6E4 kg	5.5m^3
Core support plate	5.7E3 kg	0.7m^3
Lower Head (~1/2)	5.3E4 kg	3.3m^3
Total		17.3m^3

Approximately 50% of the core debris is expected to exit the vessel shortly after vessel penetration. This assumption is consistent with MAAP results as well as Theofanous' work [4.4-15]. Total volume of debris assumed to flow into the pedestal at the time of vessel penetration is therefore approximately 8.7m^3.

The Monticello pedestal area contains two sumps 6 ft by 6 ft by more than 3 ft deep which must fill prior to debris flowing at any substantial rate into the drywell. In addition, pump cavities are located in the drywell floor, each 2.5 ft by 2 ft by over 3 ft. Figures 4.4-6 and 4.4-7 contain a plan and cross section of the drywell showing the location and size of the sumps. Note that total volume of these sumps and attached piping is therefore 9.0m^3. This is slightly larger than the volume of the debris expected at the time of vessel penetration. Even excluding the pump cavities outside the pedestal area, more than 6.5m^3 of volume are available to retain the debris. With a drywell floor of 94m^2, the debris thickness is only 2 cm once the debris spreads over the drywell floor.

The second important aspect of the accident progression at the time of vessel penetration is the rate of flow of the debris. While little or no debris is expected to flow out of the pedestal region following vessel penetration, the limited amount that may be available would flow out at the end of the initial pour.

A melt spreading analysis for Mark I containments has also been presented by Kazimi [4.4-16]. The analysis assumes that the melt spreads over the pedestal floor and through the doorway into the annular floor region of the drywell. The melt progression is assumed to be semi-circular in the annular region. The analysis then estimates the distance the melt will progress until its temperature reaches the freezing point. An energy balance at the semi-circular leading edge, which accounts for downward heat transfer to the concrete and heat generation from the full oxidation of zirconium by gases generated in the concrete, is considered. The downward heat flux to concrete is described by the work of Kao and Kazimi [4.4-17] which characterizes the heat flux to concrete by an analogy with nucleate boiling. Upward heat loss due to the presence of water or thermal radiation into the drywell environment was not considered. A range of pour rates and initial temperatures were modeled, and the three types of concrete usually used were considered: limestone, limestone/common sand, and basaltic.

Based on calculations from this model, Kazimi concludes that at relatively low pour rates, there is an effective cooling period due to heat transfer to concrete even in the absence of water. Furthermore, for a mostly oxidic melt, Kazimi concludes that the melt may freeze in a short distance. For a metallic melt, it is possible that the sensible heat will be removed before reaching the wall if the initial superheat and pour rate are not too great. This work also suggests that if the rapid quenching potential of water is included, it will be unlikely that a melt pour will reach the shell with significant superheat.

Finally, it should be recalled from the level 1 results that a number of scenarios in the Monticello PRA occur with the reactor at high pressure. Low pressure coolant injection systems are likely to be available but are simply prohibited from cooling the fuel in the vessel because of the inability to depressurize the reactor. For these sequences, penetration of the lower head by core debris will result in a flow of water to the pedestal and drywell once the vessel depressurizes. Again, this is expected by the time the initial pour from the vessel ends and will provide significant cooling and freezing of the last of the debris if it flows from the pedestal into drywell area.

Given the large sumps located in the pedestal area and their capability of retaining most or all of the debris that may exit the vessel; the relatively low pour rate for any debris that might flow into the drywell from the pedestal once the sumps fill; and the potential for water flow from low pressure injection systems at the end of the initial pour, the potential for debris reaching the drywell shell and melting through the liner is very small at Monticello. Provisions for liner meltthrough are included in the CETs following failure of in-vessel injection systems to prevent vessel penetration, but is assigned a probability of zero. A sensitivity study was done by assigning liner melt a probability of 0.5 with no substantial change to the back-end results.

4.4.8 Direct Containment Heating

Direct containment heating is a postulated mechanism for rapid heat transfer from molten core debris to the drywell atmosphere. This could happen if the reactor vessel is penetrated when its internal pressure is high, causing rapid dispersal and fine fragmentation of the exiting debris. When combined with additional energy generated by debris oxidation and hydrogen combustion, direct containment heating may be sufficient to cause an early containment failure if a large portion of the molten core is involved. The extent of pressurization thus depends upon the amount of debris which is discharged at vessel failure; the configuration of the plant which may enhance or hinder dispersal beyond the pedestal; the fraction of the debris which can be finely fragmented and dispersed throughout the containment atmosphere; and the ability of debris to transfer heat into various areas of containment.

BWRs have several design and operating characteristics that significantly limit the magnitude of the pressure rise associated with direct containment heating:

- Reactor Depressurization System
- Inerted Containment
- Suppression pool

The most significant means of preventing direct containment heating is to assure reactor depressurization (<200 psia). The Monticello plant has eight SRVs, any one of which is capable of assuring low reactor pressure at the time of vessel penetration. The CETs explicitly account for the potential for depressurization with this system. The effects of direct containment heating apply only to those accident sequences in which depressurization is unsuccessful.

The Monticello containment is also normally inerted. Additional heat addition by hydrogen combustion during reactor blowdown is not possible under inerted conditions.

The Monticello containment also has the suppression pool to absorb the heat that is released from the reactor during blowdown. For transients in which the reactor is at full pressure, a large fraction of the non-condensibles are forced into the wetwell airspace during blowdown, causing a pressure rise in the containment. A bounding analysis was performed to determine the pressure rise associated with this additional heat input at the time of vessel penetration. In this analysis, all of the latent heat and sensible heat from the debris as well as heat from oxidation of the remaining zirconium in the ejected debris is added to the containment. The various components of this heat addition include:

- Debris sensible heat = 8E10 Joules (assumes 50% of the UO_2, Zr, and core support plate and 25% of the lower head)
- Debris latent heat of fusion - 2E4 Joules
- Zirconium oxidation = 1.2E11 Joules (assumes no oxidation occurs in-vessel)

For Classes 1A (transient at high reactor pressure) and 1B (station blackout), core damage would occur several hours into the event. The suppression pool is subcooled at the time of blowdown. MAAP analysis of these sequences shows suppression pool temperatures <150° F subsequent to the blowdown. Adding the total amount of heat from the ejected debris to the drywell forces more steam from the drywell to the suppression pool, heating the pool further. Assuming there are 2E6 kg water in the suppression pool yields a 45° temperature rise in the pool. This results in only a 5 to 7 psi rise in containment pressure.

As noted above, the Monticello CETs explicitly account for the effects of reactor pressure at the time of vessel penetration on containment and containment system response to core melt ejection. Direct containment heating loads apply only to those sequences in which reactor depressurization is unsuccessful. Even then, the suppression pool's capacity for accommodating heat from the ejected debris means the containment pressure rises only a few psi.

4.4.9 Core Concrete Interaction

Molten core debris ejected from a failed reactor vessel would come into contact with the containment floor and may eventually erode a large enough volume of concrete that either the reactor pedestal walls would lose their load-carrying capability; the basemat would be penetrated and core debris would exit the containment; or sufficient non-condensible gases would be generated to fail the containment on overpressure.

In a BWR plant, the concrete surface that experiences the most severe thermal attack is the pedestal floor. The heat transfer between the core debris and concrete drives the thermal decomposition and erosion of the concrete. The thermal attack on the concrete can be broken up into three different phases:

1. a short-term, localized attack as debris leaves the reactor pressure vessel;

2. an aggressive attack by high-temperature debris immediately after the core material leaves the reactor; and

3. a long-term attack in which the debris temperature would remain essentially constant and the rate of attack is determined by the internal heat generation.

<u>Localized Attack</u>

Immediately after vessel failure, debris is discharged from the vessel into the pedestal region. This material, which may be molten, induces an aggressive localized jet attack upon the concrete surface. The thermal attack is confined to the area where the jet impinges. Estimates of this attack based on analyses in [4.4-1] show the eroded depth to be perhaps 10 to 20 centimeters, depending upon the primary system conditions at vessel failure.

<u>Attack by High-Temperature Debris</u>

After the jet attack, the reactor cavity or pedestal region may be covered by high-temperature debris which aggressively attacks the concrete substrate. Free water, bound water, and other gases generated by concrete decomposition are then released. The gases agitate the melted material and promote convective heat transfer between this material and the concrete. The combination of (a) the sensible heat added to the concrete, (b) the endothermic chemical reactions involved in releasing water vapor and decomposing the concrete, and (c) the latent heat of fusion for melting the substrate extracts a considerable amount of energy from the high-temperature melted material. In fact, the aggressive attack generally absorbs more energy than is generated by the decay power. Additional internal heat generation in the melt can result from the oxidation of metallic constituents by the gases released from the concrete substrate.

Typically, the high-temperature, aggressive attack is driven by the internal heat generation from metal oxidation and to a lesser extent by the initial stored energy of the debris.

Long-Term Attack

During the long-term attack, the debris remains at an essentially constant temperature, and the rate of attack is determined by the difference between the internal heat generation and the heat losses to the containment environment. These heat losses are principally due to convection and radiation, and are somewhat influenced by the natural convection of high-temperature gases throughout the containment. The resulting concrete attack rate is much reduced from that typical of the high-temperature attack phase and occurs over a much longer interval. The non-condensible gases generated during this period may contribute to long-term overpressurization of the containment.

The major physical phenomena affecting the extent of concrete erosion by core debris are closely interrelated and therefore difficult to separate. For the purpose of this discussion, these phenomena are:

1. the rate and amount of core debris expelled from the reactor vessel,

2. the configuration of the debris mass on the concrete,

3. depth of the melt bed, and

4. the quenching effect of water.

The most notable feature of the Monticello plant response to core concrete interaction is the size of the containment sumps. As noted in Section 4.4.7, the sumps are capable of containing virtually all of the debris expected at the time of vessel penetration. To account for the effects of this sump geometry, the sump was modeled in the MAAP parameter file as a pedestal floor area equivalent to that of the sumps with an elevation 1 meter below the drywell floor.

Debris cooling characteristics associated with the containment sumps were derived using DECOMP, a subroutine of MAAP code. DECOMP considers the debris to be either a solid cylinder or a molten pool surrounded by a crust, depending upon its energy. Curst growth/shrinkage based on energy balance describes the solidification process occurring within the molten debris, and temperatures are determined from phase diagrams based on the composition of the debris. Transient conduction calculations are carried out in the concrete floor, sidewall in contact with the debris, and upper surface. Concrete ablation is allowed at all surfaces. The heat transfer coefficient at the molten pool/crust interface and to the overlying pool of water, when present, are user-defined constants. Sensitivity studies were performed by varying the ability to transfer heat to the coolant, establishing a range of potential effects from non-condensible gas generation and containment pressurization. Using realistic assumptions for the ability of the coolant to penetrate into the debris, debris cooling and termination of non-condensible gas generation is expected following vessel penetration. Approximately 162 Kg moles of gas are generated which contributes to the containment pressure increase.

Generic Letter 88-20 states that if the debris bed thickness is less than 25 cm, the debris can be considered coolable. Above a debris thickness of 25 cm, both coolable and non-coolable outcomes should be considered. The Generic Letter does not provide a basis for the 25 cm debris coolability criterion. For the purpose of the Monticello IPE, mechanisms which might prevent cooling the debris were postulated in order to perform relevant sensitivity studies.

- Impermeable crust formation. The sumps at Monticello are large, 6 ft on a side and >3 ft deep. Formation of a structurally stable impermeable crust across a span this large is difficult to conceive. Furthermore, water flowing over the crust would cause shrinkage and cracking, allowing water to penetrate into the debris below the crust. In addition, sparging of the debris by gases generated from core concrete interaction would be expected to break up any crust that forms. This mechanism is not expected at Monticello.

- Inability of water to penetrate into the debris bed. An upward flow of steam and gases from core concrete interaction may prevent water from penetrating deeply into the debris bed. In this instance, the ability to cool the debris can be determined simply by performing a heat balance comparing the transfer of energy to the overlying pool of water and assuming the residual energy results in concrete ablation.

The second of these two postulated debris cooling processes was used for performing molten core concrete interaction sensitivity studies. In the analysis performed, the sumps in the pedestal were assumed to be completely full of debris, roughly half that originally contained in the vessel. The heat flux associated with 25 cm of debris was determined by considering decay heat rates at 6 hours following the initiating event. This heat removal rate was applied to the surface of the debris throughout the problem.

As additional debris gradually flows from the vessel, it is assumed either to flow out into the drywell or be quenched prior to entering the sump. As the molten debris in the sump generates heat, it begins eroding the sump downward and sideways, slowly increasing the surface area of the debris and heat transfer to the overlying pool of water. The sideways to downward erosion rate is assumed to be 0.3 based on the results of Beta tests [4.4-4]. Concrete erosion is assumed to stop when the surface area of the debris is sufficiently great to accommodate all of the heat being generated in the debris.

Under these conditions, a surface area of 26 m^2 is necessary to transfer all decay heat to the water overlying the debris. An area of this size centered in containment would encompass the pedestal support for the reactor vessel. However, given the finite amount of debris available to erode the concrete in the downward direction as well as horizontal, the pedestal may be left supported by a ledge. Most notable about the analysis is that during most of the erosion process, a large fraction of the heat is being removed by the water. More than 6 days are required to erode the basemat to this extent, and generate 113 Kg moles of non-condensible gases compared to the initial 273 Kg moles of gas in the containment.

4.4.10 Combustion

The Monticello containment is normally inerted. For combustion to happen, a core damage event must occur during the limited periods of time that the containment is partially deinerted. Only short periods of time during startup and shutdown of the reactor for less than 24 hours total duration are allowed. Assuming 6 to 7 shutdowns per year, this amounts to at most 2% of the time.

Should a core damage event occur while the containment is deinerted, the loads resulting from hydrogen combustion will vary depending on whether the event was initiated by a transient or by a LOCA and the availability of ignition sources.

For transient-initiated core damage events, hydrogen produced from metal-water reaction within the vessel will initially be directed to the suppression chamber. Hydrogen concentration will build in the wetwell airspace until vacuum breaker operation purges it into the drywell. LOCA events will initially release hydrogen to the drywell, although the drywell would be steam inerted because of steam released from the pipe break. Hydrogen would gradually flow to the wetwell during a LOCA as steam generated from quenching the debris bubbles through the suppression pool.

If an ignition source is available, the hydrogen may be burned as it is generated. Energized equipment within the drywell, such as valve motors, will likely act as a hydrogen ignition source unless there is no electrical power available (station blackout). Performing an adiabatic isochoric complete combustion pressure analysis with MAAP yields a final pressure of 45 psia for hydrogen burned near initially atmospheric conditions at its lower flammability limit, approximately 5% hydrogen mole fraction. The containment is capable of surviving such challenges. Burns at significantly higher hydrogen concentrations may result in overpressure failure of the containment, however.

The Monticello CETs explicitly consider the potential for hydrogen combustion leading to containment failure. A heading in the event tree representing the potential for the containment being deinerted at the time of core damage is provided. The heading assumes containment failure and relatively significant releases through the drywell upon combustion of the hydrogen. The ability of the containment to survive periodic burns and steam inerting of the drywell during LOCAs have been conservatively ignored in the quantification.

This resulted in conservative modeling of large containment failure occurring shortly after core damage if the containment is not inerted. More rigorous analysis would reduce the probability of containment failure upon hydrogen combustion, but this event is already so rare that the effort to improve the analysis is not warranted.

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TABLE 4.4-1

PENETRATION SEAL PERFORMANCE

Penetration	Material	Estimated Thermal Lifetime***				Comments
		Dry/Dry	Wet/Dry	Dry/Wet	Wet/Wet	
Electrical	Epoxy*	33 hr	21 hr	~10 days	~10 days	Thermal life is estimated for only first of two redundant barriers making up the penetration and credits the insulating properties of the non-metallic components.
	Hypalon	>33 hr	>21 hr	>10 days	>10 days	
	CLPE	>33 hr	>21 hr	>10 days	>10 days	
	Polyolefin	>33 hr	>21 hr	>10 days	>10 days	
	Polysulfone	>33 hr	>21 hr	>10 days	>10 days	
Butterfly Valve	EPT**	10 hr	7 hr	>10 hr	>10 hr	Boot seals for butterfly valves are not important to assuring leak tightness.
Drywell Head, etc.	Silicone Rubber	27 hr	18 hr	>1 month	>1 week	Seals were assumed to be exposed directly to drywell atmosphere.
MAAP Temp Reaches 700° F		33.75 hr	19.5 hr	NA	NA	

* Even after epoxy (and the other materials listed) fails the flow path out of the containment would be one of small diameter holes that could easily plug from the debris of the failed materials or aerosols.

** The number in the table for EPT are for EPT in direct contact with the drywell atmosphere and do not apply to the second of the two isolation valves that are in series in the vent and purge lines. The EPT lifetimes for these valves would be longer. A MAAP sensitivity study run shows that even if the boot seals did fail, only a small release would occur.

*** Refers to condition of containment before and after vessel penetration. Dry/Dry refers to no water on drywell floor before penetration, and no water addition after penetration.

TABLE 4.4-2
CONTRIBUTORS TO CONTAINMENT ISOLATION FAILURE

Description	Number	Size	Configuration	Position	Signals	Power/Air
Torus-Reactor Building Vacuum Breakers	X-218	20"	2 Parallel paths of 1 AOV and 1 check valve in series	N.C.	Open on cont pressure -10" H_2O	AOVs fail open on loss of air or loss of AC
Torus Ventilation Supply	X-218	18"	2 AOVs in series	N.C.	Group 2	AOVs fail closed on loss of AIR or loss of AC
Post LOCA Recombiner Return	X-218	6"	2 AOVs in series	N.C.	Group 2	AOVs fail closed on loss of air or AC
Torus Ventilation Exhaust	X-205	20"	1 AOV & 1 CV (2") in perallel with 1 AOV in series	N.C.	Group 2	AOVs & CV fail closed on loss of air or AC
Post LOCA Recombiner Return	X-205	6"	2 AOVs in series	N.C.	Group 2	AOVs fail closed on loss of air or AC
Drywell Ventilation Exhaust	X-25	18"	1 AOV & 1 CV (2") in parallel with 1 AOV in series	N.C.	Group 2	AOVs & CV fail closed on loss of air or AC
Drywell Ventilation Supply	X-26	18"	1 AOV & 1 CV (2") in parallel with 1 AOV in series	N.C.	Group 2	AOVs & CV fail closed on loss of air or AC
Floor Sump	X-18	2"	2 AOVs in series	N.O.	Group 2	AOVs fail closed on loss of air or AC pewer
Equipment Sump	X-19	2"	2 AOVs in series	N.O.	Group 2	AOVs fail closed on loss of air or AC power
CRD Drain Lines (2 lines)		2"	2 AOVs in series	N.O.	All scrams	AOVs fail closed on loss of air or AC power

TABLE 4.4-3
MONTICELLO
CONTAINMENT ISOLATION FAILURE PROBABILITY

	Transients and LOCAs	Loss of Air	SBO
Torus-Reactor Building Vacuum Breakers (2)	2E-10	8E-5	8E-5
Torus Vent Supply and Exhaust	1E-6	1E-6	1E-6
Recombiner (2)	5E-7	5E-7	5E-7
Drywell Vent Supply and Exhaust	1E-6	1E-6	1E-6
Sumps (2)	1E-4	1E-4	1E-4
CRD Scram Discharge Drains (2)	1E-4	1E-4	1E-4
	2E-4	3E-4	3E-4

Uses configurations described in Table 4.4.2 and the following failure rates:

AOV FTC = 5.2E-4/d
AOV FTRC = 1E-7/hr
Check Valve FTRC = 1E-6/hr
Common Cause β = 0.1

TABLE 4.4-4
MONTICELLO CONTAINMENT COMPONENT CAPACITIES

Component	Failure Pressure	Comments
Drywell Shell	139 psia	Monticello Containment Capacity Analysis [4.4-9]
Equipment Hatch	179 psia	Sandia 1:8 Scale Model Test [4.4-12]
Personnel Airlock	165 psia	Sandia/CB&I Airlock Test [4.4-13]
Mechanical Penetrations	135 psia	INEL Penetration Tests [4.4-14]
Electrical Penetrations	135 psia	INEL Penetration Tests [4.4-14]
Drywell Head	139 psia	Monticello Containment Capacity Analysis [4.4-9] (assumed to be dependent on response of gasket material)
Vent Line Bellows	120 psia	Monticello Containment Capacity Analysis [4.4-9] (based on yield stress at 86 psig-99% confidence of maintaining integrity)
Torus Shell (wetwell air space)	139 psia	Monticello Containment Capacity Analysis [4.4-9]

Figure 4.4-1
REPRESENTATIVE CLASS 1D TEMPERATURE PROFILE
FOR PENETRATION THERMAL ATTACK DRY/DRY CATEGORY

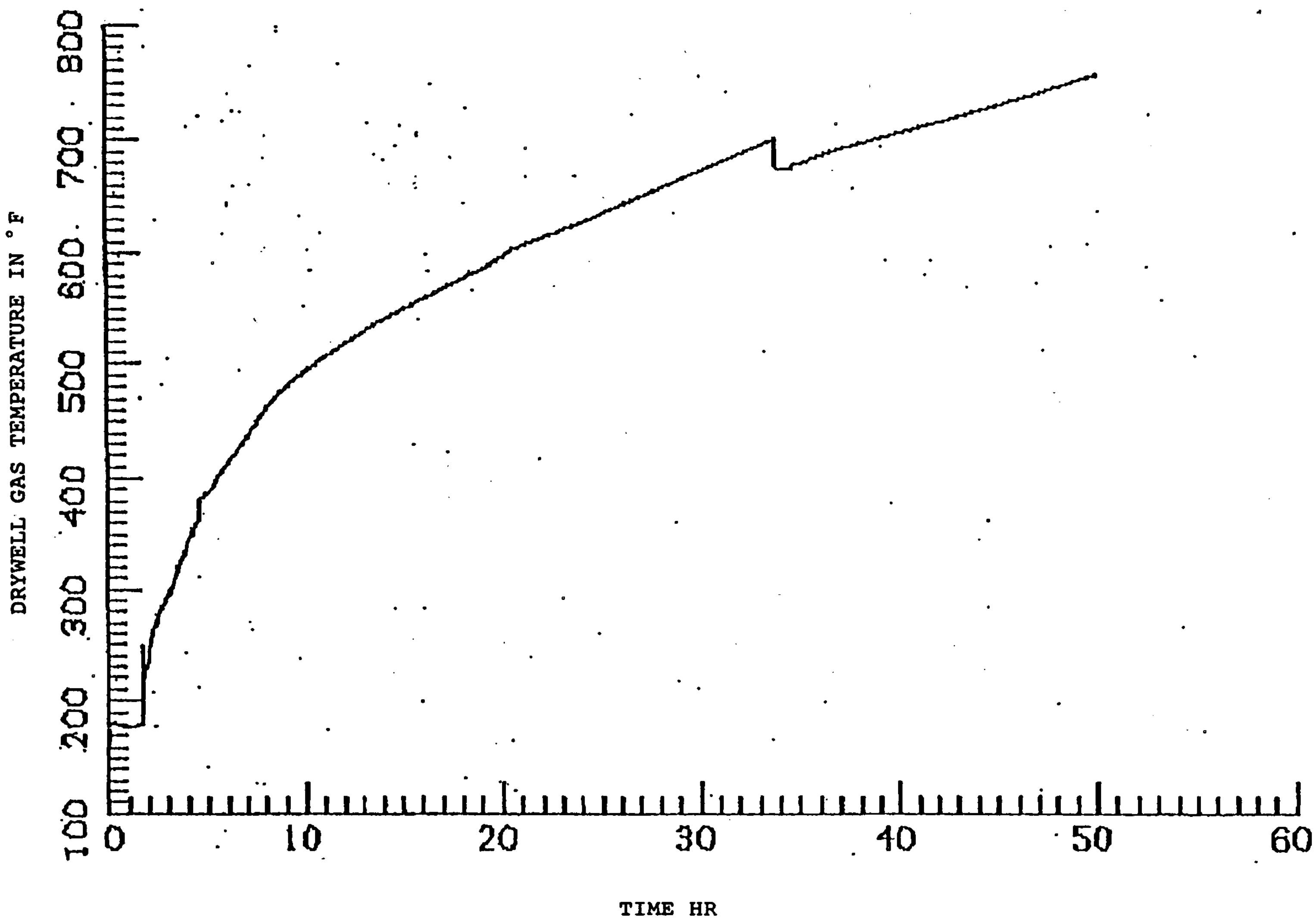


Figure 4.4-2
REPRESENTATIVE CLASS 3C TEMPERATURE PROFILE
FOR PENETRATION THERMAL ATTACK WET/DRY CATEGORY

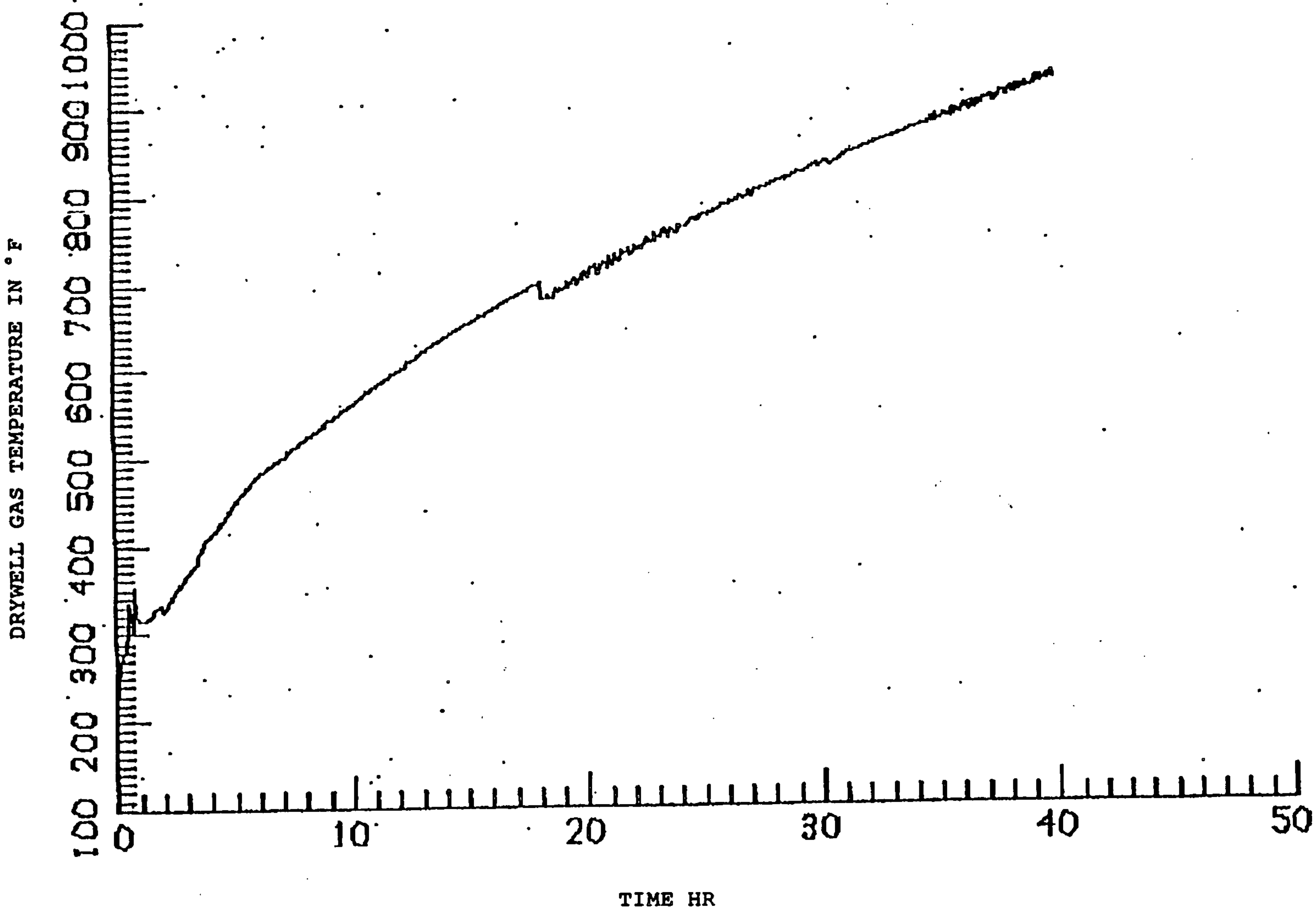


Figure 4.4-3
REPRESENTATIVE CLASS 1A TEMPERATURE PROFILE
FOR PENETRATION THERMAL ATTACK DRY/WET CATEGORY

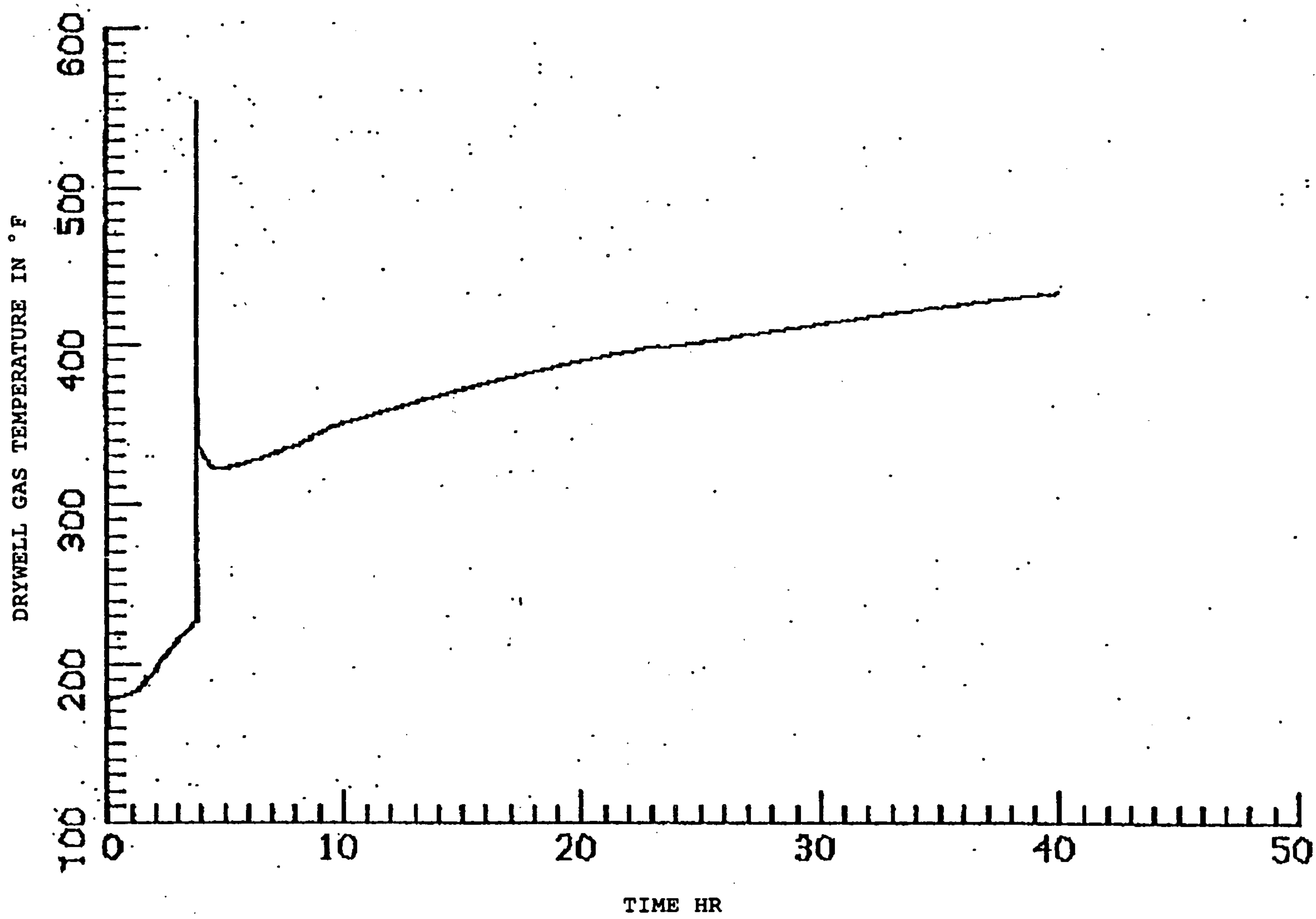


Figure 4.4-4
REPRESENTATIVE CLASS 3B TEMPERATURE PROFILE
FOR PENETRATION THERMAL ATTACK WET/WET CATEGORY

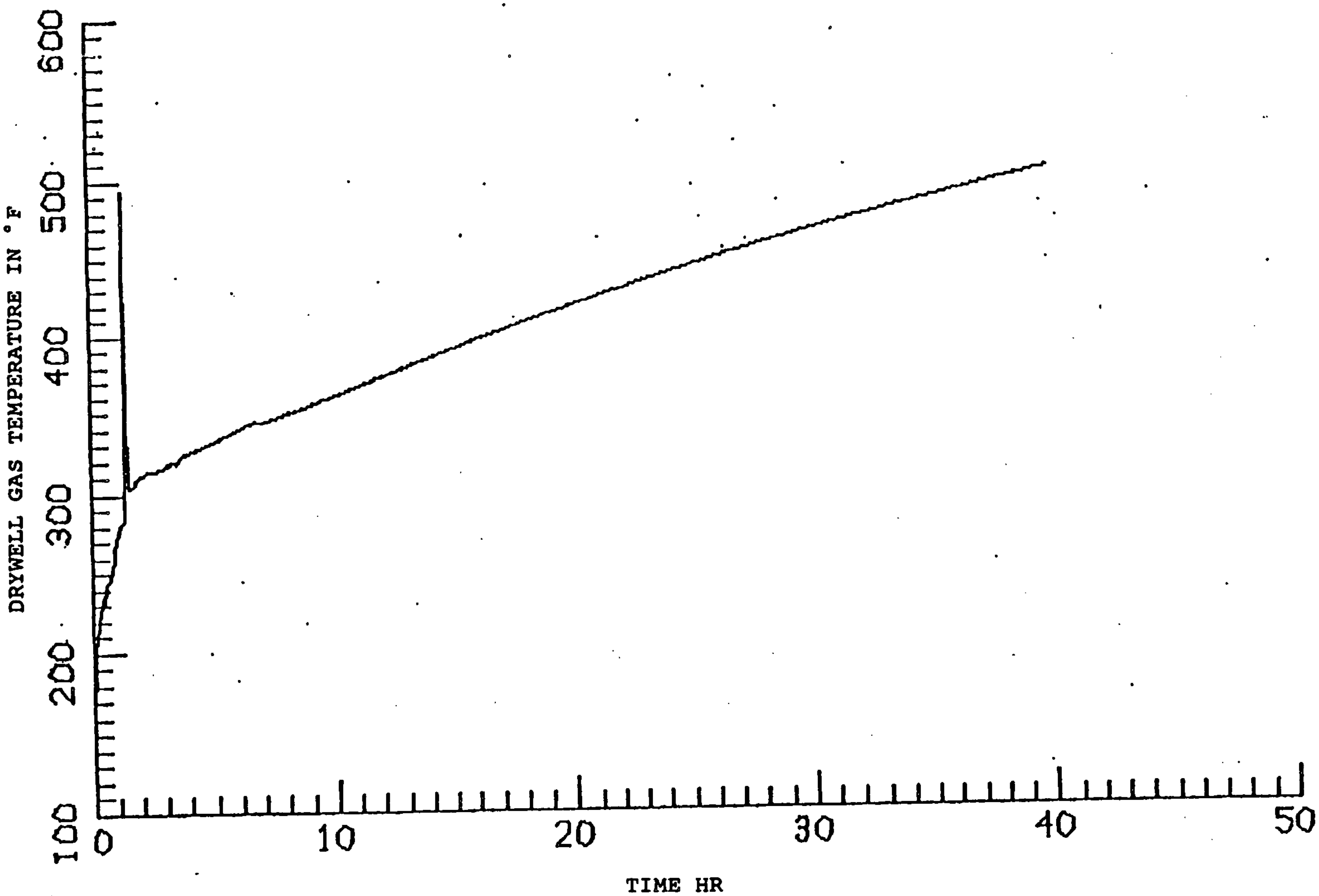


Figure 4.4-5
CONTAINMENT FAILURE PROBABILITY DISTRIBUTION

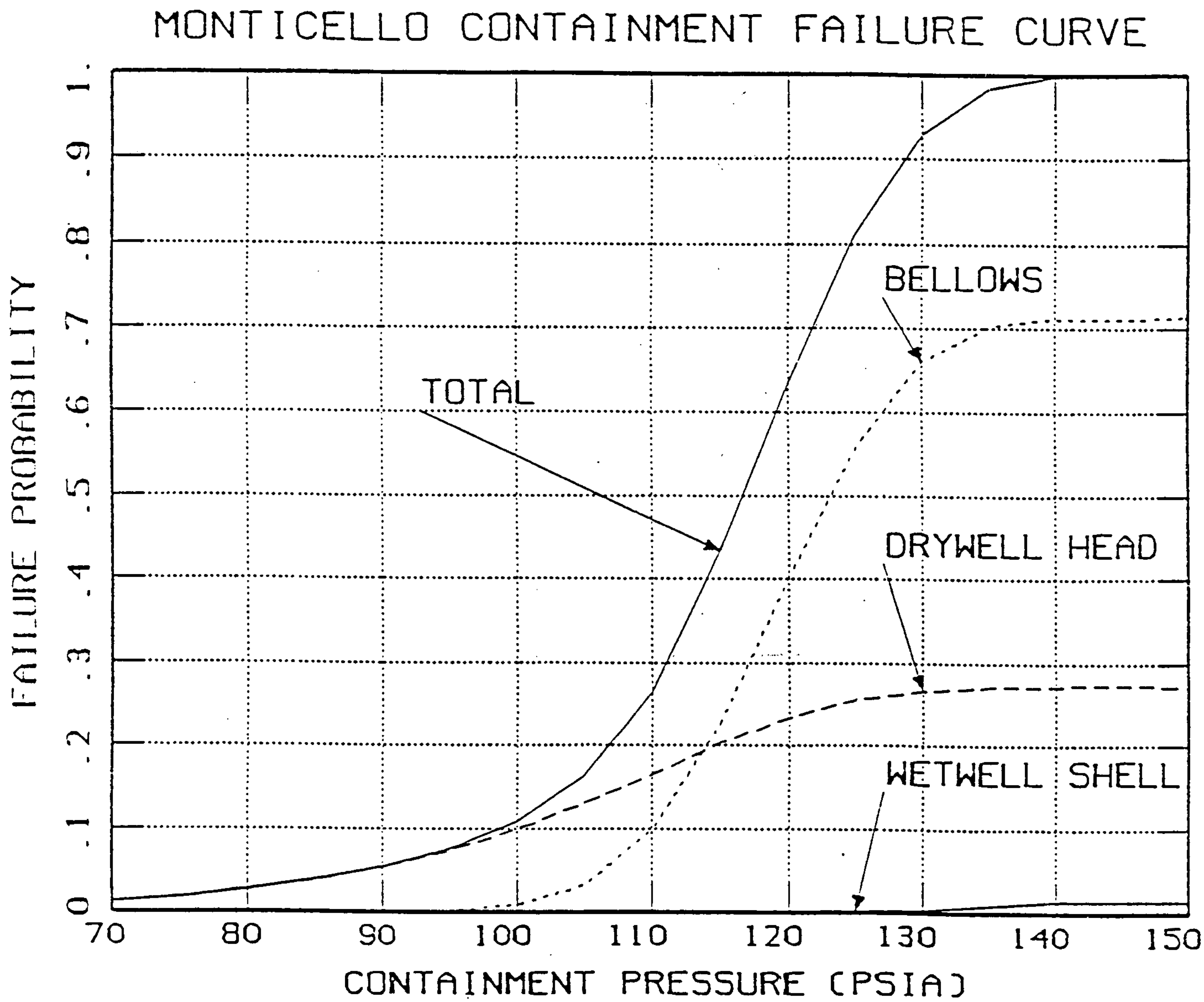


Figure 4.4-6
PLAN VIEW OF DRYWELL SUMPS
(Adapted from NSP Drawing NF-36143, Rev 7)

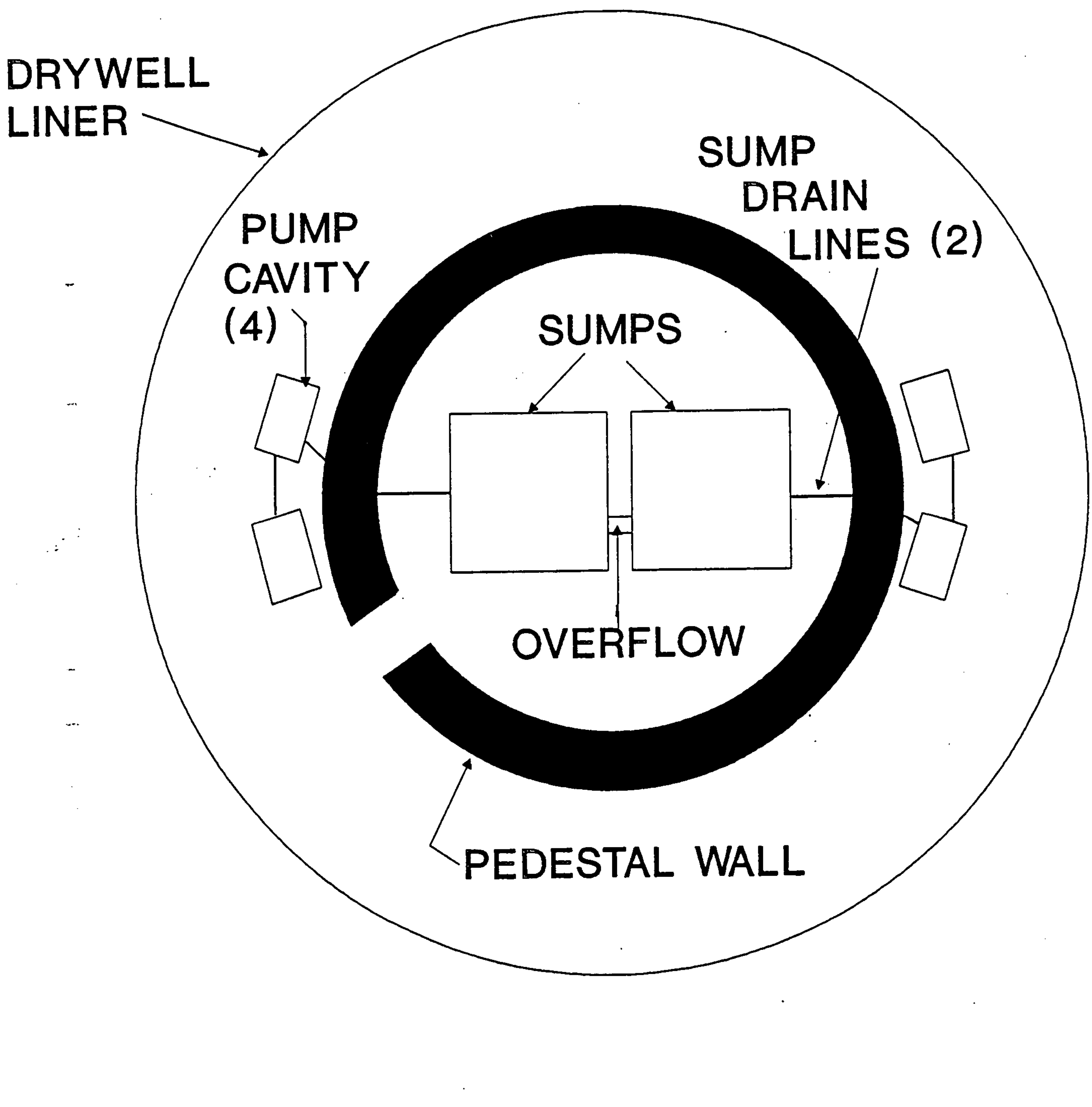
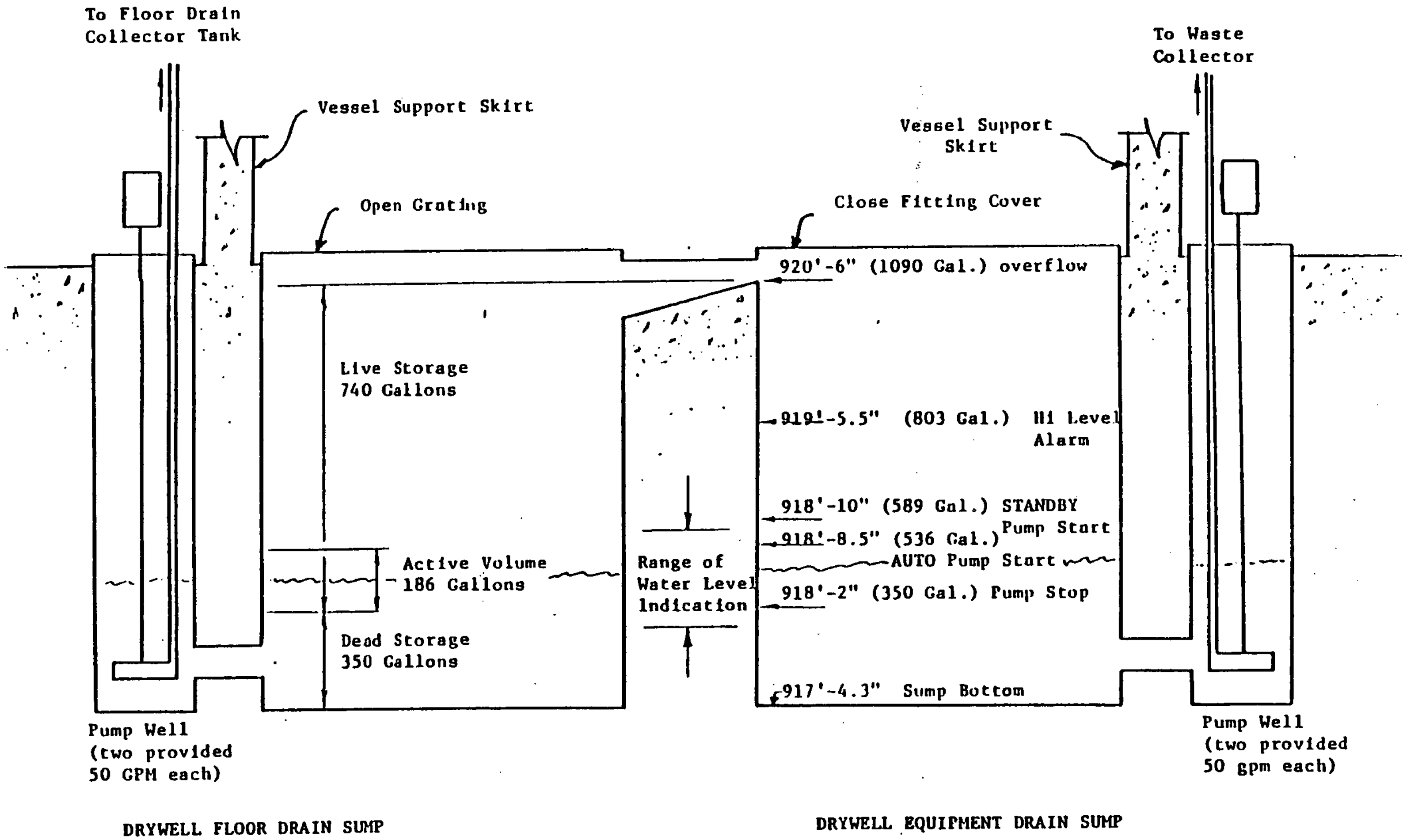


Figure 4.4-7
CROSS-SECTIONAL VIEW OF DRYWELL SUMPS
(From Operations Manual B.7.1-06, Rev 0, p. 10)



4.5 Containment Event Trees

This section discusses the containment event trees used for the level 2 analysis.

4.5.1 Introduction

Containment event trees (CETs) were used to further determine the containment response and ultimately the type of release mode given that a core damage accident has occurred. Different event trees have been prepared to address the various level 1 core damage classes. CETs were prepared for the following damage classes:

1. Class 1A (Figure 4.5-1).

2. Class 1B (Figure 4.5-2).

3. Class 1D (Figure 4.5-3).

4. Class 2 (Figure 4.5-4).

5. Class 3 (Figure 4.5-5).

6. Class 4 (Figure 4.5-6).

No CET was prepared for class 5 because the containment is already bypassed and because this damage class is very low in probability.

The class 2 CET is unique because part of the level 1 analysis was performed using the level 2 CET for the QUV function, injection after containment failure. The class 1A, 1B, 1D, and 3 CETs have basically the same structure. The class 4 and class 2 CETs are similar because they both deal with containment failure location and success of coolant makeup after containment failure.

4.5.2 CET Critical Safety Functions

This section discusses the various critical safety function (CSF) headings used in the CETs and what systems can meet the success criteria. The CSF's and success criteria are summarized in Tables 4.5-1 and 4.5-2.

The headings are:

1. ISO- Containment Isolation Failure. Failure of this event indicates a failure of the containment to isolate resulting in a release of fission products early in the event. The failure to isolate probability was calculated to be 3E-4 per demand based on valve failure rates and the Monticello plant design.

2. N2- Containment Not Inerted. Failure of this event indicates that the containment was not inerted with nitrogen when the event occurred. The result would be a likely detonation of hydrogen in the containment, a failed containment, and early release of fission products. A probability of 0.018 is used for N2 based on Monticello operating history.

3. X- Depressurization. One SRV must remain open for this function to succeed in level 2 analysis. This is different from the success criterion used in level 1, which required two SRVs to remain open. One SRV is sufficient because the decay heat rate is lower for level 2 analysis than for level 1.

   A lower human error rate for initiating ADS is used in level 2 compared to level 1 because more time is available to initiate ADS. This allows successful use of ADS in level 2 if its failure in level 1 was due only to operator error.

4. VSP- Vapor Suppression. This function is challenged when the vessel fails at high pressure, creating a blowdown similar to a LOCA. As for a LOCA, if two vacuum breakers are open at the time of the vessel failure, a failed containment is the assumed result.

5. IV- In-vessel early injection. This function credits injection systems that are not known to have failed in the level 1 analysis. For the accident classes where IV is used, high pressure coolant injection systems have either failed as a part of level 1 or were rendered useless by the initiating event, such as large LOCA. For accident classes 1A and 1B, all low pressure coolant injection systems (LPCI, Core Spray, Condensate, and

RHRSW cross-tied to LPCI) were counted toward IV. This is because these accident classes never challenge low pressure injection until after core damage has occurred. There is no pre-determination that any of these systems are unavailable for level 2.

For accident classes 1D and 3, low pressure coolant makeup is known to have failed as part of level 1. However, RHRSW cross-tie to LPCI was assigned a very high human error rate (0.75) in level 1. A lower human error rate (0.25) for RHRSW cross-tied to LPCI was credited for level 2 because there is additional time to align the cross-tie. This essentially makes successful in-vessel coolant injection a recovery of a human error in level 1.

Success or failure of the IV function is important because if it is successful, the nuclear accident can be terminated with the reactor vessel and containment intact.

6. CS- Containment Sprays/Liner Meltthrough. For the Class 2 and 4 CETs CS represents the use of containment sprays to cool core debris on the floor of the containment. Both RHR and Division I RHRSW cross-tied to the RHR system were credited with a success criteria of one pump running.

   The CSF "CS" was originally included on the Class 1A, 1B, and 1D CETs to model use of containment sprays prior to vessel failure. However, the proper temperature and pressure conditions for initiating containment sprays are never reached, so the function was used to model whether liner meltthrough would occur, given a dry containment floor. Since liner meltthrough is not expected at Monticello (Section 4.4.7), the success branch of this CSF is always followed.

7. ICE- Containment spray or vessel injection from external sources late in the event. The primary reason for this function on the CETs is to analyze the effect of flooding the containment. If the containment is intact, venting the drywell may be required to cope with the flooding. The combined effect of flooding and venting the drywell maintains the

containment intact. Condensate and RHRSW are credited as external water sources. This function occurs before ICR on the CETs because the operator is directed to flood the containment from external sources when the reactor vessel fails and level indication is unavailable.

8. ICR- Containment spray or vessel injection from internal sources late in the event. This function is for the case where RHR or core spray are being used in a recirculation mode. Decay heat removal is challenged if this function is successful. RHR and core spray are recirculation sources.

9. W- Containment Pressure Control. This CSF considers only the RHR functions to remove heat from the containment for the level 2 analysis. This is because the wetwell and drywell vents have a significant impact on source term so they are considered separately. The main condenser is unavailable for level 2 because the main steam lines isolate upon core damage if they did not isolate earlier. This CSF is considered when the core damage event is arrested in-vessel or ex-vessel with an intact containment. Torus cooling and shutdown cooling, if the reactor is intact and depressurized, are possible decay heat removal systems.

10. VWW- Wetwell vent. This CSF is considered when the W function has been unsuccessful. Scrubbing through the suppression pool minimizes the release of fission products and controls containment pressure as well. This is more desirable than drywell venting and would be attempted before drywell venting.

11. VDW- Drywell vent. Assuming the W and VWW CSFs are unsuccessful, or if containment flooding is used, then venting through the drywell is considered. Drywell venting is similar to wetwell venting except suppression pool scrubbing does not occur.

12. SP- Vacuum Breakers. For suppression pool scrubbing to be successful, the torus to drywell vacuum breakers must all remain closed to prevent a bypass of the pool scrubbing effect.

13. CETREC- AC Power Recovery. A variety of recovery factors were used to represent the various probabilities of recovering AC power, either offsite or diesels, depending on the time permitted by the particular accident sequence being considered.

14. QU - High Pressure Coolant Injection. This CSF appears only on the CETs for Class 2 and Class 4 accidents following containment failure. Only feedwater is credited for high pressure coolant makeup after containment failure because high area temperatures in the reactor building could trip isolation switches for the HPCI and RCIC steam lines. Since the CRD pumps are in a relatively exposed location, the hostile environment may disable them and CRD is not credited in the level 2 analysis.

15. V - Low Pressure Coolant Injection. This is exactly the same as function IV for classes 1A or 1B. V is used on the CETs for Classes 2 and 4 and is composed of condensate, core spray, LPCI and RHRSW cross-tied to LPCI.

16. WW- Wetwell Water Space Containment Failure. If the containment fails, consideration is given to the possibility of the wetwell water space failing. This would eliminate ECCS pumps and scrubbing capability with the wetwell. This was determined to be a low probability event because other containment locations would probably fail at a lower pressure.

17. DW- Drywell Failure. If the containment fails, consideration is given to the possibility of the drywell air space failing. Suppression pool scrubbing would not be possible unless the reactor vessel was intact.

    If a containment overpressure condition existed and both the drywell and wetwell waterspace CSFs were successful, then the failure occurs in the wetwell airspace.

Significant assumptions used in the CETs were:

1. If the containment was not inerted with nitrogen, hydrogen gas explosion was assumed to occur and the containment was assumed to fail as a result.

2. If the vessel failed from core damage at high pressure, a blowdown similar to a large LOCA would occur. This would require at least 7 vacuum breakers to be closed. A great deal of uncertainty is associated with this assumption. There may be little water available to produce a blowdown similar to a LOCA. Vapor suppression may not be required.

3. Class 1A core damage events result in vessel breach 2 hours after the initiating event, based on MAAP simulation.

4. Class 1D core damage events result in vessel breach 1.5 hours after the initiating event, based on MAAP simulation.

5. Class 1B core damage events result in vessel breach at 2 hours after the initiating event with loss of high pressure injection, at 5 hours with a SORV and HPCI success, and at 8 hours if everything works properly until the batteries deplete. These assumptions are supported by MAAP analysis.

6. Class 3 small LOCA QUX events responded like transients.

7. Class 3 medium and large LOCAs do not need to take credit for depressurization in the CETs because the reactor will depressurize from the break.

8. CRD is not considered as an injection source with core damage because of the increased heat generation possible in a core damage event from metal water reactions.

9. Containment failure from ATWS and loss of DHR result in large failures of the containment.

TABLE 4.5-1
DEFINITION OF CRITICAL SAFETY FUNCTIONS FOR
LEVEL 2 ANALYSIS

CSF CODE	DESCRIPTION	LEVEL 1 ACCIDENT CLASS	SYSTEMS CREDITED
CS	Containment Spray	2 & 4	1) RHR to Drywell Sprays 2) RHRSW crosstied to RHR sprays
ICE	Injection to the Containment from External Sources	All	1) Condensate 2) RHRSW crosstied to LPCI injection 3) RHRSW to Drywell Sprays
ICR	Injection to the Containment from Recirculation Sources (i.e. suppression pool)	All	1) LPCI 2) Core Spray 3) RHR to Drywell Sprays
IV	In vessel coolant injection prior to vessel failure	1A & 1B	1) RHRSW crosstied to LPCI 2) Core Spray 3) LPCI 4) Condensate
		1D & 3	1) RHRSW crosstied to LPCI
QU	High Pressure Coolant Makeup to the Reactor	All	Feedwater
V	Low Pressure Coolant Makeup to the Reactor	2 & 4	1) RHRSW crosstied to LPCI 2) LPCI 3) Core Spray 4) Condensate
W	Containment Pressure Control (excluding vents)	1A, 1B & 3	Torus Cooling
		1D	1) Torus Cooling 2) Shutdown Cooling

TABLE 4.5-1 (Continued)
DEFINITION OF CRITICAL SAFETY FUNCTIONS FOR
LEVEL 2 ANALYSIS

CSF CODE	DESCRIPTION	LEVEL 1 ACCIDENT CLASS	SYSTEMS CREDITED
VDW	Drywell Vent	All	Drywell Vent
VWW	Wetwell Vent	All	Wetwell Vent
VSP	Vapor Suppression	All	≤ 1 vacuum breakers stuck open
SP	Suppression Pool Scrubbing	All	No vacuum breakers stuck open
X	Reactor Depressurization	All	ADS with at least 1 SRV working
------------MISCELLANEOUS POINT ESTIMATES------------			
DW	Conditional probability that failure location will be in the drywell if the containment ultimate pressure is reached	All	0.50
WW	Conditional probability that failure location will be in the wetwell below the waterline if the containment ultimate pressure is reached	All	0.005
ISO	Containment isolation failure	All	3.0E-4
N_2	Containment not inerted	All	0.018
CETREC	Conditional probability that AC power will not be recovered prior to containment failure, given that station blackout conditions existed at core melt.	1B	Various, depending on timing of accident sequence

TABLE 4.5-2
LEVEL 2 SUCCESS CRITERIA

SYSTEM	SUCCESS CRITERIA
RHR Drywell Sprays	One RHR pump running, providing water to drywell spray sparger.
RHRSW to Drywell Sprays	One Division I RHRSW pump running, crosstied to RHR, and supplying drywell spray sparger. The human error failure rate to crosstie RHRSW to RHR is lower for level 2 analysis than for level 1 because more time is available.
Condensate	One condensate pump running, drawing a suction from the hotwell and supplying water to the reactor. Since the condensate storage tanks would be depleted after 24 hours or so, a human error failure to supply alternate water makeup to the hotwell is assessed for the condensate system in level 2 (See assumption 1 in Section 4.2.2).
RHRSW to LPCI Injection	One Division I RHRSW pump running, crosstied to RHR and injecting to the reactor via the LPCI injection line. The human error failure rate to crosstie RHRSW to RHR is lower for the level 2 analysis than for level 1 because more time is available.
LPCI	One RHR pump running, injecting to the reactor.
Core Spray	One core spray pump running, injecting water to the reactor.
Feedwater	One feedwater pump running with one condensate pump running to provide adequate suction pressure. The hotwell makeup requirements for condensate also apply to feedwater.
Torus Cooling	One RHR pump running, drawing a suction from the suppression pool, pumping the water through the RHR heat exchanger and returning it to the suppression pool. One RHRSW pump in the same division pumping cooling water through the RHR heat exchanger.
Shutdown Cooling	(Same as success criteria for torus cooling except substitute the word "reactor" for "suppression pool".)

TABLE 4.5-2 (Continued)
LEVEL 2 SUCCESS CRITERIA

SYSTEM	SUCCESS CRITERIA
Drywell Vent	Drywell vent valves AO-2386 and AO-2387 open. Ductwork downstream of the valves is assumed to rupture, releasing containment atmosphere to the reactor building.
Wetwell Vent	Wetwell vent valves AO-2383 and AO-2896 open. Ductwork downstream of these valves is assumed to rupture, releasing containment atmosphere to the reactor building.
Vapor Suppression	At most, one vacuum breaker is stuck open.
Suppression Pool Scrubbing	No vacuum breakers are stuck open.
ADS	At least one Safety Relief Valve is held open to depressurize the reactor. Also, the human error rate to initiate ADS is lower for level 2 than for level 1 because of the additional time available to perform this action.

FIGURE 4.5-1
CET for Class 1A

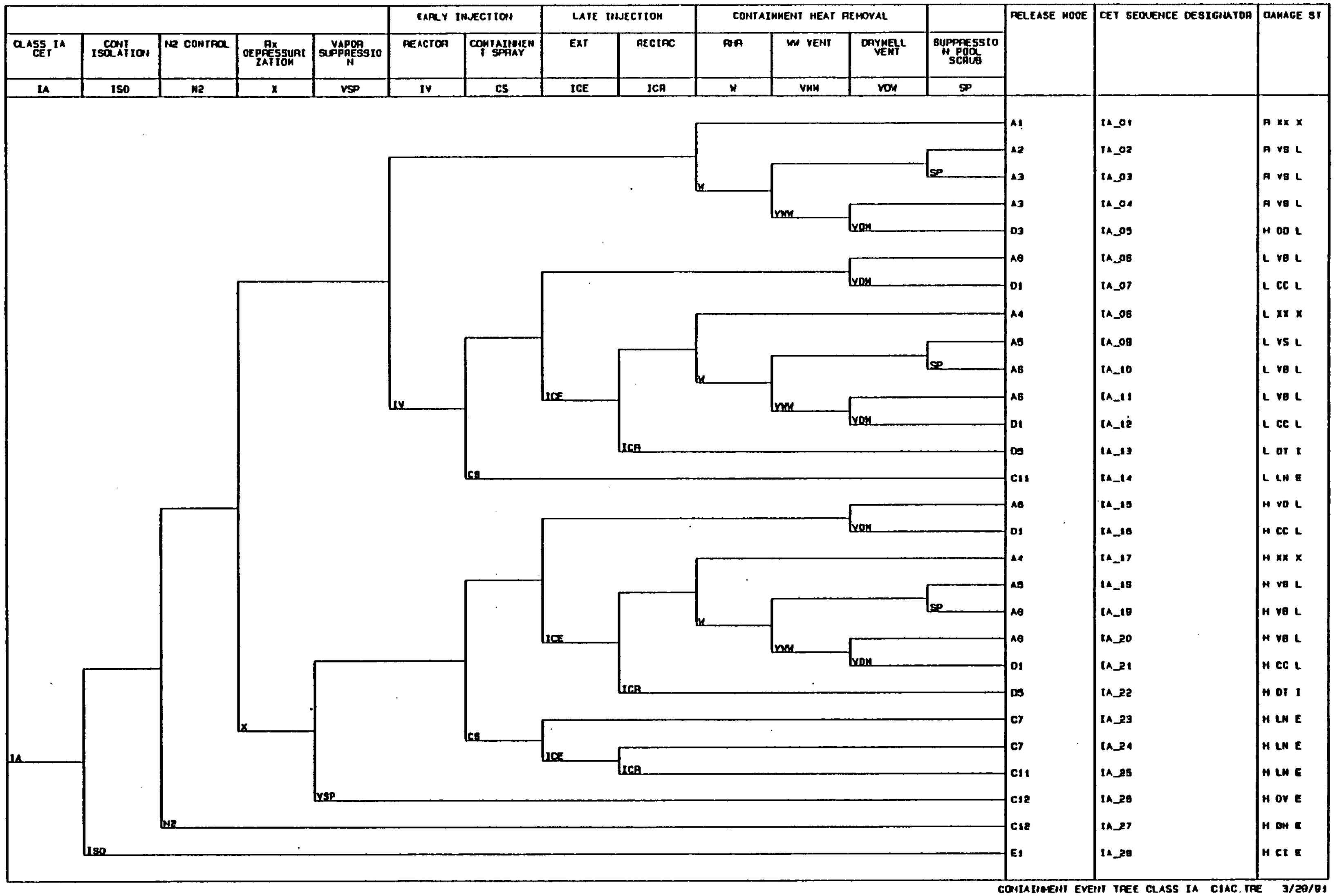


FIGURE 4.5-2
CET for Class 1B

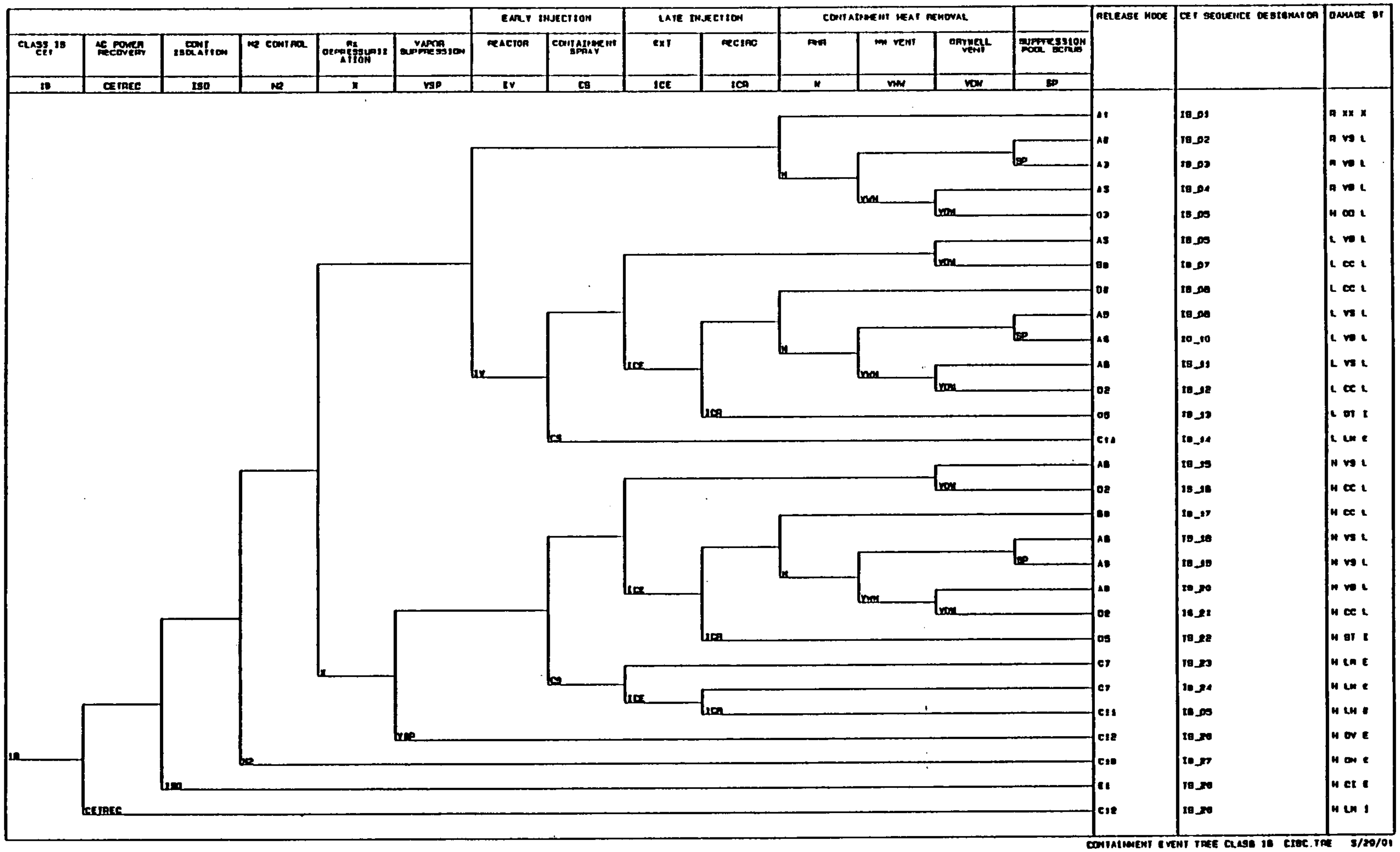
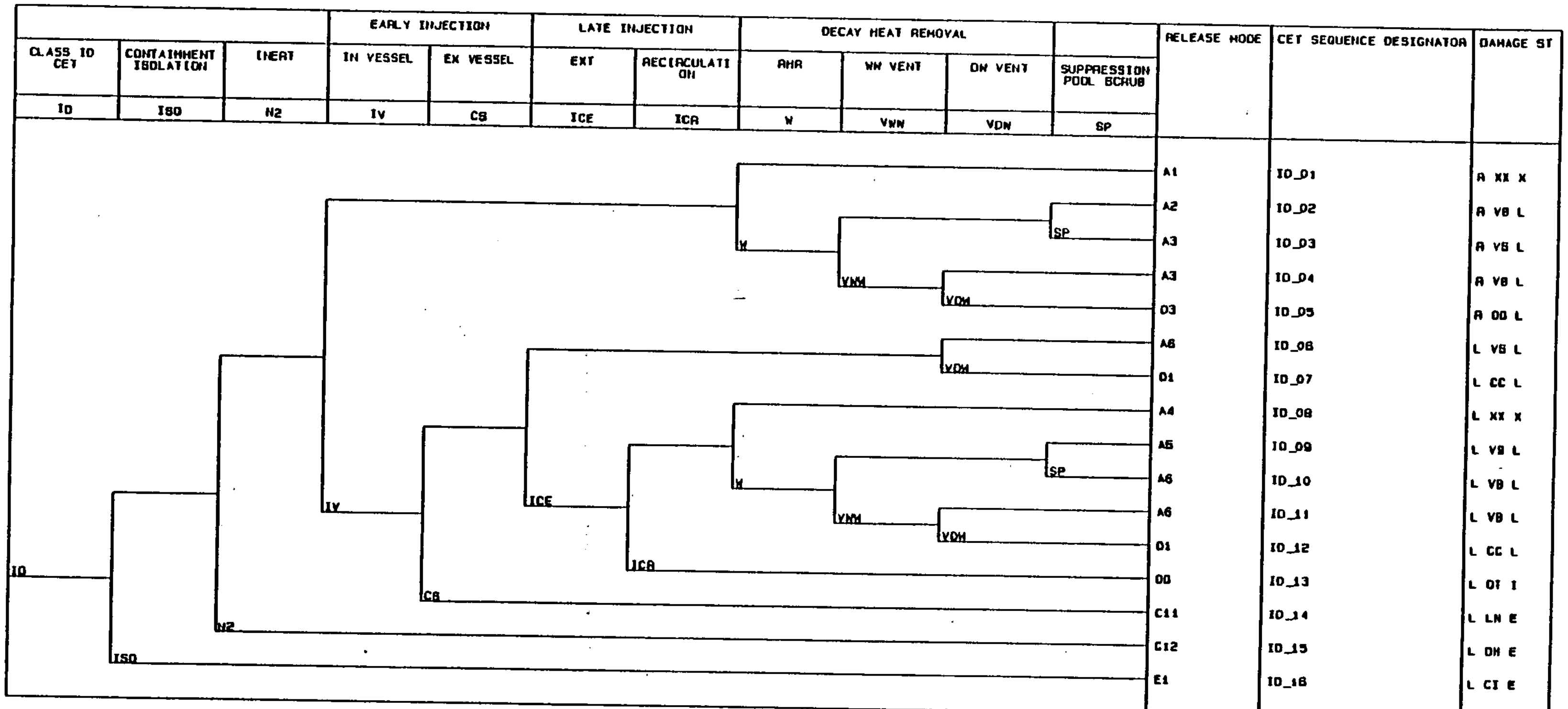


FIGURE 4.5-3
CET for Class 1D



CONTAINMENT EVENT TREE CLASS ID CIDC.TRE 3/29/91

FIGURE 4.5-4
CET for Class 2

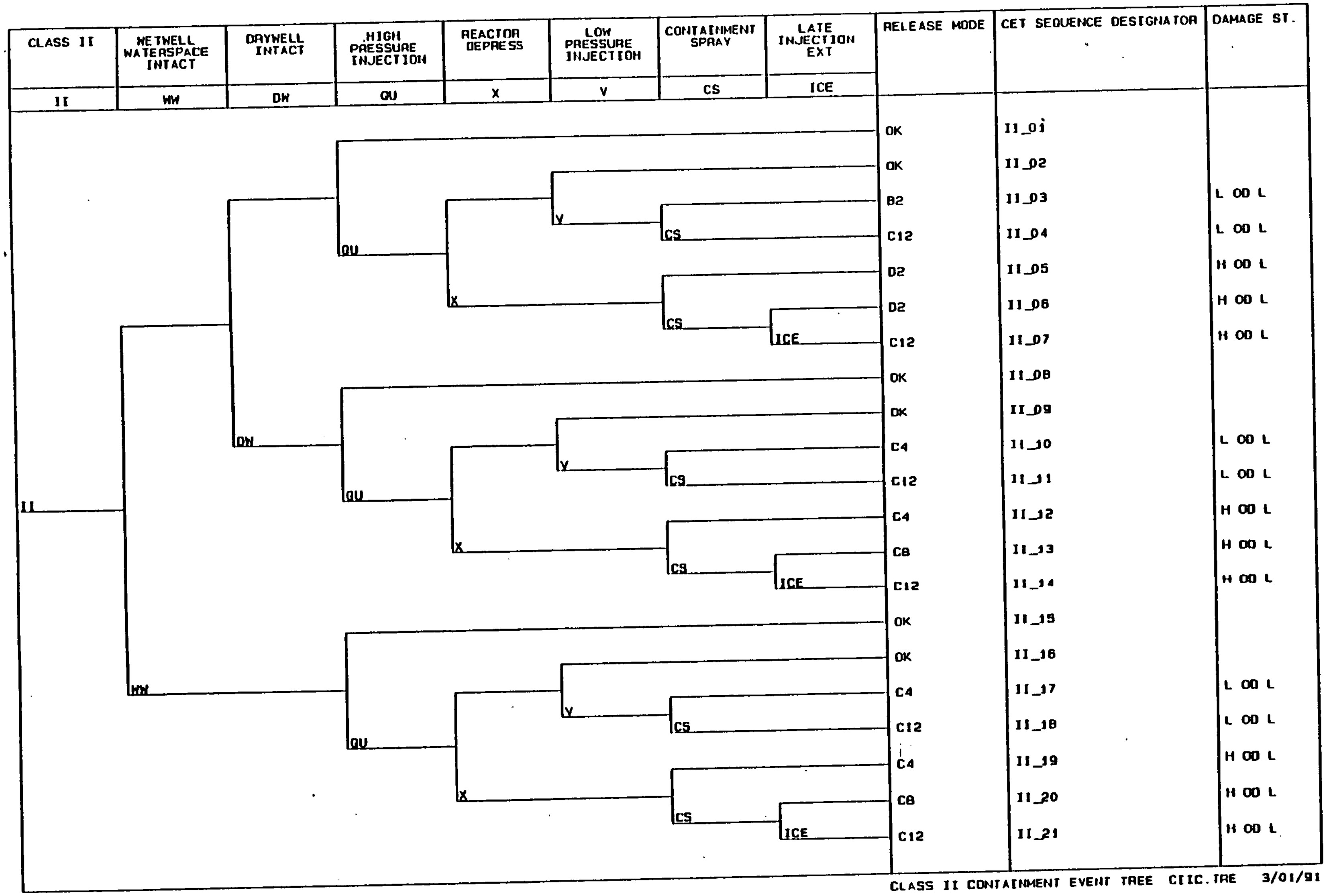


FIGURE 4.5-5
CET for Class 3

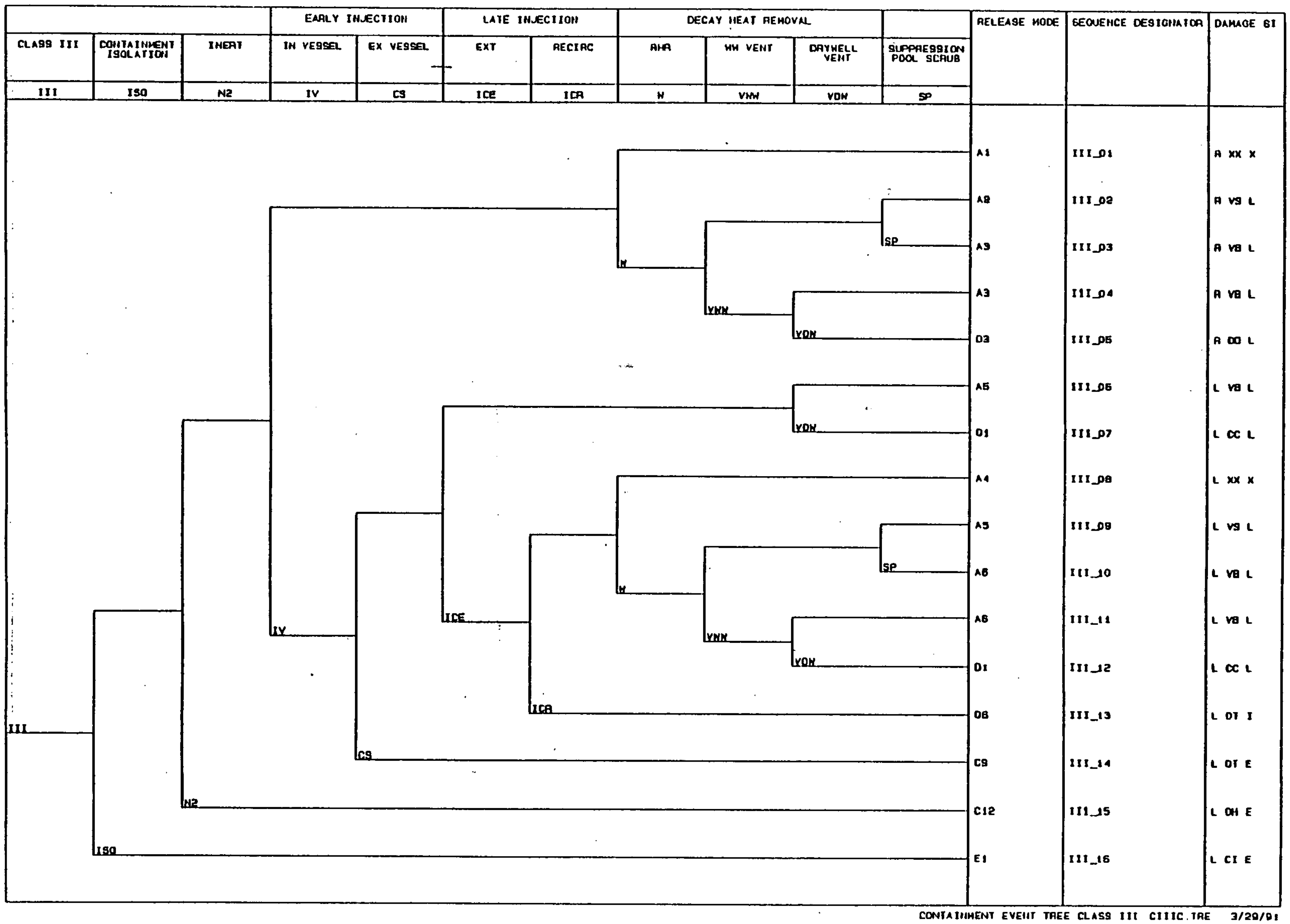
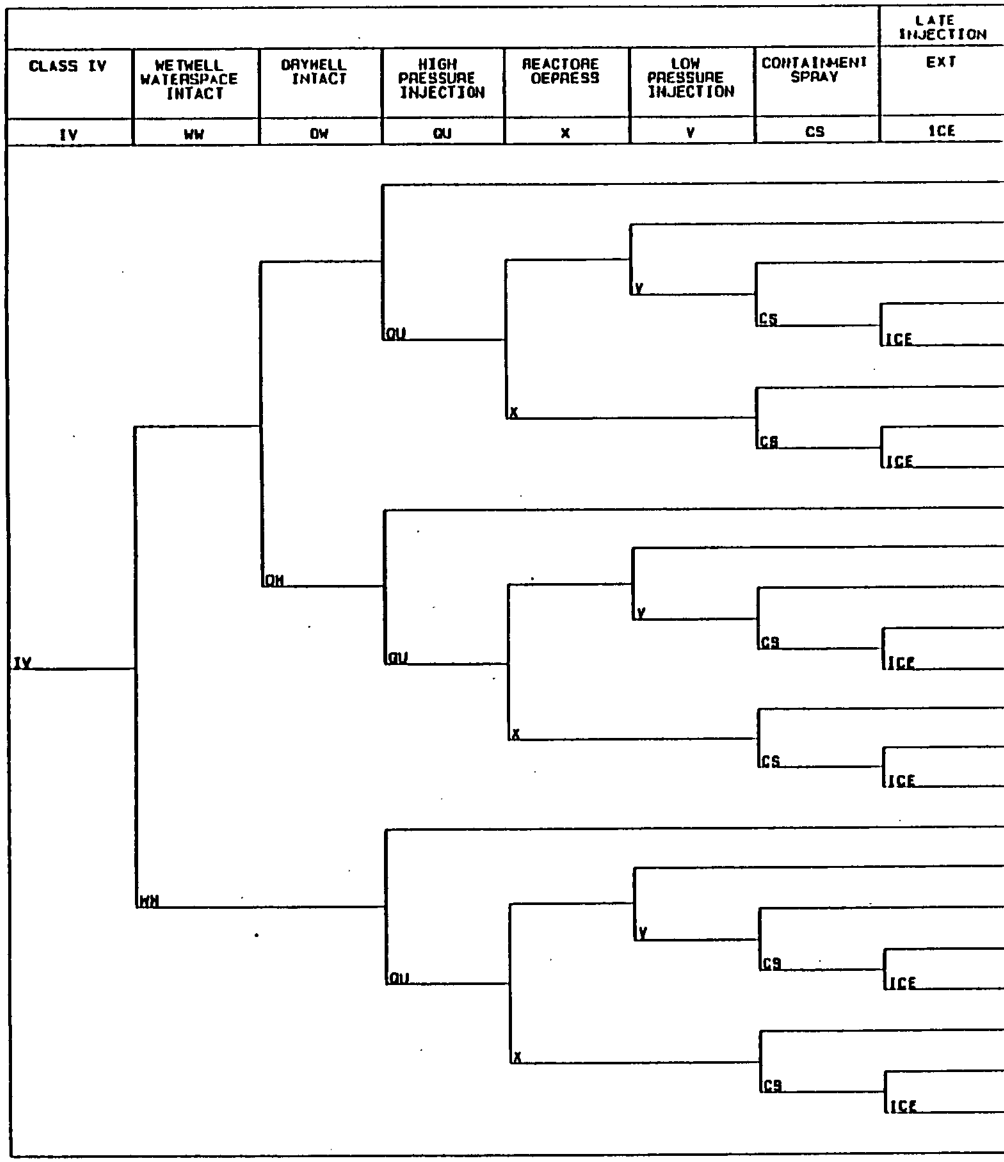


FIGURE 4.5-6
CET for Class 4



RELEASE MODE	CET SEQUENCE DESIGNATOR	DAMAGE ST.
B2	IV_01	R DA E
B2	IV_02	R DA E
B2	IV_03	L DA E
B2	IV_04	L DA E
C12	IV_05	L DA E
B2	IV_06	H DA E
B2	IV_07	H DA E
C12	IV_08	H DA E
C8	IV_09	R DA E
C8	IV_10	R DA E
C4	IV_11	H DA E
C8	IV_12	H DA E
C12	IV_13	H DA E
C4	IV_14	H DA E
C8	IV_15	H DA E
C12	IV_16	H DA E
C8	IV_17	R DA E
C8	IV_18	R DA E
C4	IV_19	L DA E
C8	IV_20	L DA E
C12	IV_21	L DA E
C4	IV_22	H DA E
C8	IV_23	H DA E
C12	IV_24	H DA E

CLASS IV CONTAINMENT EVENT TREE CIVC.TRE 3/01/91

4.6 Results of Level 2 Sequence Quantification

The results of the level 2 Sequence Quantification are presented first for the internally initiated nuclear accidents and then for accidents arising from the external initiator of internal flooding. So far, the only external initiator that has been analyzed for Monticello is internal flooding.

It is appropriate to segregate the findings from analyzing internally initiated accidents from those of externally initiated accidents. This is because the "external" initiators and the responses of the plant to them are less understood than the internal initiators. The approach in analysis is a scoping study seeking broad insights rather than calculational accuracy. The results are quite conservative compared to those of internal events analysis which uses a "best estimate" approach. It is therefore not considered appropriate to fully integrate the findings of the internally initiated events analysis with the less accurate and generally conservative findings of the flooding analysis.

4.6.1 Releases from Internally Initiated Nuclear Accidents

4.6.1.1 Release Mode

As Figure 4.6-1 shows, release of radionuclides is entirely avoided 60% of the time core damage results from an internal initiator. The containment is maintained intact by using either the wetwell or drywell vent 18% of the time, and containment failure occurs 21% of the time.

Note that the drywell vent is used more frequently than the wetwell vent. This may confuse some readers as the wetwell vent is preferred because fission products are retained by scrubbing the release through the suppression pool. Many of the vent sequences occur as a result of containment flooding. In those cases, the wetwell vent would be submerged, so the drywell vent must be used to relieve pressure as the containment is filled with water. Some fission product scrubbing could therefore be expected in most of the drywell vent sequences due to the presence of water overlying at least part of the core debris during venting.

When containment failure occurs, it often precedes or occurs simultaneously with vessel breach. This is labeled on the pie chart as "Ctmt Fails </= RPV". This portion of releases is dominated by ATWS where the containment fails by overpressure prior to core damage. Thus, when the vessel fails, the fission products are released directly into the reactor building without any substantial benefit from fission product deposition mechanisms in the containment.

4.6.1.2 Reactor Vessel Failure Pressure

The pie chart in Figure 4.6-2 shows releases distributed by the pressure of the reactor at vessel failure. A key assumption of this analysis is reflected in this distribution; i.e. core damage can be arrested in vessel if coolant injection can be restored prior to vessel breach. This assumption is supported by the experience of the Three Mile Island nuclear accident where the molten core was resolidified before vessel breach. Arresting core damage within the reactor vessel represents 72% of the calculated core damage frequency from internal initiators. Note that high pressure vessel failure dominates the remaining sequences. This is driven by long-term station blackout where the HPCI and RCIC systems eventually fail because of battery depletion and ADS cannot be used for the same reason. The vessel is maintained at high pressure until failure.

4.6.1.3 Containment Failure Mechanisms

The pie chart of Figure 4.6-3 shows the distribution of containment failure mechanisms for the level 2 results. Like the first pie chart, 60% is No Release and 18% is vented release. The remaining portions show small contributions to containment failure by hydrogen combustion, overtemperature, and molten core-concrete interaction (overpressure by non-condensible gas generation).

The reader might wonder why containment overpressure is such a dominant failure mechanism when the Monticello containment is equipped with vents. The answer is that in many cases, the support systems for the vents are disabled by the initiating event or by subsequent support system failures. The current vents are disabled by loss of offsite power, loss of instrument air pressure, or loss of service water.

Efforts are underway to ensure that the hardened vent planned for Monticello will use reliable support systems, and it should substantially reduce the calculated fraction of containment overpressure failures.

4.6.1.4 Release Timing

The next chart (Figure 4.6-4) shows the timing of radionuclide releases. This is important because the release timing figures substantially in the character of the source term. Late releases allow mechanisms for fission product deposition to occur and allow short-lived isotopes to decay. The early releases are dominated by ATWS, where the containment fails prior to vessel breach. The late releases are typified by long-term station blackout, where the containment maintains its integrity until eventual overpressurization.

4.6.2 Releases from Nuclear Accidents Initiated by Internal Flooding

4.6.2.1 Release Mode

The chart in Figure 4.6-5 shows that internal floods more often lead to delayed containment failures than internal initiators. This is because flooding sequences do not include the impact of ATWS which was evident in the internal events sequences as dominating the early containment failure modes.

4.6.2.2 Vessel Failure Pressure

The chart for vessel failure pressure of nuclear accidents arising from internal flooding appears in Figure 4.6-6. There is nothing noteworthy about the distribution of vessel failure pressures. This chart is included merely for completeness.

4.6.2.3 Containment Failure Mode

One observes from Figure 4.6-7 that internal flooding initiators are more likely to lead to core-concrete interaction and containment overpressure failure modes, while the wetwell and drywell vent releases are not significant enough to appear on the pie chart. All of this can be explained by the susceptibility of the current vents to support system failures. Internal flooding is dominated by breaks of service water piping in various plant locations. The loss of service water consequently disables the instrument air compressors. The vent is not able to operate without instrument air pressure, so situations which pressurize the containment, such as the non-condensible gases generated by the molten core attack of concrete, lead to overpressurization instead of a vented release.

The modification discussed in section 4.6.1.3 will be as effective against loss of venting capabilities caused by flooding as those caused by internal initiators.

4.6.2.4 Release Timing

Figure 4.6-8 shows flood-induced nuclear accidents distributed by release timing. Like the release mode chart, this shows that late releases dominate because the influence of ATWS is not present in the flood analysis.

FIGURE 4.6-1

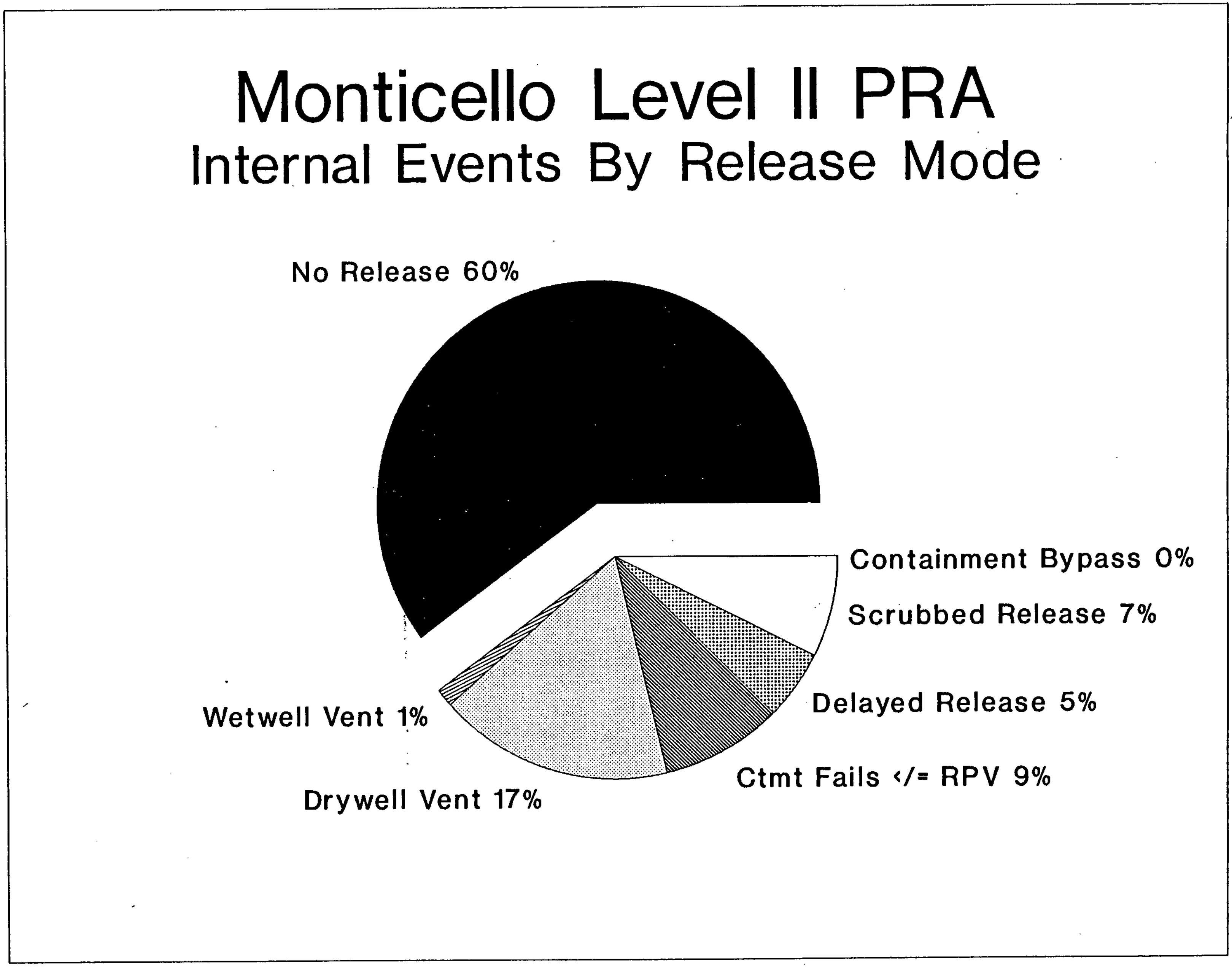

Monticello Level II PRA
Internal Events By Release Mode
No Release 60%
Containment Bypass 0%
Scrubbed Release 7%
Delayed Release 5%
Ctmt Fails </= RPV 9%
Drywell Vent 17%
Wetwell Vent 1%

FIGURE 4.6-2

Monticello Level II PRA

Internal Events By Vessel Failure Press.

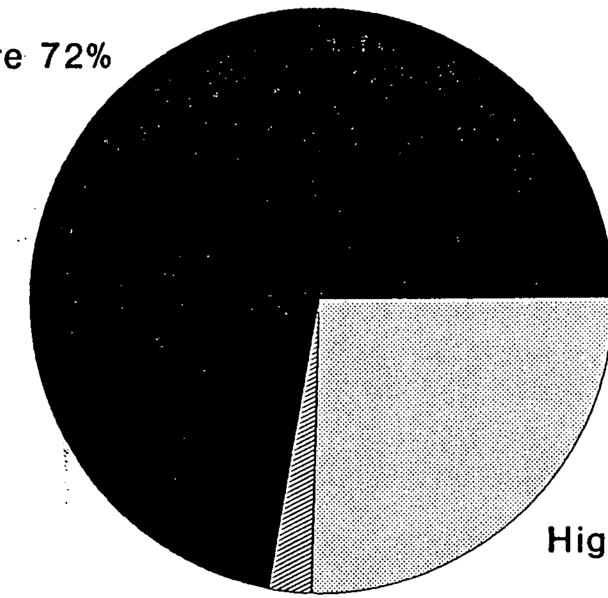


FIGURE 4.6-3

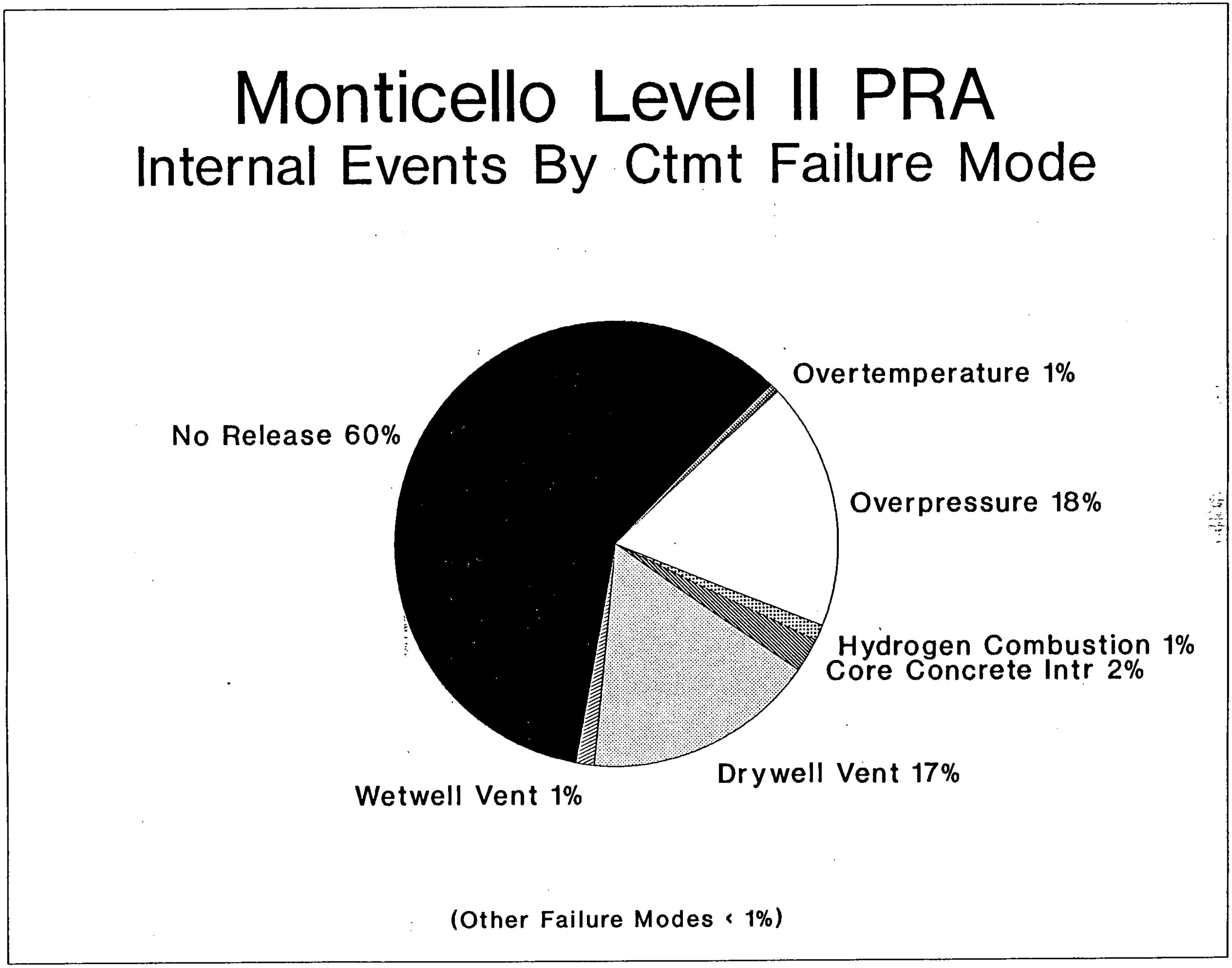

Monticello Level II PRA
Internal Events By Ctmt Failure Mode
Overtemperature 1%
No Release 60%
Overpressure 18%
Hydrogen Combustion 1%
Core Concrete Intr 2%
Drywell Vent 17%
Wetwell Vent 1%
(Other Failure Modes < 1%)

FIGURE 4.6-4

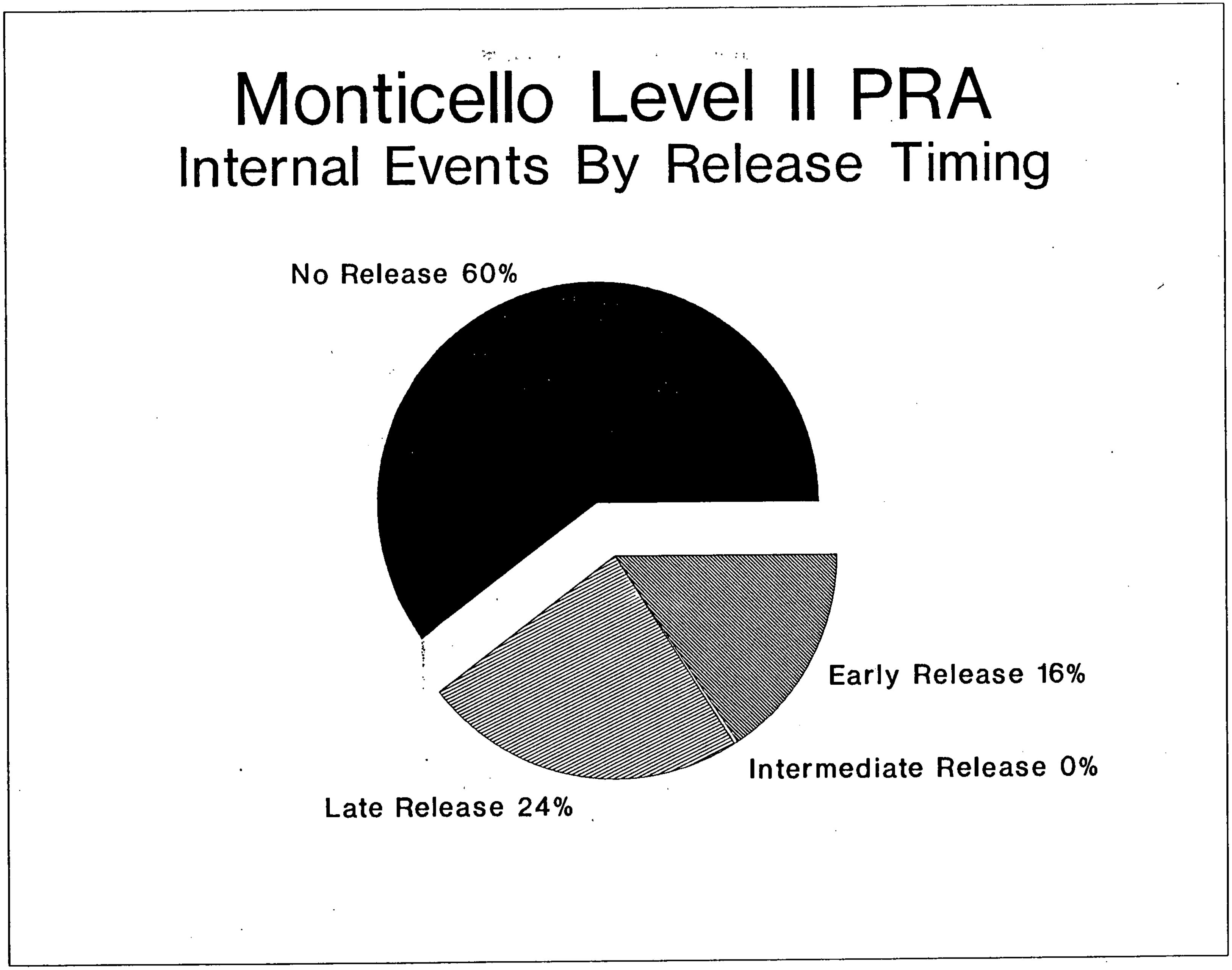

Monticello Level II PRA
Internal Events By Release Timing
No Release 60%
Early Release 16%
Intermediate Release 0%
Late Release 24%

FIGURE 4.6-5

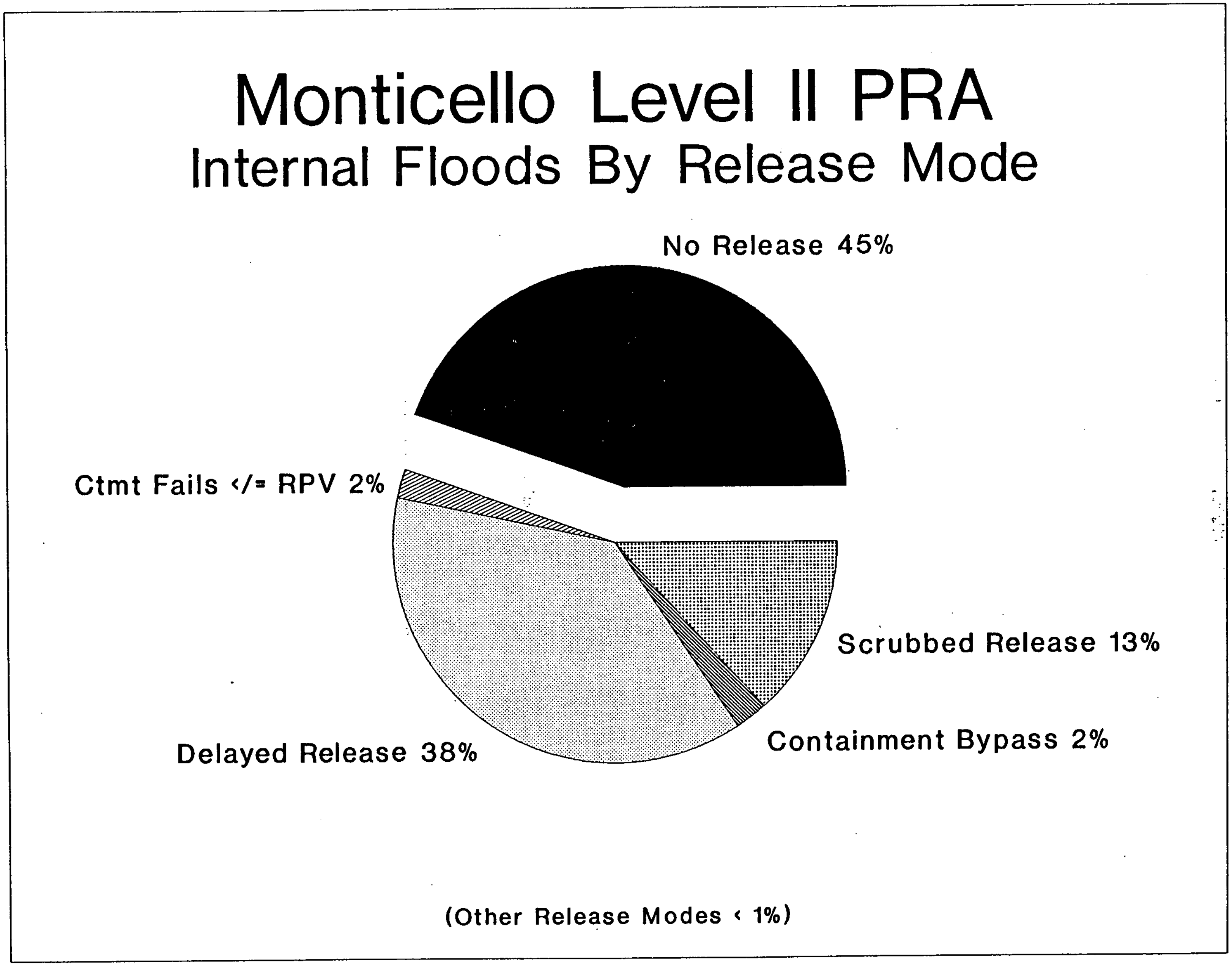
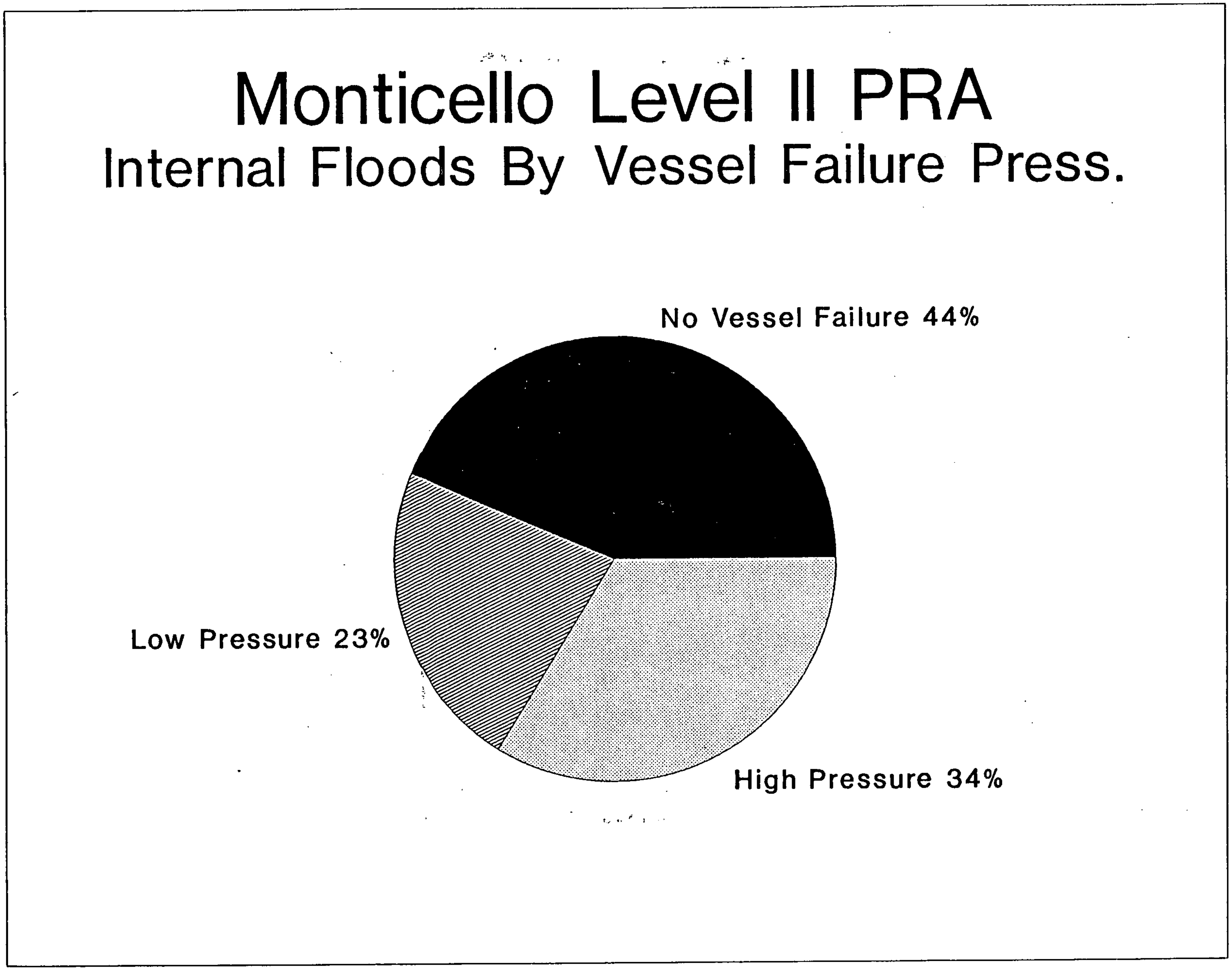

Monticello Level II PRA
Internal Floods By Release Mode
No Release 45%
Ctmt Fails </= RPV 2%
Scrubbed Release 13%
Containment Bypass 2%
Delayed Release 38%
(Other Release Modes < 1%)

FIGURE 4.6-6



Monticello Level II PRA
Internal Floods By Vessel Failure Press.
No Vessel Failure 44%
Low Pressure 23%
High Pressure 34%

FIGURE 4.6-7

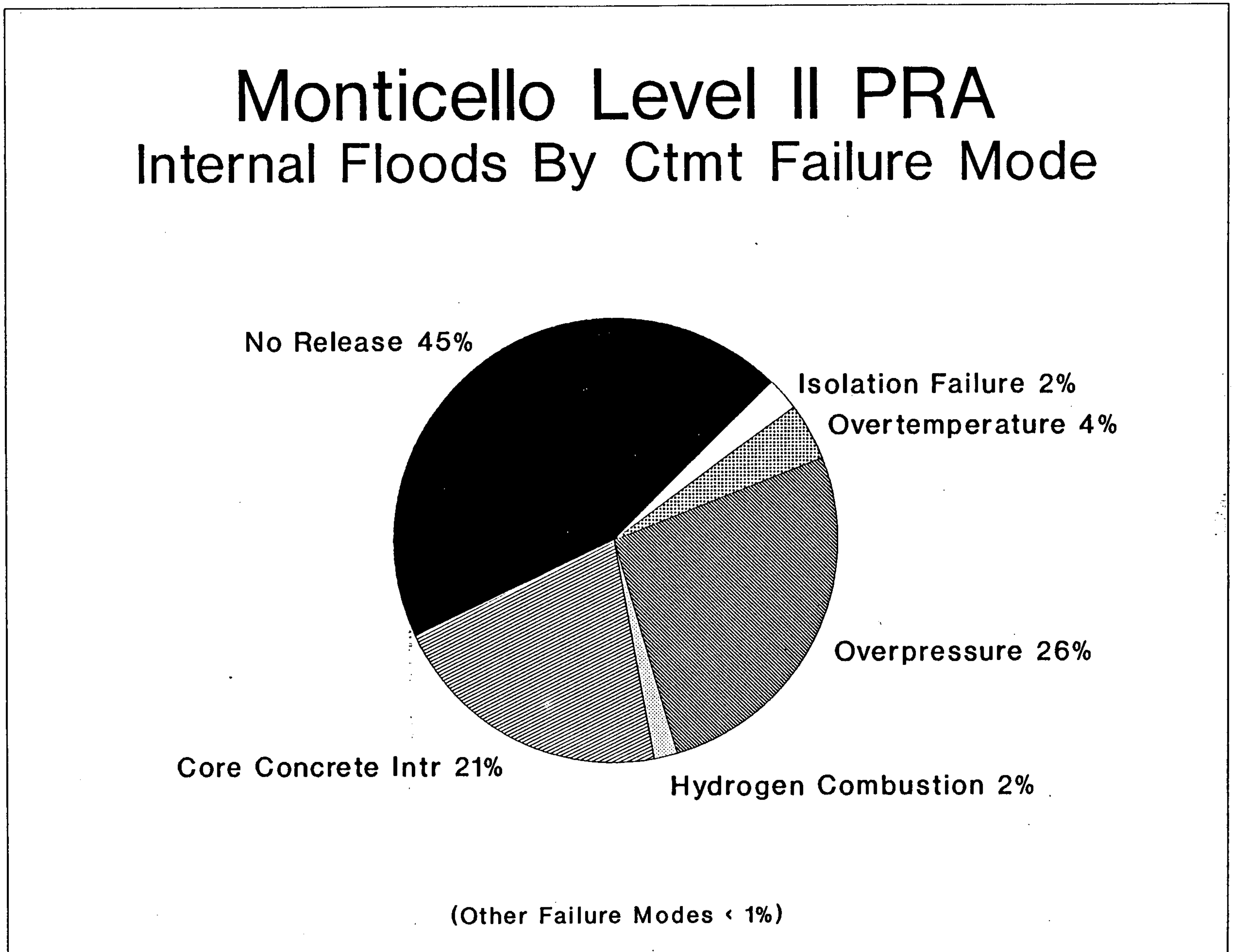
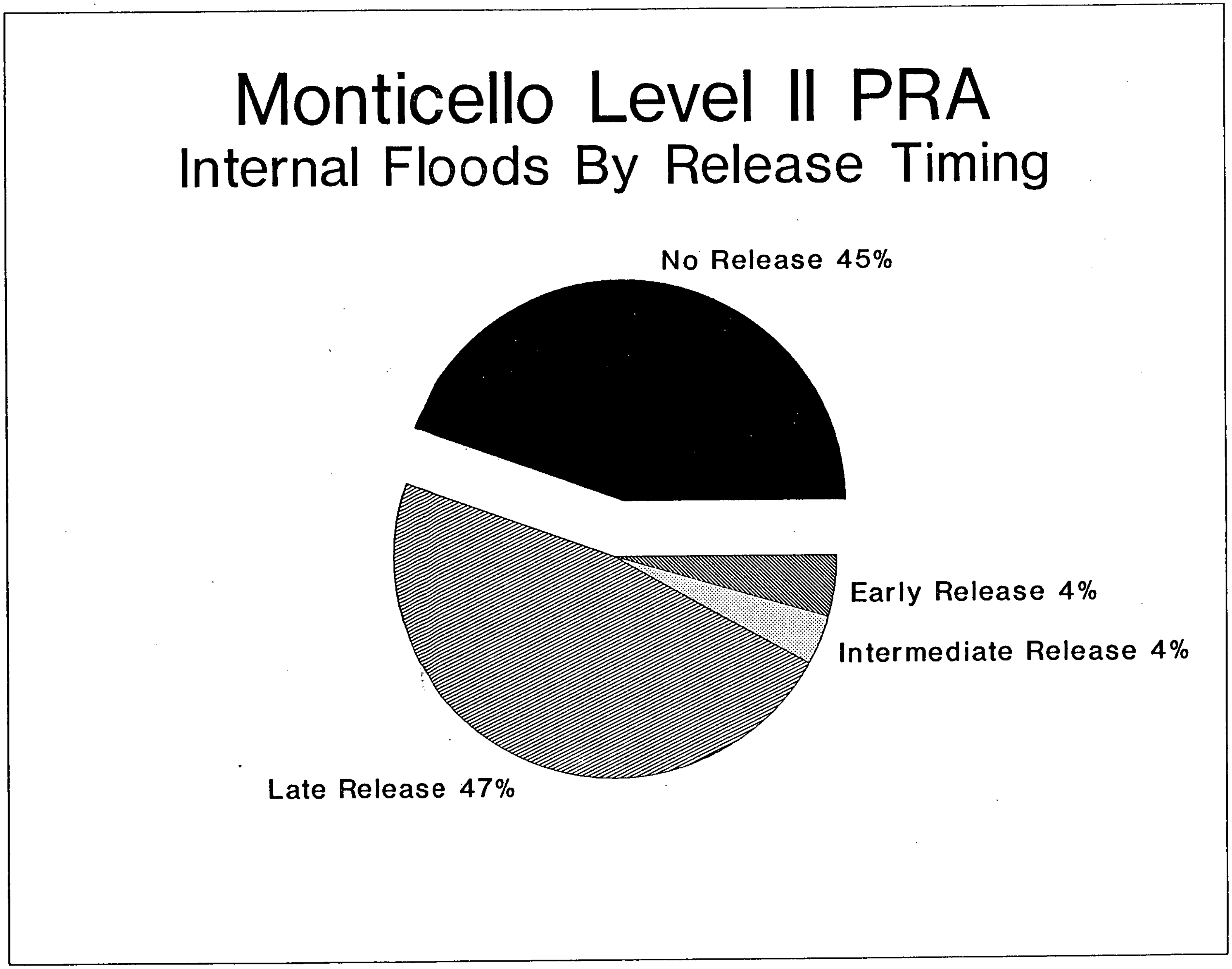

Monticello Level II PRA
Internal Floods By Ctmt Failure Mode
No Release 45%
Isolation Failure 2%
Overtemperature 4%
Overpressure 26%
Hydrogen Combustion 2%
Core Concrete Intr 21%
(Other Failure Modes < 1%)

FIGURE 4.6-8


Monticello Level II PRA
Internal Floods By Release Timing
No Release 45%
Early Release 4%
Intermediate Release 4%
Late Release 47%

4.7 Source Term

4.7.1 Determination of Radionuclide Fractions Released

MAAP analysis was conducted for selected release modes to produce plots of fission product fractions released as a function of time following accident initiation. These plots were produced for all twelve fission product groups modeled in MAAP and the results are tabulated in Table 4.7-1. The results given here are for fission product releases from the containment. In reality, some fission product retention in the reactor building is expected so the release of radionuclides to the environment would probably be somewhat less severe than shown in this report.

Other release modes not explicitly modeled were assigned the same source term as the modeled release mode that best approximates it.

4.7.2 Assessment of Source Term Importance

The twelve fission product groups can be binned to gain understanding of the significance of the various release modes from the standpoint of radiological hazard. Three bins were chosen.

Noble gases and inert aerosols. (fission product group 1): Although a large fraction of this group is released in any containment failure scenario, these are not important from a hazard standpoint because of their chemically inert nature.

Volatile compounds. (fission product groups 2, 3, 6, and 11): These pose the greatest hazard since they contain the important caesium, iodine and tellurium isotopes.

Non-volatiles. (fission product groups 4, 5, 7, 8, 9, 10, and 12): These are not ordinarily released in any large amount.

The significance of a release of radionuclides is best exemplified by the amount of volatiles released. There are four categories of release defined largely by the amount of volatile radionuclides released.

This table shows the release categories used in this report. They are based primarily on the percentage of volatile radionuclides released because they pose the greatest hazard.

	PERCENTAGE OF RADIONUCLIDES RELEASED		
CATEGORY	NOBLES	VOLATILES	NON-VOLATILES
0	0	0	0
1	≤ 100%	< 2%	< 0.1%
2	≤ 100%	2-10%	< 2%
3	≤ 100%	> 10%	> 2%

These categories are used to produce the charts shown in Tables 4.7-2 and 4.7-3 and Figures 4.7-1 and 4.7-2.

Not every possible release mode was modeled in the source term analysis. Those selected were chosen because the expected release is large compared to the other release modes or because the frequency of occurrence is large compared to other release modes. The release modes which were not explicitly modeled were assigned the same source term as the modeled release mode which best approximates it. This is shown on Tables 4.7-2 and 4.7-3.

4.7.3 Assignment of Release Modes

Not all release modes were modeled using MAAP. Those that were not modeled were assigned the same release category as another release mode that was modeled. The rationale for assigning the unmodeled release modes to a modeled release mode is given in the following paragraphs. These are shown in the "Bounding for These Other Release Modes" column in Tables 4.7-2 and 4.7-3.

Release mode A4 bounds A1 because the only difference between these is that the core debris remains within the reactor vessel for mode A1 but exits to the drywell floor for mode A4. The containment remains intact in either case. Release mode A6 represents controlled, unscrubbed vented releases from the primary containment with the core debris on the drywell floor. This is considered to bound all scrubbed releases (modes A2, A5, and B2) as well as controlled, unscrubbed releases with the core within the vessel (mode A3).

Release mode C8 bounds both C7 and C9 because all three modes are unscrubbed releases with containment failure before or simultaneous with vessel breach, but mode C8 has the largest hole size, so its release is expected to be more severe than modes C7 or C9. Release mode C12 bounds C11 for similar reasons.

Release mode D3 is best modeled by D1 because both of these sequences are delayed releases with small containment failures and deposition of fission products in the wetwell waterspace prior to containment failure. The releases are therefore expected to be quite similar. Release mode E1 is for LOCAs outside containment and results in one of the largest releases. This certainly bounds release mode E2, which is containment isolation failure.

Table 4.7-1
FRACTION OF FISSION PRODUCTS RELEASED

<u>How to use this table:</u> Each column shows the findings of a MAAP case run to estimate the source term for a selected release mode. The first line is the MAAP case number. The second line gives the sequence number used for the MAAP case. This corresponds to the sequences shown on the Containment Event Trees. Below that is the release mode modeled by the MAAP case. The fractions of the original core inventory of the various fission product groups are shown in the column below each MAAP case. The volatile groups are shaded on this table. This aids in determining which release category the case belongs in.

MAAP Case ==>	MPB051	MPB052	MPB053	MPB055	MPB056
Sequence and Release Mode =>	1A-17 A4	1D-08 A4	3-06 A6	4-10 C8	1B-27 C12
F. P. Group ⬇					
1 (Nobles)	0	0	$1.00_{(E)}$	0.97	0.83
2 (CsI, RbI)	0	0	$1.4E-4_{(E)}$	0.50	0.15
3 (TeO_2)	0	0	0	0	0
4 (SrO)	0	0	$4E-7_{(E)}$	3.7E-4	$6E-4_{(E)}$
5 (MoO_2)	0	0	$7E-9_{(E)}$	1.3E-3	5.8E-4
6 (CsOH)	0	0	$1.8E-4_{(E)}$	0.40	0.13
7 (BaO)	0	0	$2E-7_{(E)}$	1.3E-3	$6E-4_{(E)}$
8 (lanthinides)	0	0	$1.2E-8_{(E)}$	1.8E-3	$2E-5_{(E)}$
9 (CeO_2)	0	0	$6E-8_{(E)}$	2.3E-3	$2E-4_{(E)}$
10 (Sb)	0	0	$6.4E-4_{(E)}$	0.056	0.072
11 (Te_2)	0	0	$5E-4_{(E)}$	7.3E-3	$0.02_{(E)}$
12 (U/Trans-U)	0	0	$2E-10_{(E)}$	2.9E-8	$4E-7_{(E)}$
Release Category	0	0	1	3	3

(E) = estimated because value still rising at the end of the MAAP run.

Table 4.7-1 (continued)
FRACTION OF FISSION PRODUCTS RELEASED

MAAP Case ==>	MPB057	MPB058	MPB059	MPB063
Sequence and Release Mode =>	1B-29 D5	1A-16 D1	Aout E1	2-12 C4
F. P. Group ⬇				
1 (Nobles)	1.00	1.00	1.00	1.00
2 (CsI, RbI)	$0.45_{(E)}$	$1.7E-3_{(E)}$	0.82	$0.20_{(E)}$
3 (TeO_2)	0	$1.4E-4_{(E)}$	$3.2E-4_{(E)}$	0
4 (SrO)	2.7E-6	1.5E-5	4E-3	4.0E-6
5 (MoO_2)	3.8E-7	3.4E-6	1.8E-3	1.7E-4
6 (CsOH)	$0.35_{(E)}$	$1.4E-3_{(E)}$	0.90	$0.25_{(E)}$
7 (BaO)	2.2E-6	8.0E-6	3E-3	3.3E-5
8 (lanthinides)	2.6E-7	6.8E-7	2E-4	4.2E-8
9 (CeO_2)	1.3E-6	4.7E-6	8.2E-4	5.8E-8
10 (Sb)	3.0E-3	$6E-4_{(E)}$	0.05	$3E-3_{(E)}$
11 (Te_2)	$0.04_{(E)}$	4E-4	0.06	1.3E-6
12 (U/Trans-U)	$8.0E-9_{(E)}$	1.1E-8	5E-6	0
Release Category	3	1	3	3

(E) = estimated because value still rising at the end of the MAAP run.

Table 4.7-2
SUMMARY TABLE OF SOURCE TERM FINDINGS -- INTERNAL EVENTS

MAAP Case	Release Mode Modeled	Bounding for These Other Release Modes	Combined RRF[1] Per Year	Release Category[2]
MPB051/91	A4	A1	1.3E-5	0
MPB052/91	A4	A1	1.3E-5	0
MPB053/91	A6	A2, A3, A5, B2	5.6E-6	1
MPB055/91	C8	C7, C9	1.6E-6	3
MPB056/91	C12	C11	2.8E-7	3
MPB057/91	D5	--	5.1E-7	3
MPB058/91	D1	D3	5.6E-7	1
MPB059/91	E1	E2	4.1E-9	3
MPB063/91	C4	--	3.4E-8	3

1 RRF = Radionuclear Release Frequency

2 0 = No release
1 = ≤ 100% noble gases (Group 1), < 2% volatile radionuclides (Groups 2, 3, 6, 11), and < 0.1% of non-volatile radionuclides (Groups 4, 5, 7, 8, 9, 10, 12) released.
2 = ≤ 100% noble gases and 2-10% volatile radionuclides released.
3 = ≤ 100% noble gases and >10% volatile radionuclides released.

Table 4.7-3
SUMMARY TABLE OF SOURCE TERM FINDINGS -- INTERNAL FLOODS

MAAP Case	Release Mode Modeled	Bounding for These Other Release Modes	Combined RRF[1] Per Year	Release Category[2]
MPB051/91	A4	A1	3.3E-6	0
MPB052/91	A4	A1	3.3E-6	0
MPB053/91	A6	A2, A3, A5, B2	9.6E-7	1
MPB055/91	C8	C7, C9	3.6E-8	3
MPB056/91	C12	C11	1.2E-7	3
MPB057/91	D5	--	2.6E-7	3
MPB058/91	D1	D3	2.4E-6	1
MPB059/91	E1	E2	1.7E-7	3
MPB063/91	C4	--	0	3

---

1 RRF = Radionuclear Release Frequency

2 0 = No release
1 = ≤ 100% noble gases (Group 1), < 2% volatile radionuclides (Groups 2, 3, 6, 11), and < 0.1% of non-volatile radionuclides (Groups 4, 5, 7, 8, 9, 10, 12) released.
2 = ≤ 100% noble gases and 2-10% volatile radionuclides released.
3 = ≤ 100% noble gases and >10% volatile radionuclides released.

FIGURE 4.7-1

MONTICELLO

Radionuclide Release Characterization Given Core Damage Has Occured

Internal Events

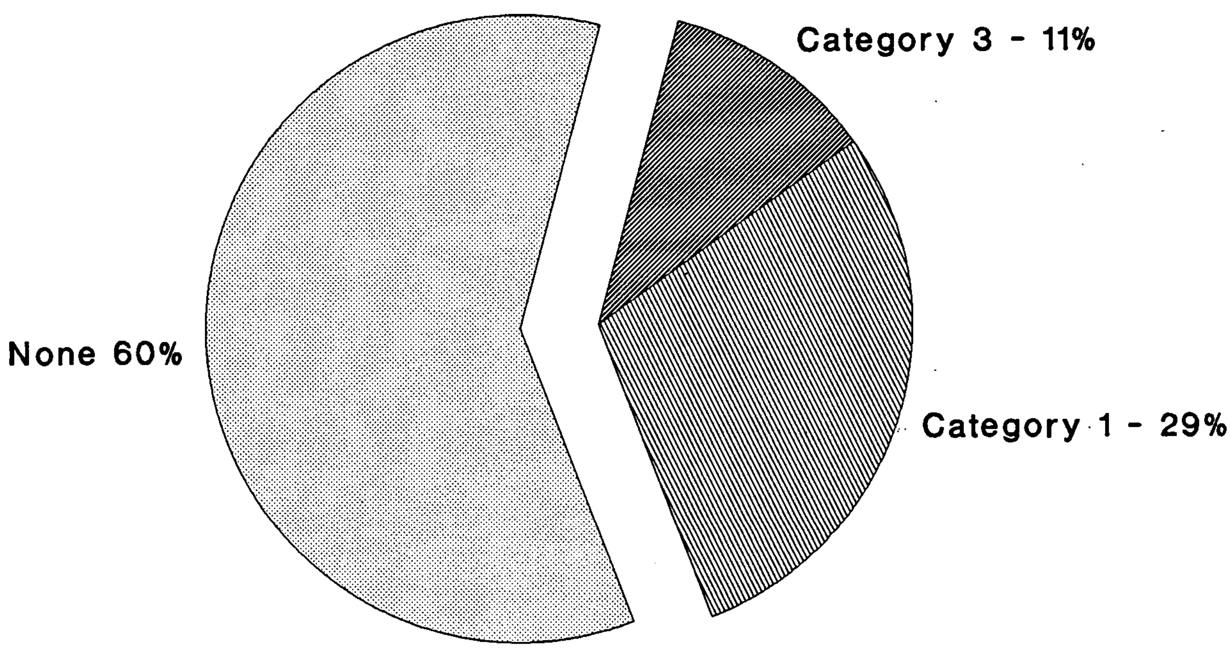
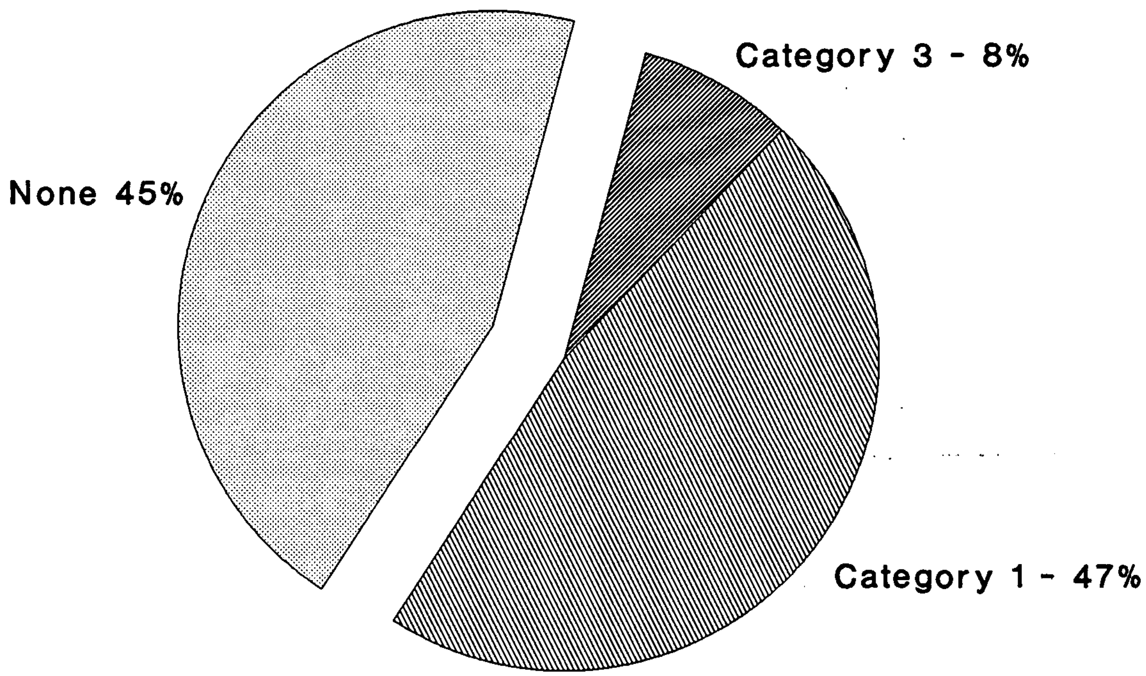


FIGURE 4.7-2

MONTICELLO

Radionuclide Release Characterization Given Core Damage Has Occured

Internal Floods



4.8 Sensitivity Studies

4.8.1 Probabilistic Sensitivity Studies

The impact of uncertain assumptions and analyses on the level 2 PRA accident sequence quantification was investigated using probabilistic sensitivity studies. These calculations were done in addition to the deterministic sensitivity studies performed using the MAAP computer program.

These studies were done by either reassigning the affected sequences to new release modes and damage states, or by altering the probabilities of basic events and partially reperforming the sequence quantification using PCSETS.

Three sensitivities were examined:

- Liner Meltthrough.
- Molten Core Concrete Interaction.
- Arrest in Vessel.

4.8.1.1 Liner Melt Sensitivity Study

The sensitivity of the results to the liner meltthrough containment failure mechanism was examined. Realistic analysis shows that liner meltthrough will not happen at Monticello because there is insufficient core debris spilled from the vessel to reach the metal containment shell. This calculation is sensitive to several factors: the amount of core debris that exits upon vessel breach; the influence of the water in the submerged piping connecting the containment sumps; and the potential impact that secondary debris spills have in adding to the volume of molten debris in the containment.

For the liner meltthrough sensitivity study, it is assumed that if the core debris is voluminous enough to contact the metal containment shell, then the depth would be less than 5 cm at the point of contact. This assumption is supported by the best estimate calculation that finds that NO core debris would contact the metal containment liner. In these cases, a meltthrough probability of 0.50 is used [4.8-1].

If liner meltthrough was a 50% probability for sequences where the core debris enters a dry containment, some of the releases currently classified as drywell vent and core concrete interaction would be reclassified as liner meltthrough. Figure 4.8-1 shows the change in the distribution of core damage that would be expected. This change is approximately 5% of the combined core damage frequency calculated for internal events and the external event of internal flooding. Since approximately 60% of core damage accidents result in no release, this redistribution represents 13% of all releases.

The release occurs earlier with liner melt than drywell vent, and it is not regulated so the total source term would be larger. Furthermore, liner melt would occur very shortly after vessel breach, minimizing the effectiveness of fission product retention mechanisms within the containment. It is not possible to determine the source term of a liner meltthrough with MAAP, because MAAP only models gaseous releases. Liner meltthrough could result in corium physically spilled into the reactor building basement.

If liner melt were modelled to occur 100% of the time core debris is spilled into a dry containment, then the effect summarized above would be the same, but twice as much of drywell vent sequences would have been shuffled to liner melt.

4.8.1.2 Molten Core-Concrete Interaction

The sensitivity of the results to molten core concrete interaction (MCCI) was examined. MCCI is expected to threaten the containment only by production of non-condensible gases. The impact of the gas production would be to increase containment pressure. This was accounted for in the original analysis by setting the variable MCCI equal to the containment failure probability at the final pressure in a "core-on-the-floor" case. If MCCI threatens the containment either through pedestal collapse or basemat penetration, then the value of MCCI could be as high as 1.00. A study was done by modifying the result to represent this worst case.

If molten core-concrete interaction was to occur every time core debris was spilled onto the containment floor, the probability of terminating a nuclear accident without a release would decrease slightly from approximately 56% to approximately 54%. This is because most of the non-release sequences involve arrest in the reactor vessel. Very little redistribution among releases occurred because many core-on-the-floor sequences involve earlier releases which would not be affected by MCCI. Figure 4.8-2 shows the redistribution if MCCI always happened.

4.8.1.3 Arrest in Vessel

The sensitivity of the results to the ability to terminate the nuclear accident in the reactor vessel was examined. Although strong evidence from the Three Mile Island accident supports the assumption that core debris can be quenched within the reactor vessel, there is a degree of uncertainty about the coolability of the debris because of uncertain debris composition and geometry. This is examined by reassigning sequences where debris cooling was successful to ex-vessel sequences where similar equipment failures occurred.

This study effectively mapped release modes A1, A2, and A3 onto A4, A5, and A6 respectively. In other words, the release did not change substantially, just the location of the core debris. No impact worth noting was observed from this study. Figure 4.8-3 shows the impact if the accident cannot be terminated with the reactor vessel intact.

If the arrest in vessel study was combined with one of the other two, then a substantial change in the calculated release mode frequencies would be expected because of tripling the core-on-the-floor sequence probabilities. NSP did not examine the impact of multiple modeling changes at the same time.

4.8.2 Phenomenological Sensitivity Studies using MAAP 3.0B

The sensitivity studies described in this section are used to examine the uncertainty introduced into the Monticello IPE because of uncertainty among experts concerning the ways in which various severe accident phenomena would influence the progression of an accident. The goal of these studies is to understand the range of plant behaviors considered possible according to different models of these phenomena.

The MAAP code allows the use of certain input variables, called model parameters, to control the manner in which various severe accident phenomena are modeled, either by altering the input used in a given model or by selecting alternative models of a particular phenomenon. This feature was used to see how the prediction of plant behavior during a severe accident changed when different assumptions were made.

These studies were prepared using Gabor, Kenton & Associates' report for EPRI, "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B."

4.8.2.1 Hydrogen Production and Core Melt Progression

During a core melt, the core geometry is expected to change as molten materials flow away from the melting region. If molten material refreezes below the melting region in such a way as to close off the fuel channel, steam flow would be blocked through the entire channel, halting oxidation of the Zircalloy in that channel and thereby halting hydrogen production. If instead the refrozen material does not block the channel, steam flow and Zircalloy oxidation would continue. The flow and temperature of gases through the core would also be affected, altering the transport of fission products. Because disagreement exists about which model is correct, MAAP 3.0B BWR allows the user to select among three different models:

1. The "no blockage" model assumes that the relocation of core materials has little effect on gas flows, hydrogen production, or fission product release rates. This model predicts more hydrogen production than the other models.

2. The "local blockage" model assumes that as the Zircalloy cladding melts, it flows away from the melting region; once all the Zircalloy has melted out of a region, no further hydrogen is produced there. Gas flows and fission product release rates are not changed by the new geometry in this model.

3. The "channel blockage" model assumes that as molten material refreezes, it blocks off the fuel channel. Steam and gas flow through the entire channel is cut off, stopping oxidation of the cladding. Below the blockage the channel will pressurize, driving water out of the channel. This model predicts less hydrogen production and lower core exit temperatures than the other two models.

The choice of blockage model affects not only the amount of hydrogen produced, but may affect the source term as well, particularly when the source term is dominated by late revaporization. The no blockage and local blockage models produce higher core exit temperatures than the channel blockage model. These higher temperatures cause more fission products to be swept into the suppression pool and captured early in the accident; when surfaces in the primary system reheat later in the sequence, fewer fission products are there to be revaporized and released. Therefore, in examining sensitivity to the choice of blockage model in MAAP, both hydrogen production and source term were reviewed.

Other MAAP parameters which influence hydrogen production and core melt progression, such as clad surface area, eutectic melting temperatures, and latent heats of fusion have less effect than the choice of blockage model. These parameters therefore were not used for sensitivity studies, since their influence is expected to fall within the range of results predicted by the different blockage models.

Station Blackout - Sensitivity to Blockage Model: A station blackout case in which HPCI and RCIC are available until the batteries fail was run first using the local blockage model, and then with the channel blockage model. As expected, the channel blockage model predicts lower core exit temperatures and much less hydrogen production. The decreased hydrogen production results in lower containment pressures. In neither case does the pressure become high enough to challenge the containment, and it fails instead on high gas temperature in the drywell. When this happens, the amount of airborne fission products in the drywell is about the same in either case; however, because of the higher drywell pressure in the local blockage case, more of these fission products are expelled into the reactor building, and a larger source term is seen. No significant revaporization is seen until after the containment has failed. When revaporization does occur, the channel blockage model does indeed predict that a greater mass of fission products becomes airborne in the containment; however, by that time the containment has depressurized, so the revaporized fission products are not driven into the reactor building. [Figures 4.8-4 through 4.8-6.]

CORE BLOCKAGE MODEL SENSITIVITY - STATION BLACKOUT

	Local Blockage	Channel Blockage
Reactor vessel failure time	9.1 hr	10.1 hr
Drywell overtemperature failure time	27 hr	32 hr
Drywell pressure at time of failure	86 psia	72 psia
H_2 produced by time of drywell failure	1300 lbm	900 lbm
Fission products released to the reactor building:		
noble gases:	85%	86%
CsI/RbI:	26%	9%
CsOH:	22%	8%
Te_2:	8%	6%
Sb:	5%	1%
other fission products:	-	-
Peak airborne fission product mass during revaporization	40 lbm	50 lbm

It should be noted that the source terms used in this IPE to characterize the different accident classes were derived using the no blockage model, since local blockage was not available in revision 7.0 of the MAAP 3.0B BWR code. All three models predict a category 3 release for this accident sequence. These sensitivity studies were run using revision 7.02.

Deinerted Station Blackout - Sensitivity to Blockage Model: The Monticello containment is normally inerted with nitrogen, but hydrogen burning in the containment would be possible if an accident occurred during a period when the containment was deinerted. A sensitivity study was done using a deinerted station blackout event to see the effect of the different hydrogen generation rates predicted by the different blockage models.

The local blockage model predicts that hydrogen burning will begin in the deinerted wetwell at 6.9 hours; channel blockage predicts a slightly later burn, at 7.2 hours, but in either case the containment remains intact during the burn. The channel blockage model again predicts that more fission products are retained in the primary system, but very little revaporization occurs and this does not influence the source term. Instead, as in the inerted cases, it is seen that the additional hydrogen predicted by the local blockage model causes a higher drywell pressure at the time of drywell failure, which leads to a larger release of the airborne fission products from the containment. [Figures 4.8-7 and 4.8-8.]

CORE BLOCKAGE MODEL SENSITIVITY - DEINERTED STATION BLACKOUT

	Local Blockage	Channel Blockage
Hydrogen burning in the wetwell	6.9 hr	7.2 hr
Reactor vessel failure time	10 hr	9 hr
Drywell overtemperature failure time	30 hr	38 hr
Drywell pressure at time of failure	87 psia	79 psia
H_2 produced by time of drywell failure	1270 lbm	970 lbm
Fission products released to the reactor building:		
noble gases:	87%	87%
CsI/RbI:	13%	6%
CsOH:	11%	5%
Te_2	3%	1%
Sb:	5%	2%
other fission products:	-	-

4.8.2.2 Recovery of a Badly Damaged Core

MAAP 3.0B is not considered to be an appropriate tool for predicting whether a badly damaged core could be recovered before vessel failure, and it was not used to do so. Instead, the assumptions made in the level 2 analysis about the time available for recovery in the vessel were checked against the times predicted by MAAP for high fuel temperatures (> 2020°F) and vessel failure to ensure that those assumptions were reasonable.

4.8.2.3 Fission Product Revaporization

It is possible that chemical reactions would take place between steel surfaces in the primary system and the fission products which settle there, particularly cesium iodide and cesium hydroxide, allowing these fission products to concentrate in one area rather than being evenly dispersed through the primary system. This would affect the timing and extent of revaporization, and may increase the source term if it causes vaporization to be in progress at the time of containment failure. Such reactions are not modeled in MAAP; the best available model parameter for examining their impact is a multiplier on the vapor pressure of cesium iodide. As noted by Gabor and Kenton, this parameter affects not only vaporization, but also the mass of vapor relative to the aerosol mass at a given temperature. However, since there is no mechanism in MAAP at this time to alter the revaporization calculation alone, this approach was used.

Station Blackout - Sensitivity to CsI Vapor Pressure:

A station blackout case was run in which the vapor pressure of cesium iodide was reduced by a factor of ten. This significantly increased the amount of cesium iodide initially retained in the primary system, and therefore more became airborne during revaporization. However, as in the base case, the containment failed while the cesium was still bound in the primary system, rather than during revaporization; the net effect is that less cesium was released to the reactor building. [Figures 4.8-9 through 4.8-11.]

REVAPORIZATION SENSITIVITY - STATION BLACKOUT

	Base Case	CsI Vapor Pressure
Multiplier for CsI vapor pressure	1.0	0.1
Reactor vessel failure time	10.1 hr	10.3 hr
Containment failure time	26.7 hr	27.8 hr
CsI at time of containment failure:		
in drywell	28%	23%
in wetwell	24%	16%
in rx vessel & primary system	48%	61%
CsI released to reactor building:	26%	8%
Peak airborne fission product mass during revaporization	40 lbm	65 lbm

An examination of the distribution of cesium iodide indicates that this is not due solely to the fact that less cesium iodide is in the drywell when it fails. In the base case, 28% of the cesium iodide is in the drywell at the time it fails, as opposed to 23% in the second case. However, the amount released to the reactor building in the base case is 26%, but with the reduced vapor pressure in the second case it is only 8%, meaning that a smaller fraction of the cesium iodide in the drywell is released. This suggests that if the same reactions which bind cesium compounds to steel in the primary system also bind them to steel surfaces in the drywell, this effect could lead either to increased releases when containment failure occurs during revaporization, or to decreased releases when it does not.

It is noted that the impact of revaporization phenomena such as this chemical binding is defeated in these cases by the timing of containment failure. It is also noted that retention of fission products in the primary system delays containment failure somewhat, since containment heatup is slower.

4.8.2.4 Core Material Remaining Within Original Core Boundary (FMAXCP)

The model parameter FMAXCP is used to assign a point at which all remaining core materials slump out of the core region; raising this value causes this collapse to occur earlier rather than allowing a gradual melt. In the nominal cases it is assumed that when 90% of the original core has melted out of the core region, the remaining materials collapse. This point is never reached in many cases, so that some fuel remains in the core region throughout the sequence. In order to examine the uncertainty associated with calculations of the amount and behavior of such material, a station blackout case was run in which slumping of this material from the core region is coincident with vessel failure.

Because the core slumps into the lower plenum in this case moments before the vessel fails, a large amount of debris is ejected at high pressure into the pedestal and drywell. Ablation of the concrete leads to an increase in airborne fission products in the drywell, with further increases seen as the corium remaining in the lower plenum melts and drops out of the vessel onto the existing debris in the pedestal. The mass of airborne fission products reaches a peak when the last of the debris melts out of the vessel, and falls thereafter. Revaporization is seen in the reactor vessel between 24 hours and 32 hours. The drywell air temperature rises more slowly when the fuel assemblies are not retained in the core, delaying containment failure until fewer fission products are airborne and thereby reducing releases to the reactor building. [Figures 4.8-12 and 4.8-13.]

FMAXCP SENSITIVITY - STATION BLACKOUT

	Base Case	Core dump criterion
"Core dump" criterion (FMAXCP)	0.1	0.8
Reactor vessel failure time	10.1 hr	10.1 hr
Containment failure time	26.7 hr	36.3 hr
Drywell pressure at time of failure	86 psia	87 psia
CsI at time of containment failure:		
in drywell	28%	14%
in wetwell	24%	49%
in rx vessel & primary system	48%	37%
CsI released to reactor building:	26%	2%
Peak airborne fission product mass	40 lbm	235 lbm
Airborne fission product mass at time of containment failure	35 lbm	10 lbm
Hydrogen produced	1300 lbm	1240 lbm

4.8.2.4 Coolability of Debris in Containment

There remains much uncertainty about the conditions under which a bed of core debris covered by a pool of water will be coolable. To see the range of predicted behaviors associated with varying degrees of coolability, several cases were run in which the heat transfer rate between the core debris and the overlying water is varied by changing the model parameter FCHF. The base value corresponds to a heat transfer rate of about 1200 kW/m^2 at atmospheric pressure; an uncoolable state is represented in the sensitivity case by lowering this to a value corresponding to cooling by conduction alone.

A TQUX sequence was used in which all injection to the reactor vessel is lost and the vessel is not depressurized. The core melts and is ejected at high pressure, after which one loop of LPCI provides flow through the failed vessel onto the debris, and torus cooling is initiated. As expected, the uncoolable debris caused more extensive damage to concrete, generating a great deal of concrete aerosol, increasing hydrogen generation, and causing a very large mass of fission products to become airborne in the containment during the first few hours after vessel failure. The containment pressure was only slightly increased, but the heatup rate of the drywell gas was doubled. Neither case led to containment failure during the first 40 hours of the accident, but a linear extrapolation of the drywell gas temperature shows that in the base case, the failure temperature of 700°F would be reached in about four days, while in the uncoolable case this would take about two days. However, no fission products are airborne after about 24 hours in either case, and the cesium iodide distribution indicates that most of the volatiles are swept into the suppression pool and captured when the vessel fails. The change in failure time therefore may not significantly change the source term, unless one of these failure times is concurrent with revaporization of the fission products retained in the primary system. [Figures 4.8-14 through 4.8-17.]

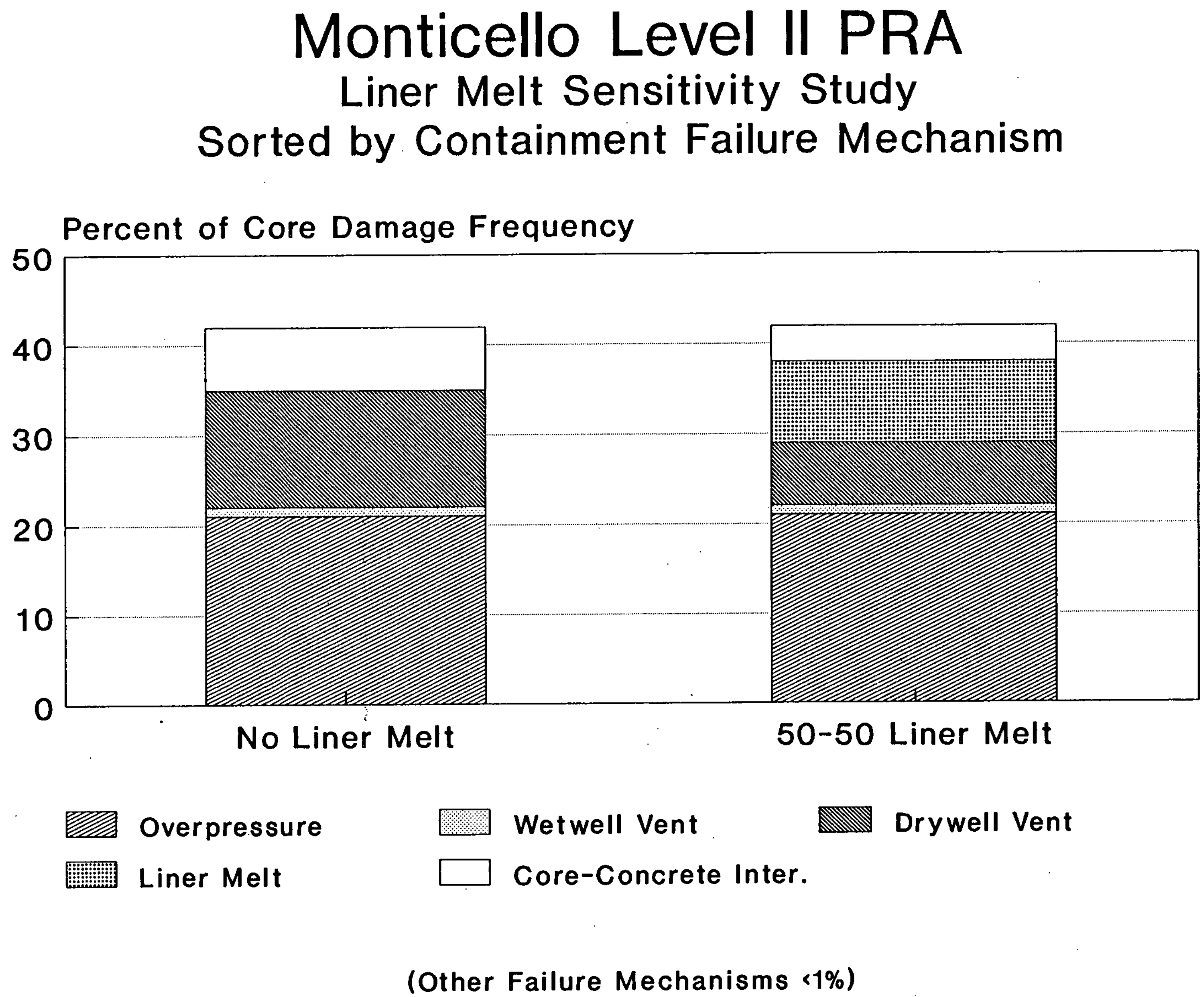
DEBRIS COOLABILITY SENSITIVITY - TQUX

	Base Case	Uncoolable Debris
Model parameter FCHF	0.14	0.02
Vessel failure time	3.8 hr	3.8 hr
Containment failure time (estimated)	4 days	2 days
Concrete aerosol generated	42 lbm	4896 lbm
Hydrogen generated	720 lbm	920 lbm
Corium elevation @ 40 hr:		
pedestal sump	5.1 ft	4.2 ft
drywell	0.1 ft	0.9 ft
Concrete ablation in pedestal sump @ 40 hr	0.1 ft	2.6 ft
Peak airborne fission product mass in containment (@ 4.5 hr)	90 lbm	430 lbm

4.8.3 References

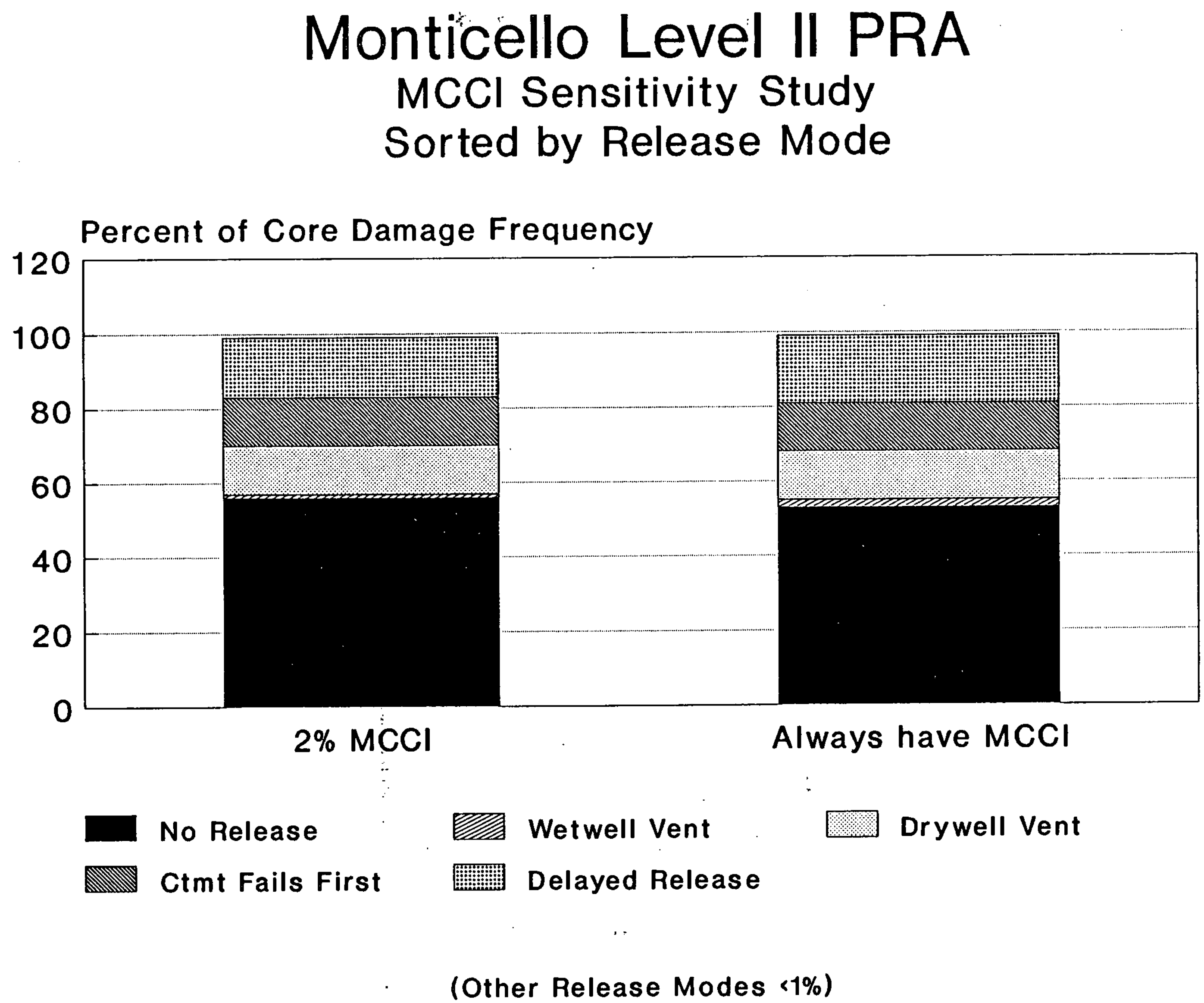
4.8-1 T. G. Theofanous, W. H. Amarasooriya, H. Yan, and U. Ratnam, The Probability of Liner Failure in a Mark I Containment, NUREG/CR-5423, University of California at Santa Barbara, July 1989.

Figure 4.8-1



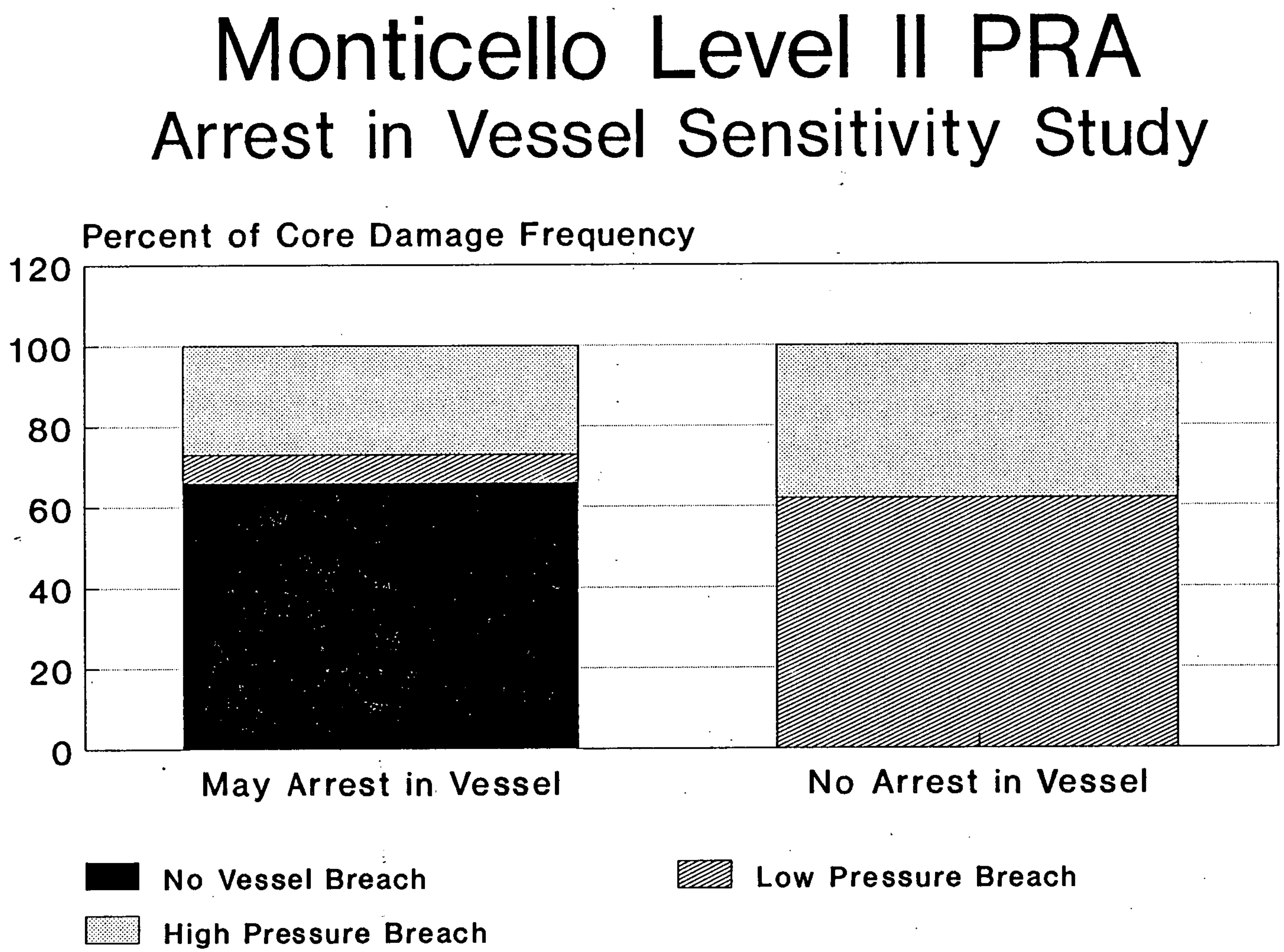
Monticello Level II PRA
Liner Melt Sensitivity Study
Sorted by Containment Failure Mechanism
Percent of Core Damage Frequency
50
40
30
20
10
0
No Liner Melt
50-50 Liner Melt
Overpressure
Wetwell Vent
Drywell Vent
Liner Melt
Core-Concrete Inter.
(Other Failure Mechanisms <1%)

Figure 4.8-2



Monticello Level II PRA
MCCI Sensitivity Study
Sorted by Release Mode
Percent of Core Damage Frequency
120
100
80
60
40
20
0
2% MCCI
Always have MCCI
No Release
Wetwell Vent
Drywell Vent
Ctmt Fails First
Delayed Release
(Other Release Modes <1%)

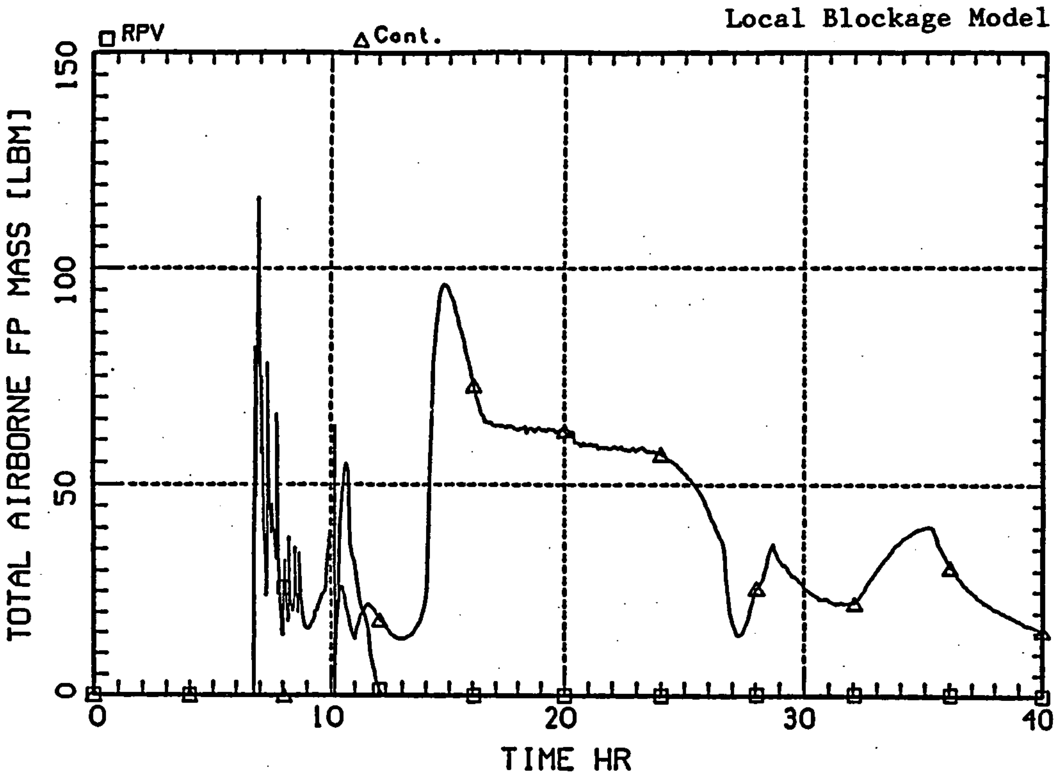
Figure 4.8-3



Monticello Level II PRA
Arrest in Vessel Sensitivity Study
Percent of Core Damage Frequency
120
100
80
60
40
20
0
May Arrest in Vessel
No Arrest in Vessel
No Vessel Breach
Low Pressure Breach
High Pressure Breach

Figure 4.8-4

CORE BLOCKAGE MODEL SENSITIVITY - STATION BLACKOUT
Total Airborne Fission Product Mass



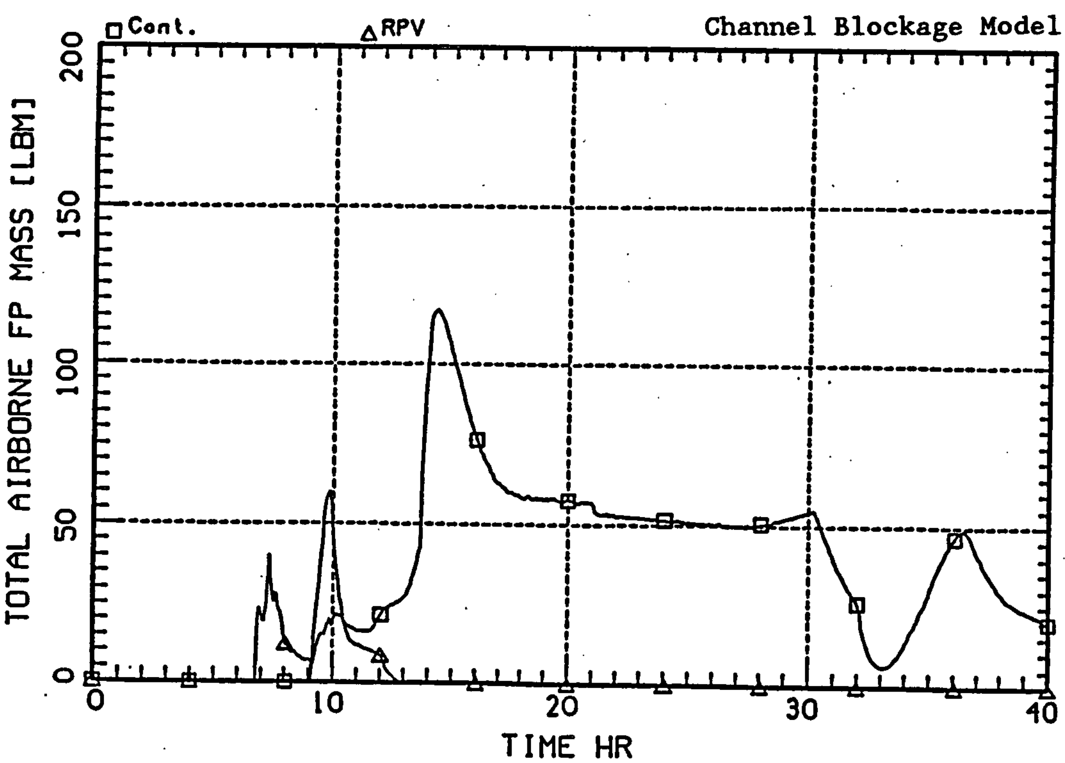
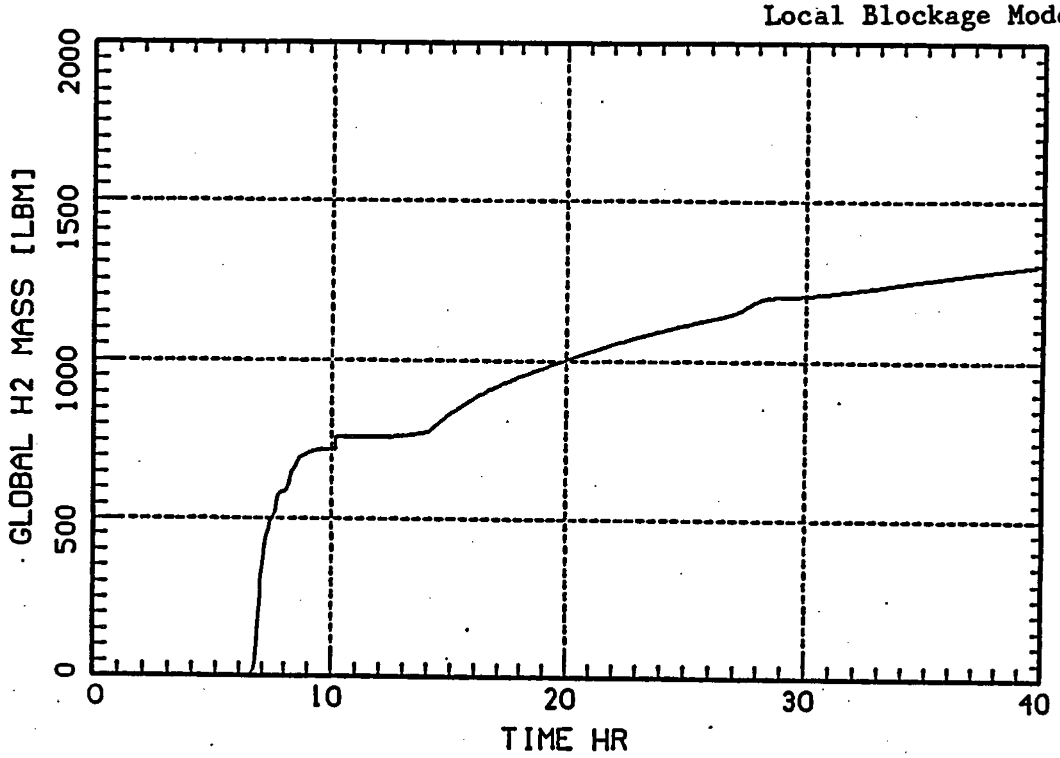


Figure 4.8-5

CORE BLOCKAGE MODEL SENSITIVITY – STATION BLACKOUT

Hydrogen Production

Local Blockage Model



Channel Blockage Model

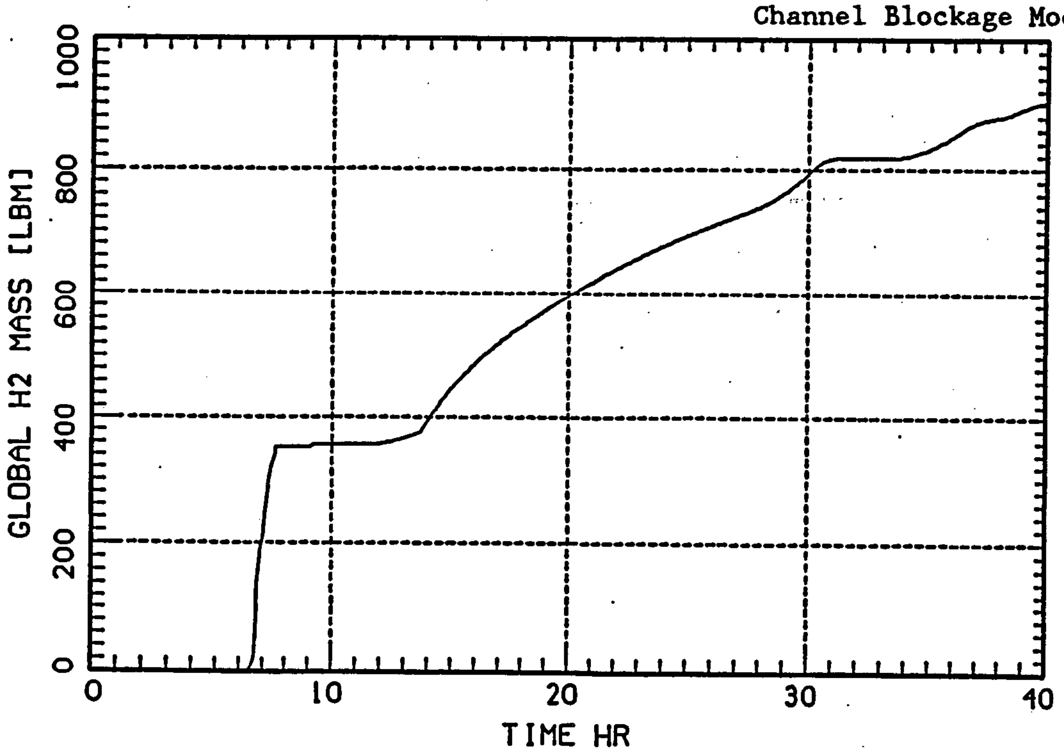
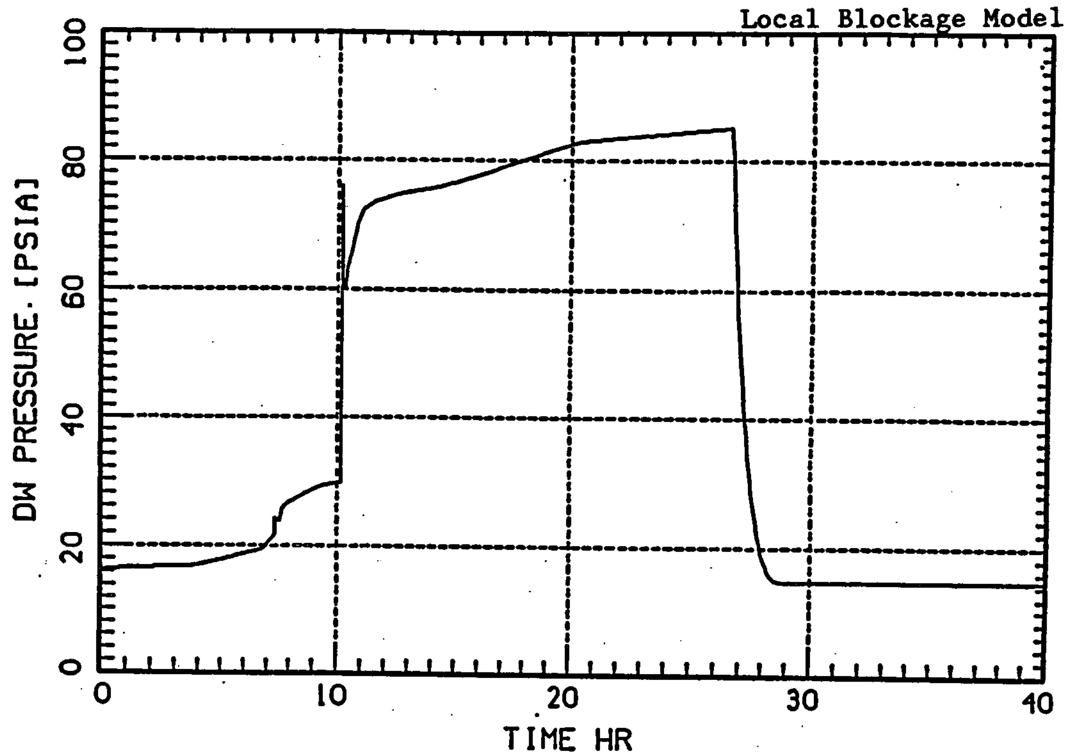


Figure 4.8-6

CORE BLOCKAGE MODEL SENSITIVITY – STATION BLACKOUT
Drywell Pressure



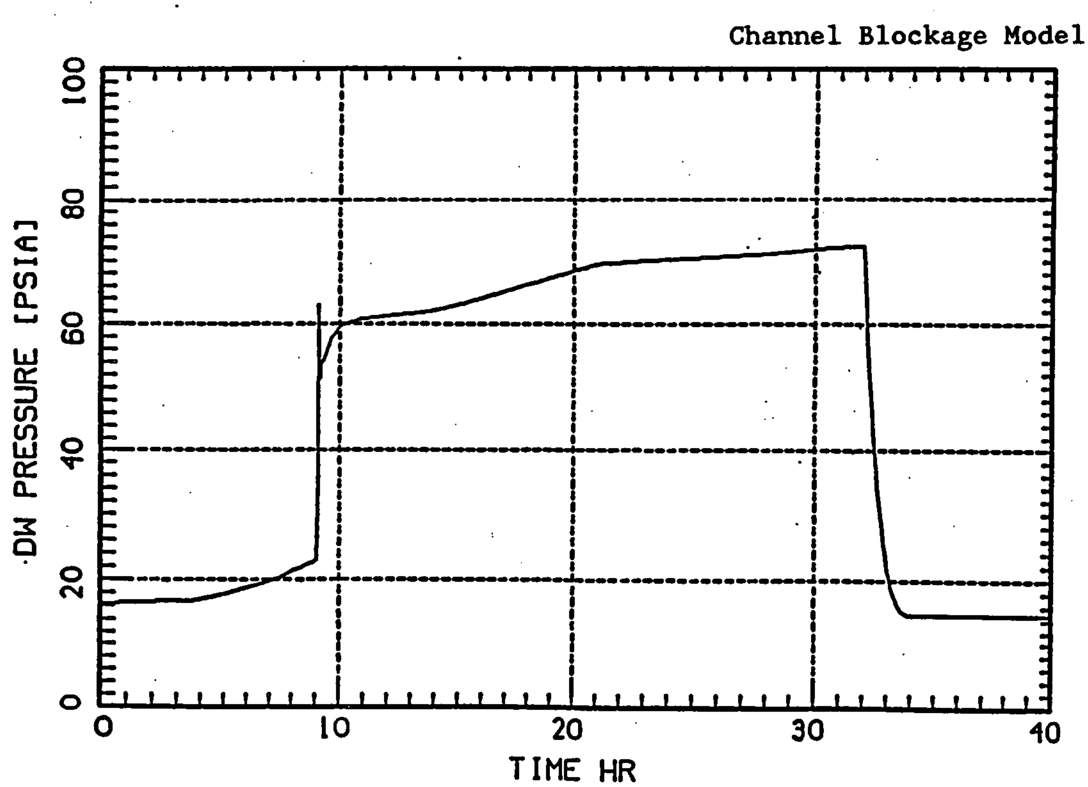
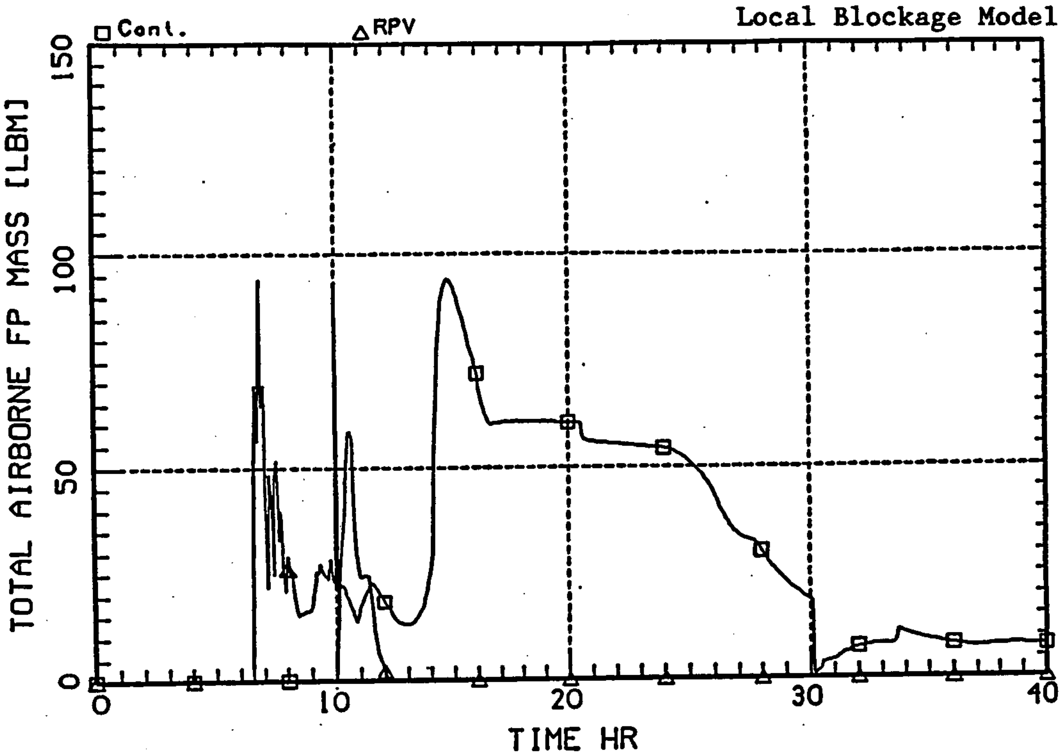


Figure 4.8-7

CORE BLOCKAGE MODEL SENSITIVITY - DEINERTED STATION BLACKOUT
Total Airborne Fission Product Mass



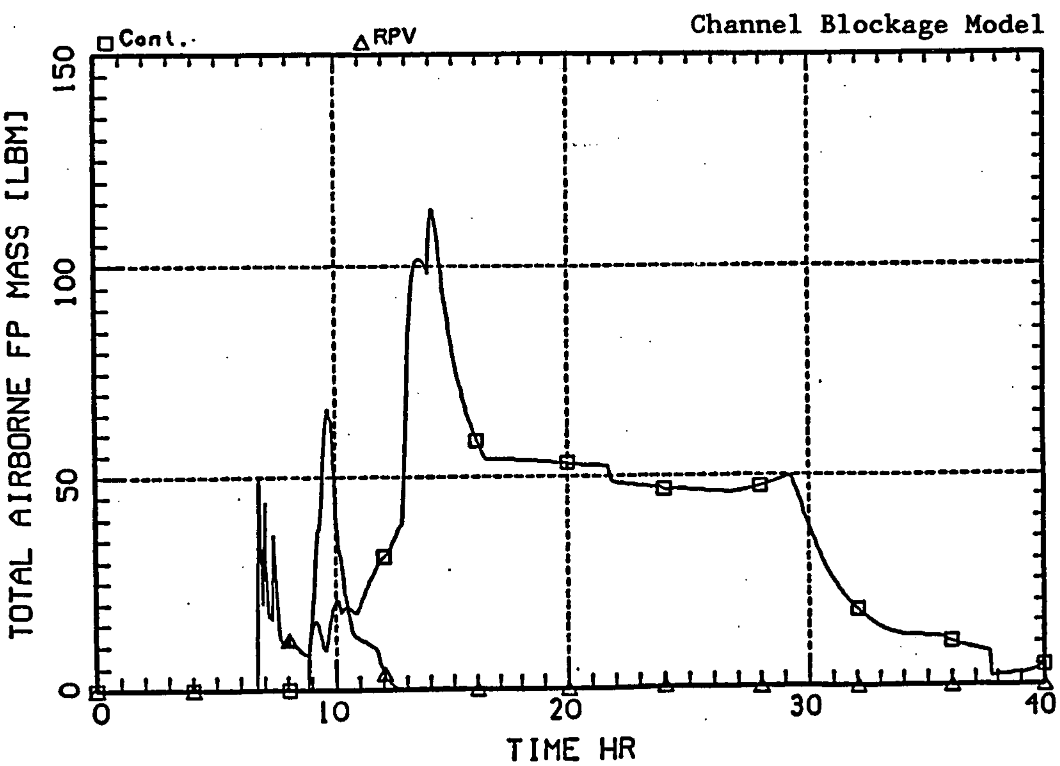
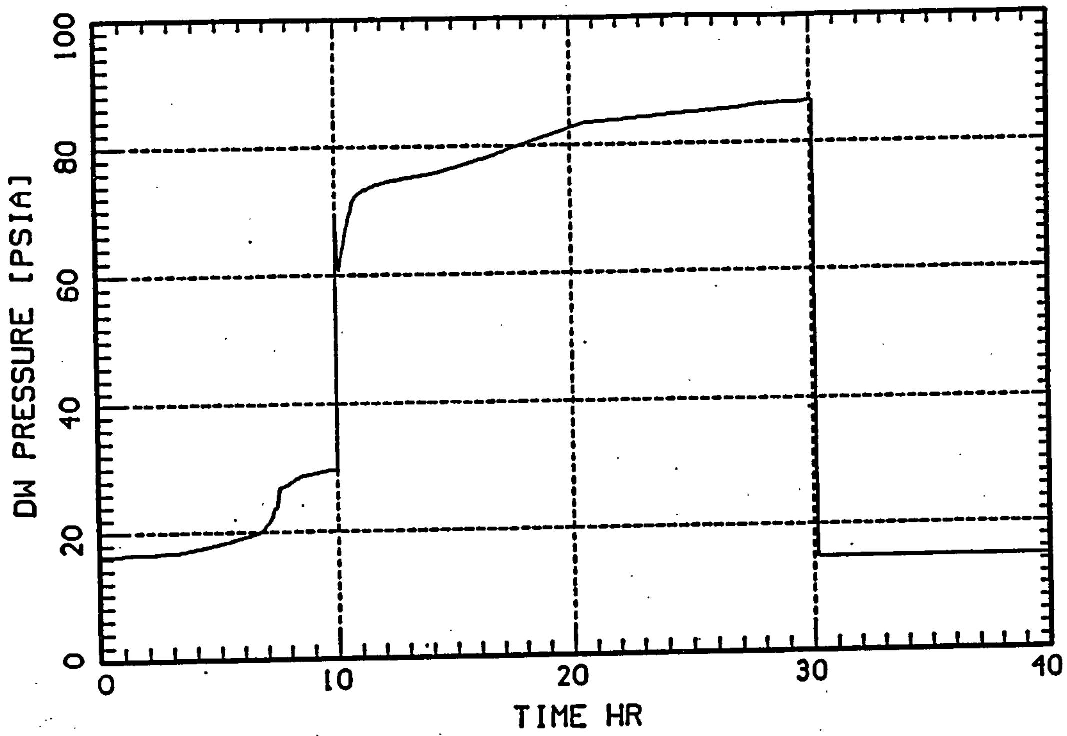


Figure 4.8-8

CORE BLOCKAGE MODEL SENSITIVITY - DEINERTED STATION BLACKOUT
Drywell Pressure

Local Blockage Model



Channel Blockage Model

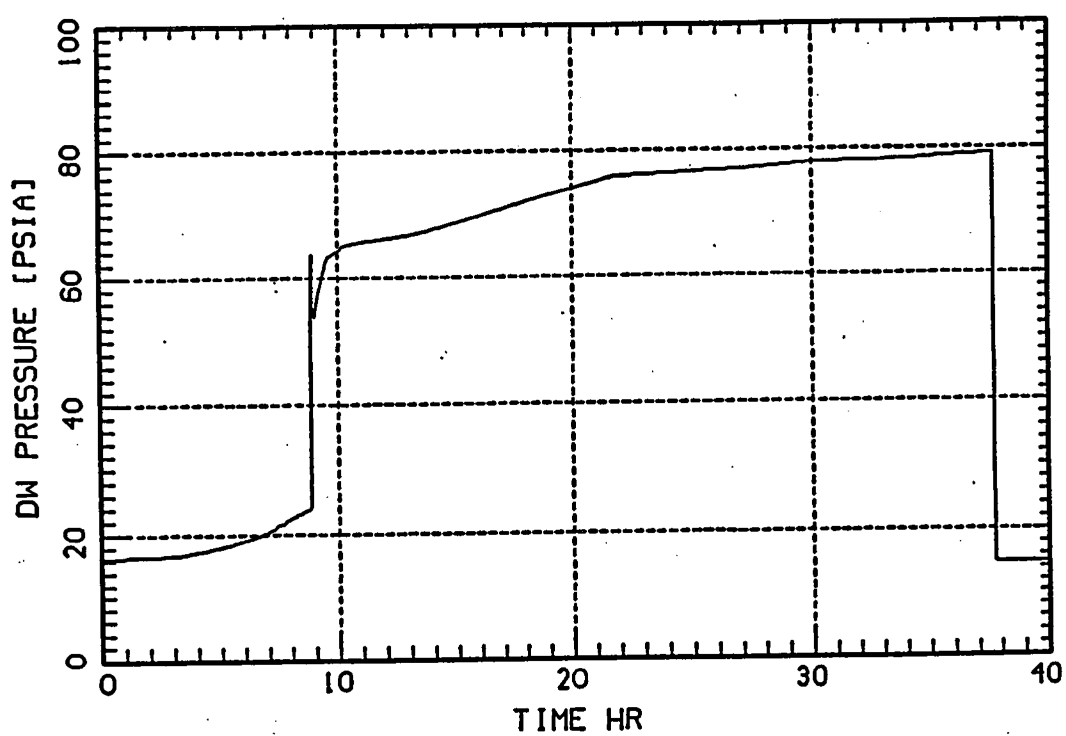
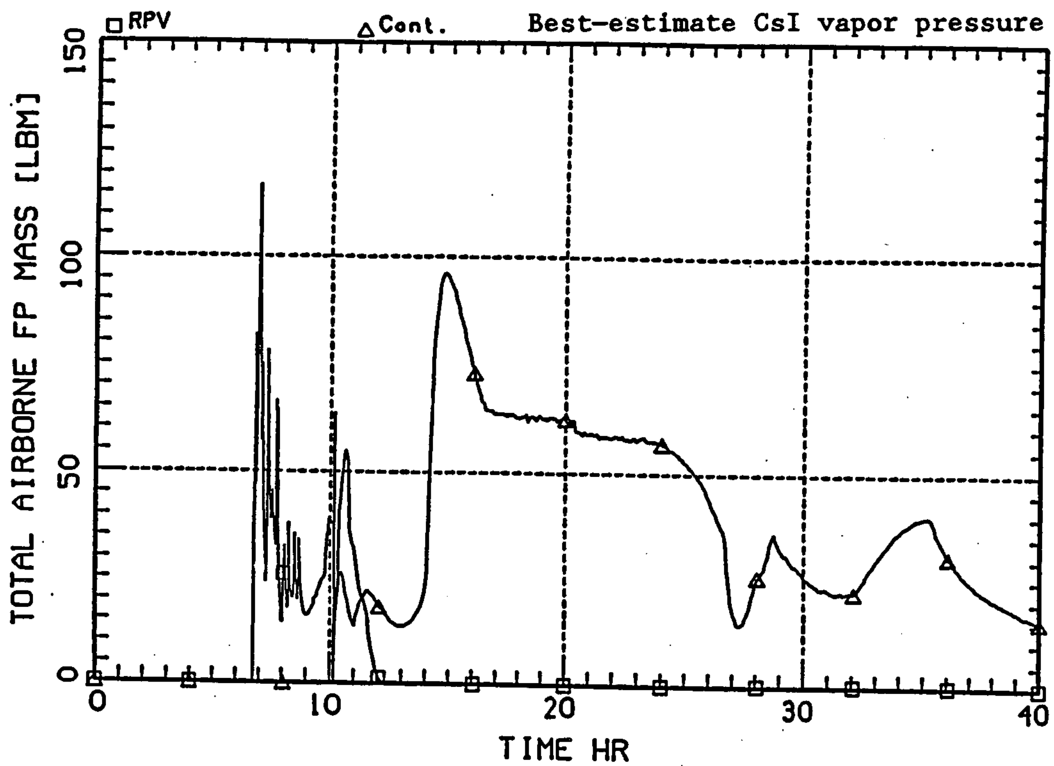


Figure 4.8-9

CsI VAPOR PRESSURE SENSITIVITY – STATION BLACKOUT
Total Airborne Fission Product Mass



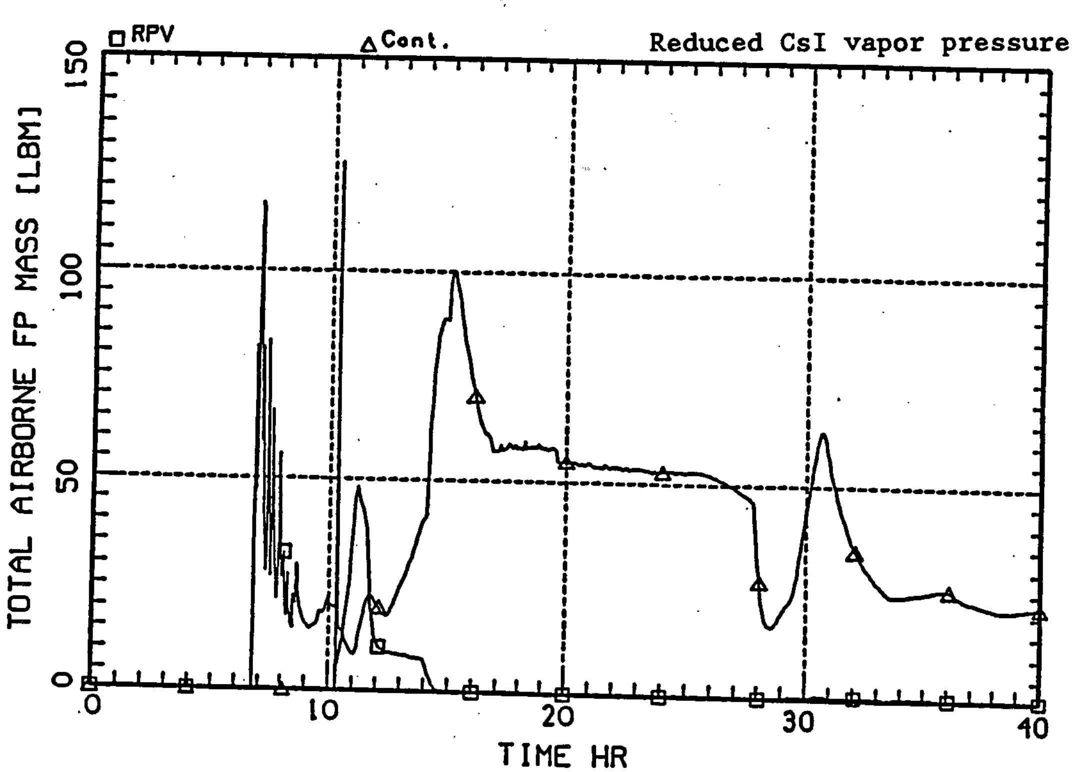
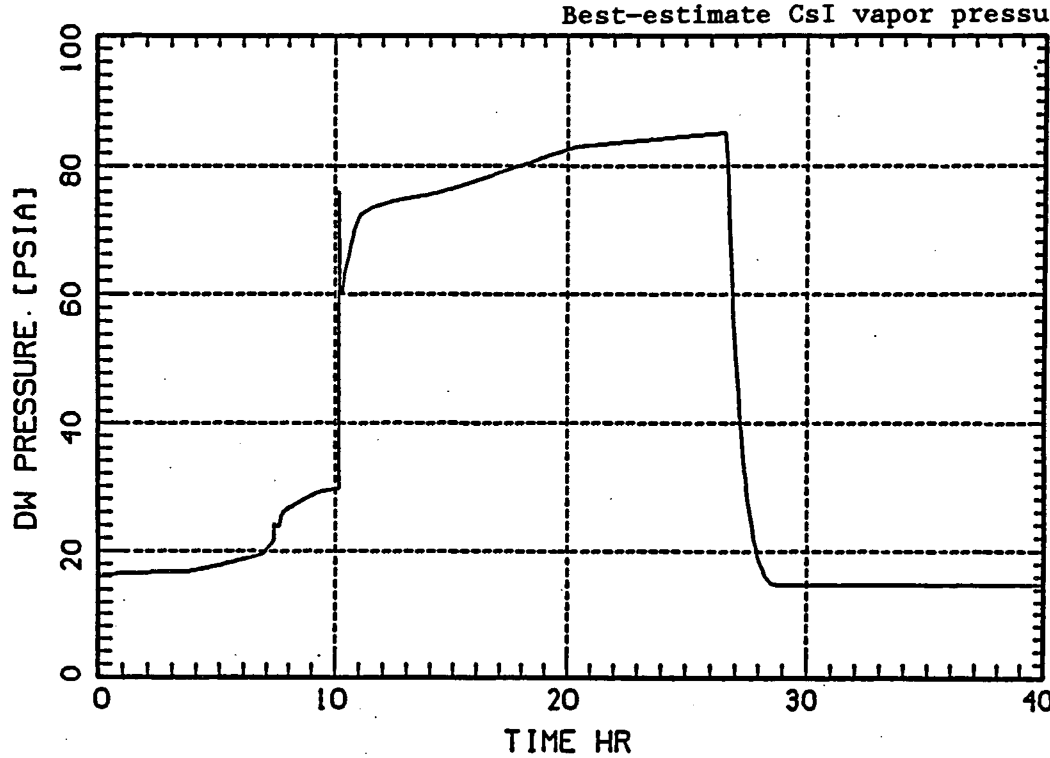


Figure 4.8-10

CsI VAPOR PRESSURE SENSITIVITY - STATION BLACKOUT
Drywell Pressure

Best-estimate CsI vapor pressure



Reduced CsI vapor pressure

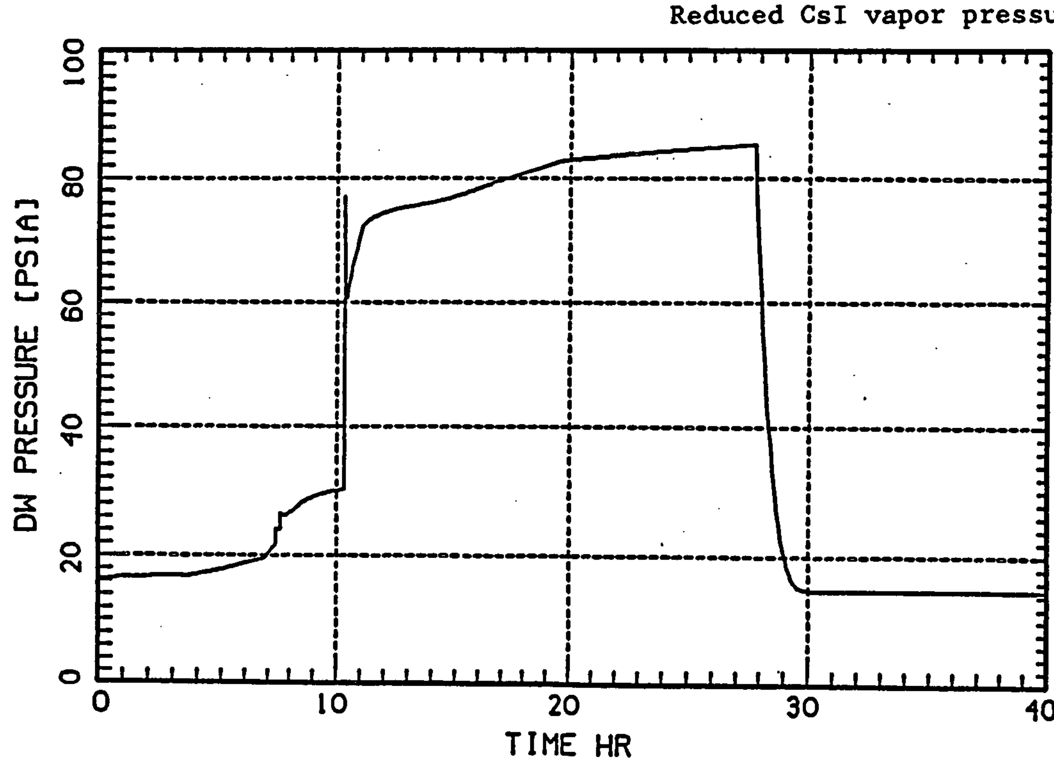
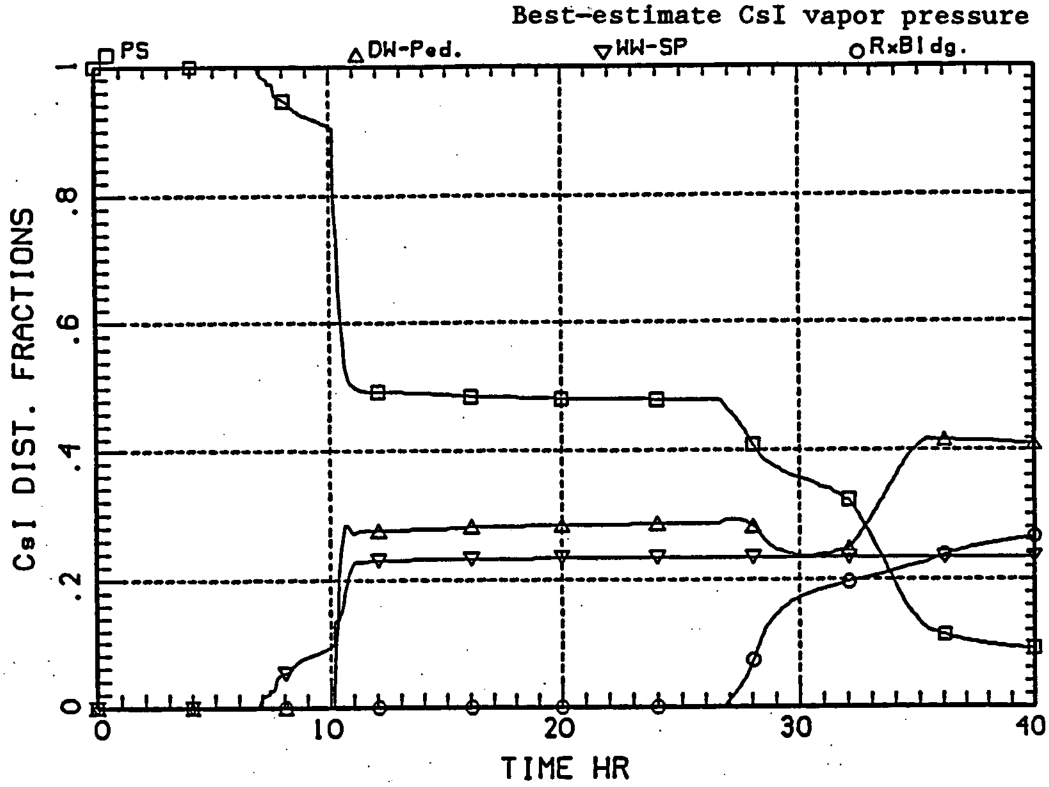


Figure 4.8-11

CsI VAPOR PRESSURE SENSITIVITY - STATION BLACKOUT
CsI Distribution Fractions



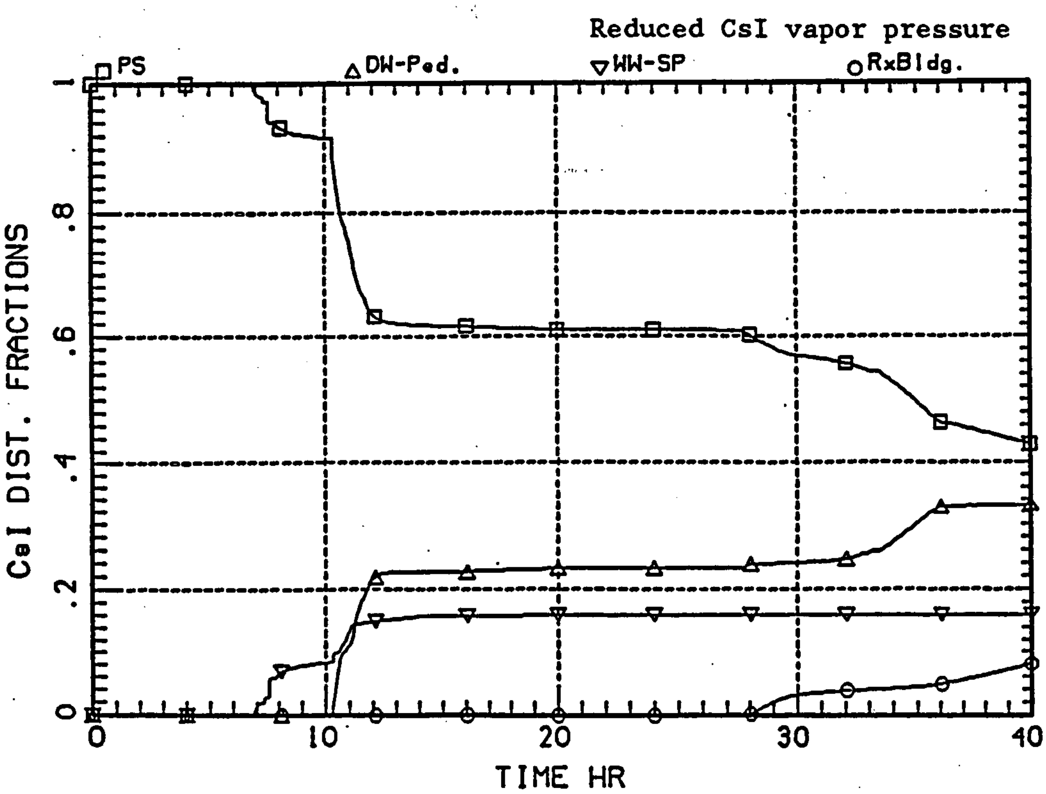
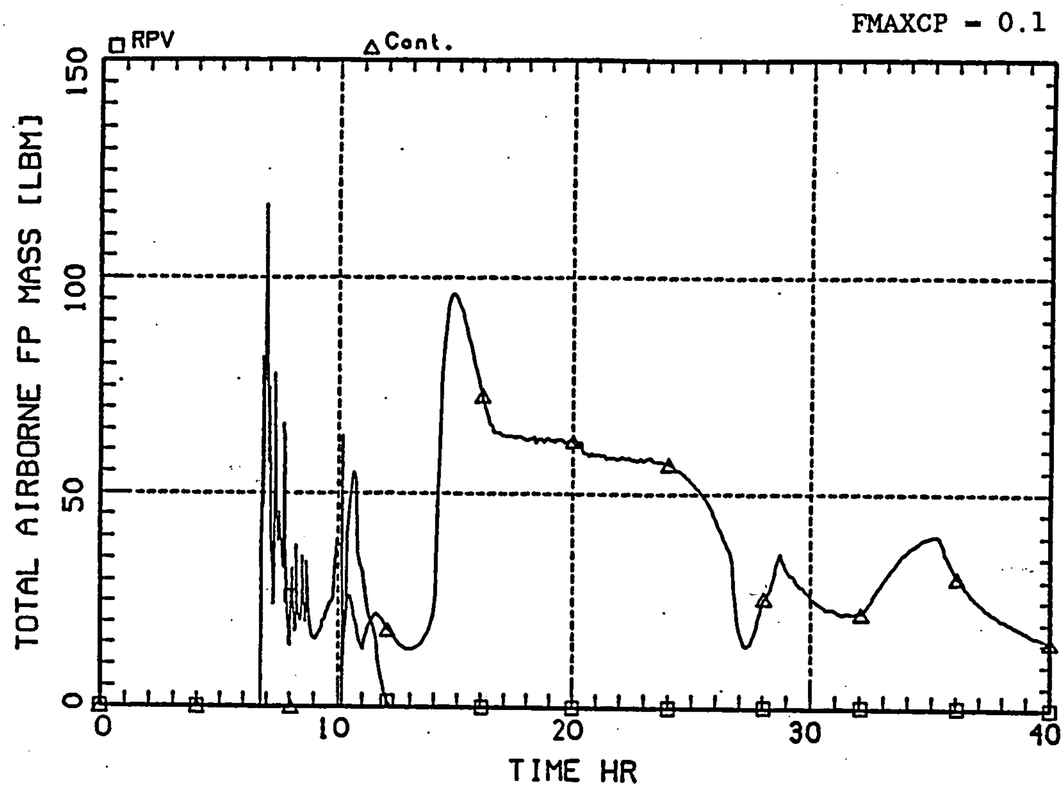


Figure 4.8-12
FMAXCP SENSITIVITY - STATION BLACKOUT
Total Airborne Fission Product Mass



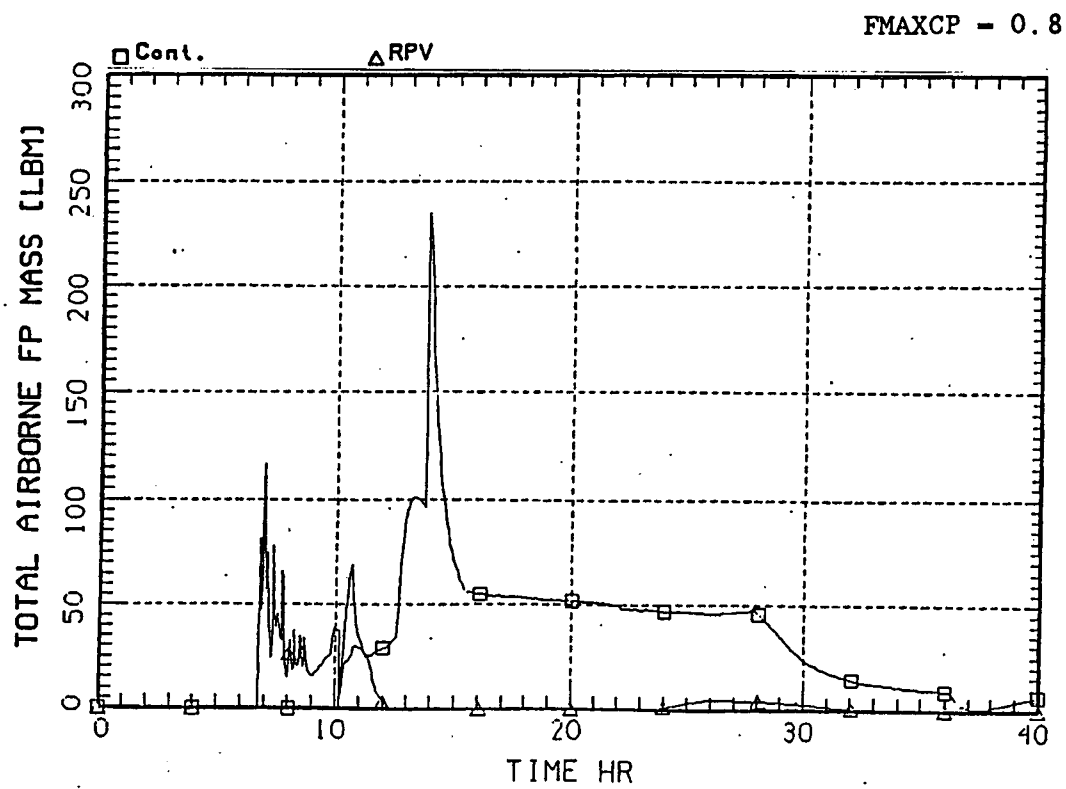
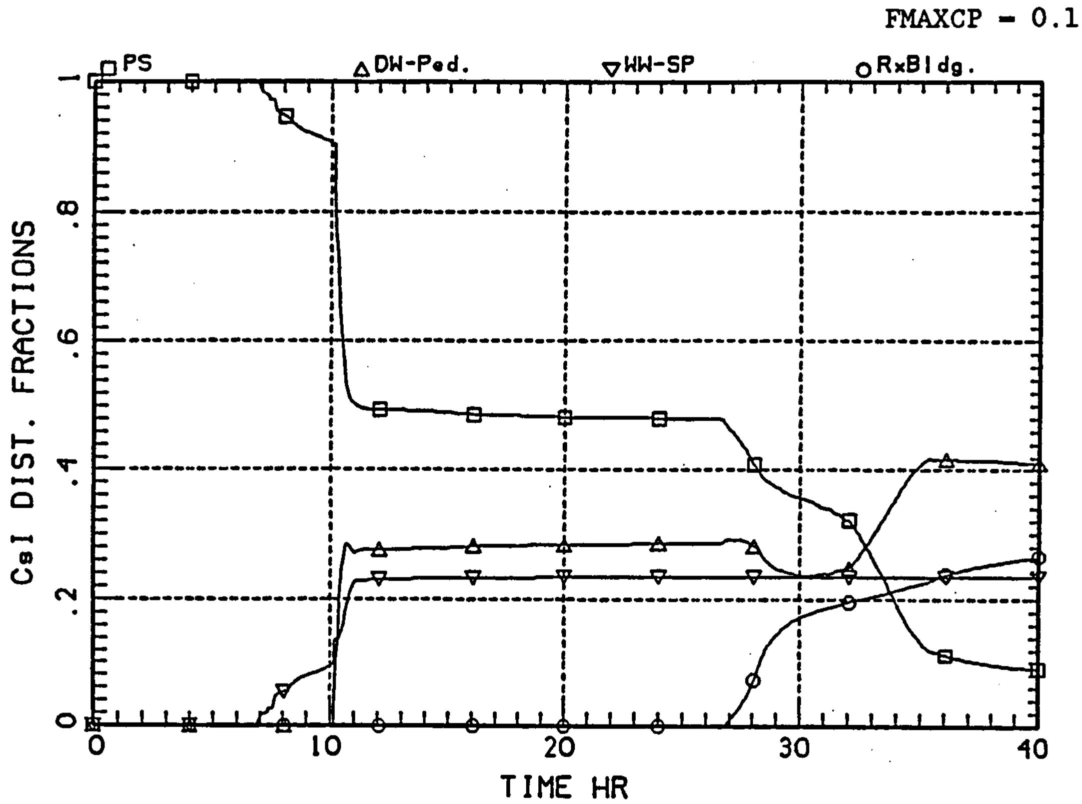


Figure 4.8-13
FMAXCP SENSITIVITY - STATION BLACKOUT
CsI Distribution Fractions



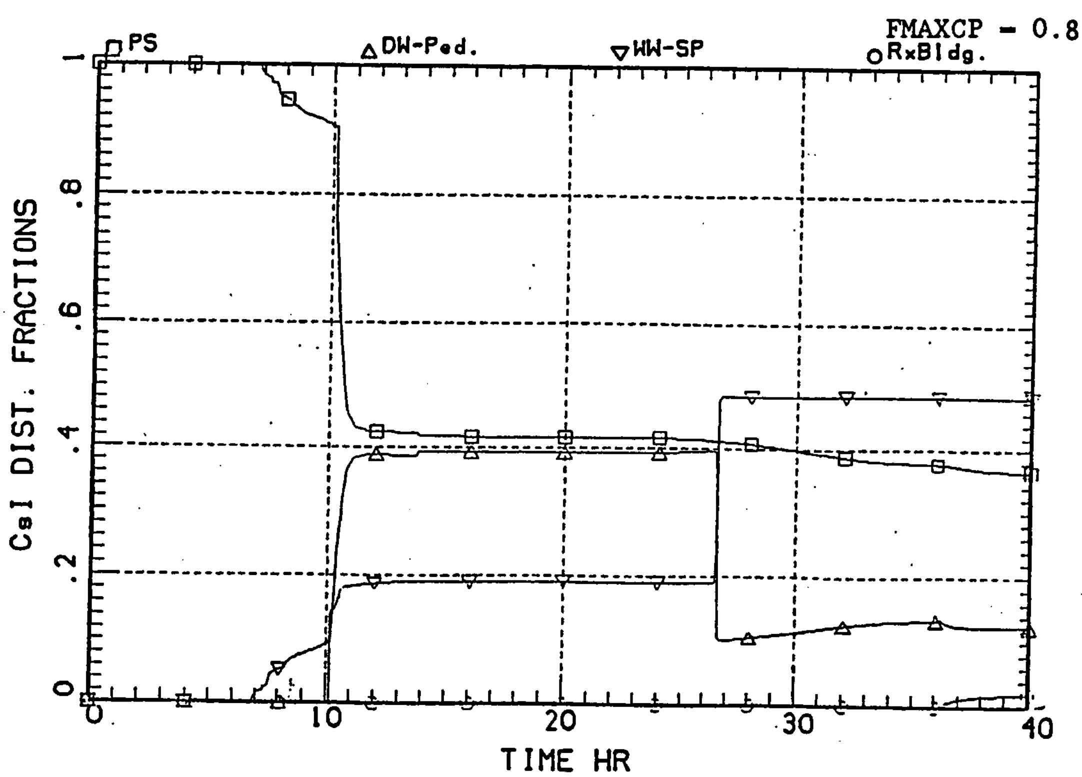
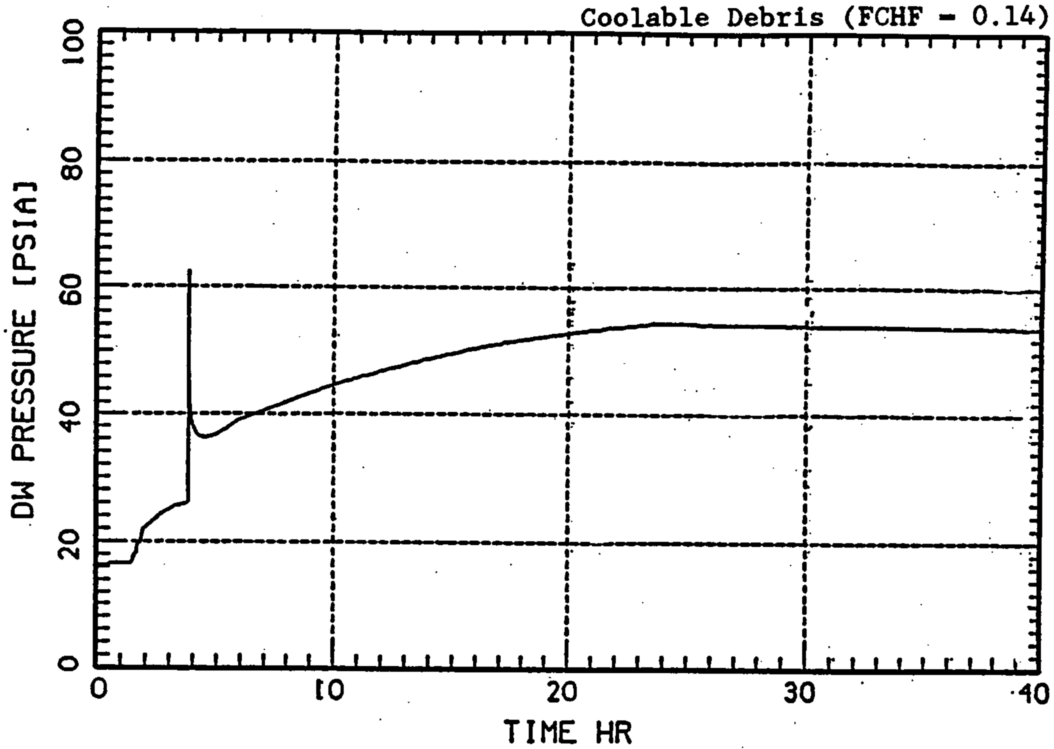


Figure 4.8-14

DEBRIS COOLABILITY SENSITIVITY - TQUX
Drywell Pressure



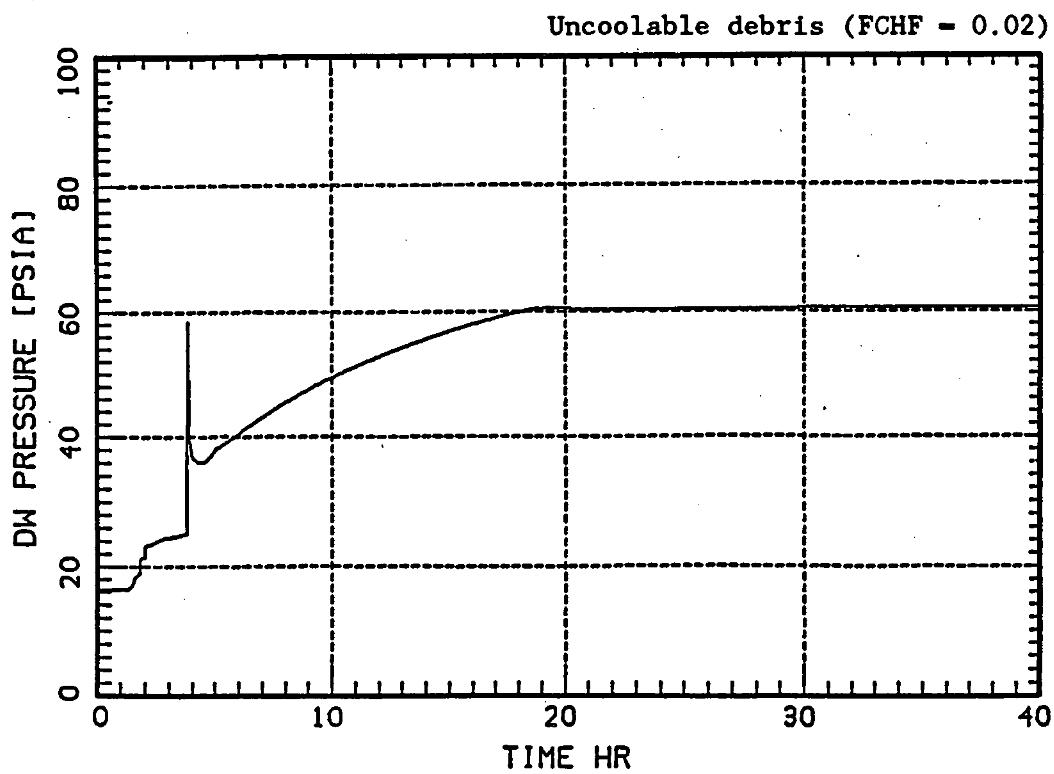
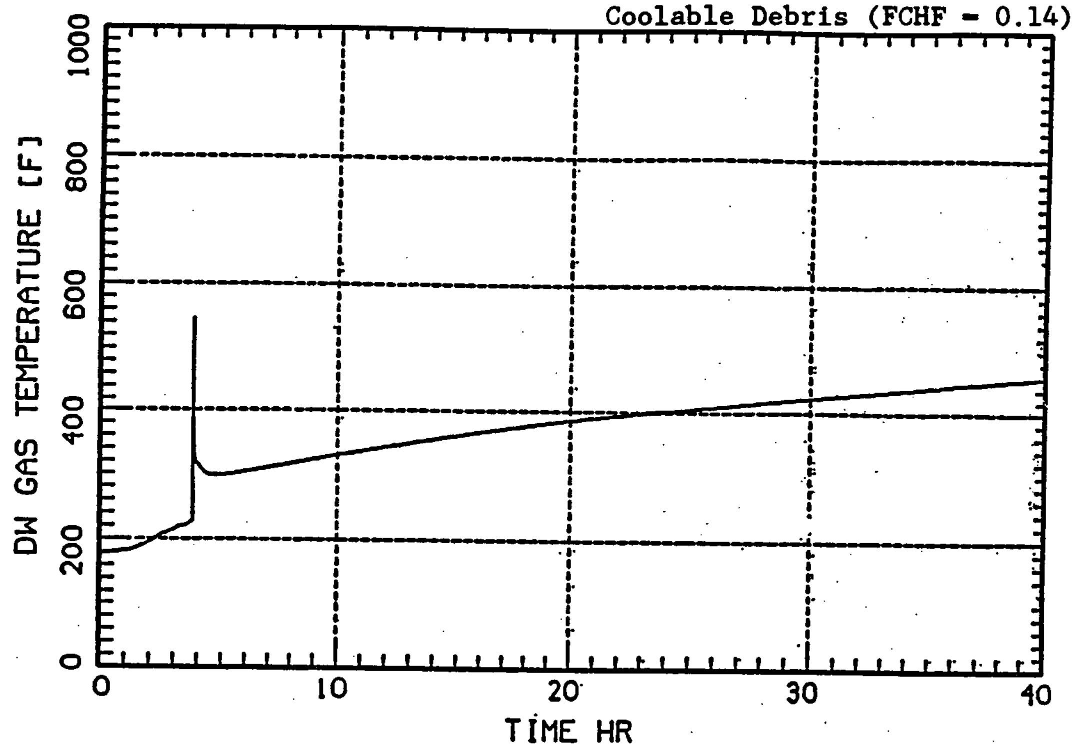


Figure 4.8-15

DEBRIS COOLABILITY SENSITIVITY - TQUX
Drywell Gas Temperature



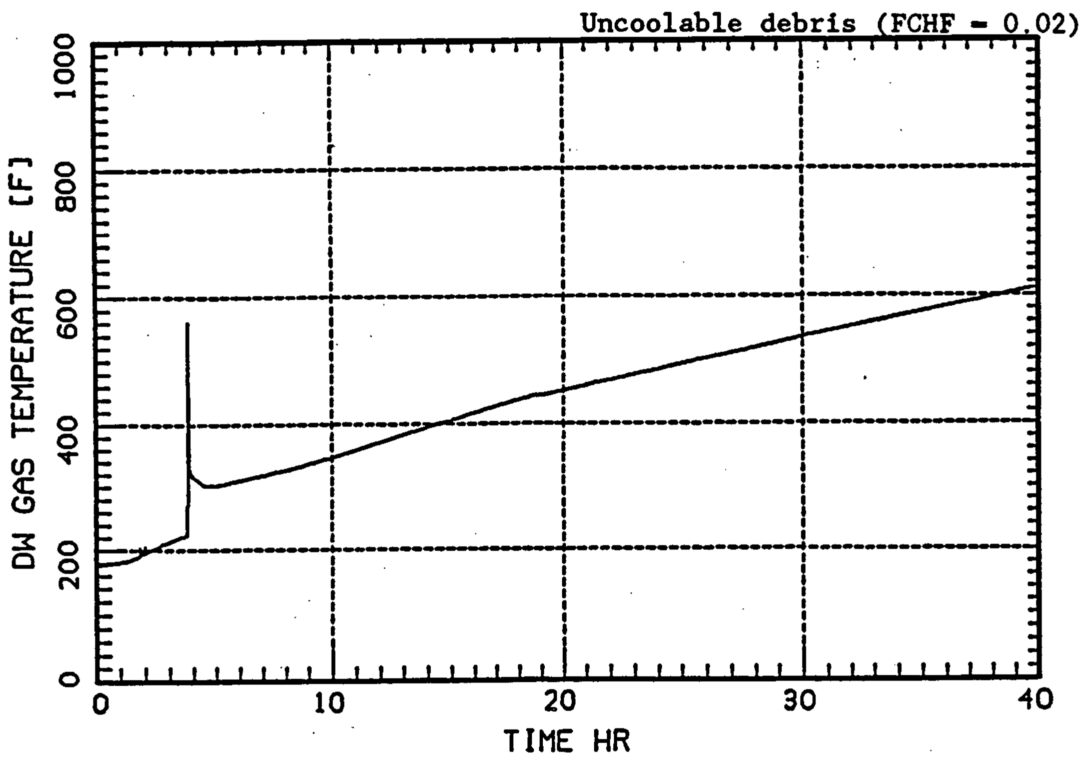
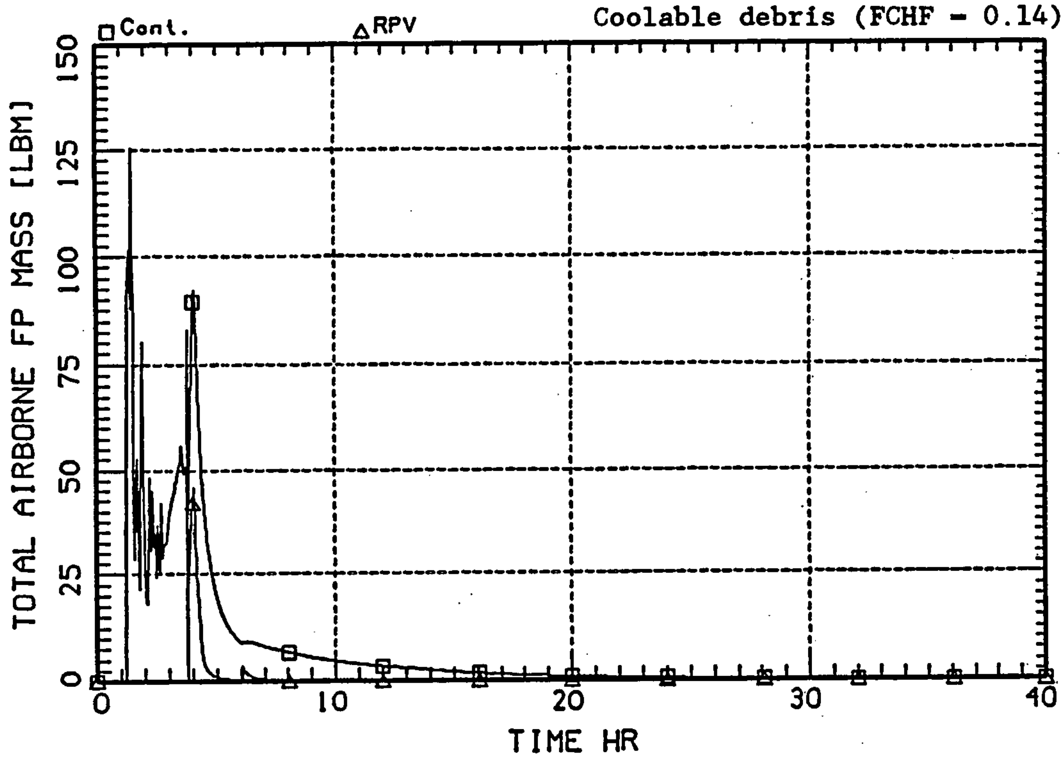


Figure 4.8-16

DEBRIS COOLABILITY SENSITIVITY - TQUX
Total Airborne Fission Product Mass

Coolable debris (FCHF = 0.14)



Uncoolable debris (FCHF = 0.02)

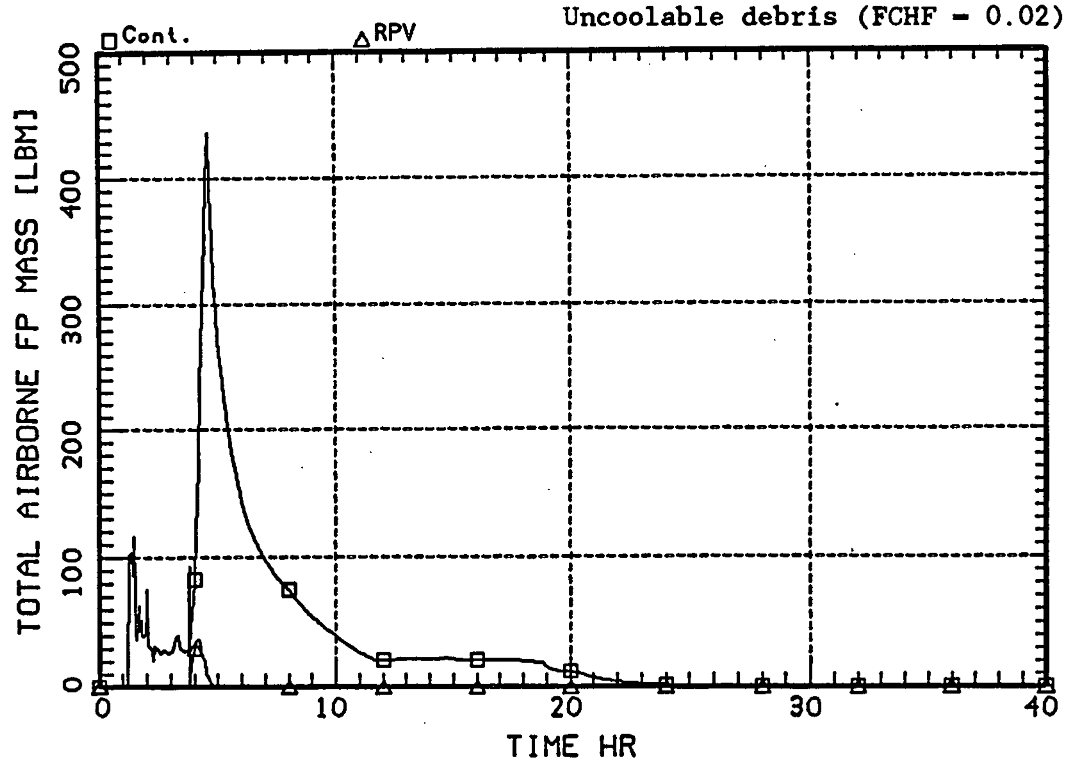
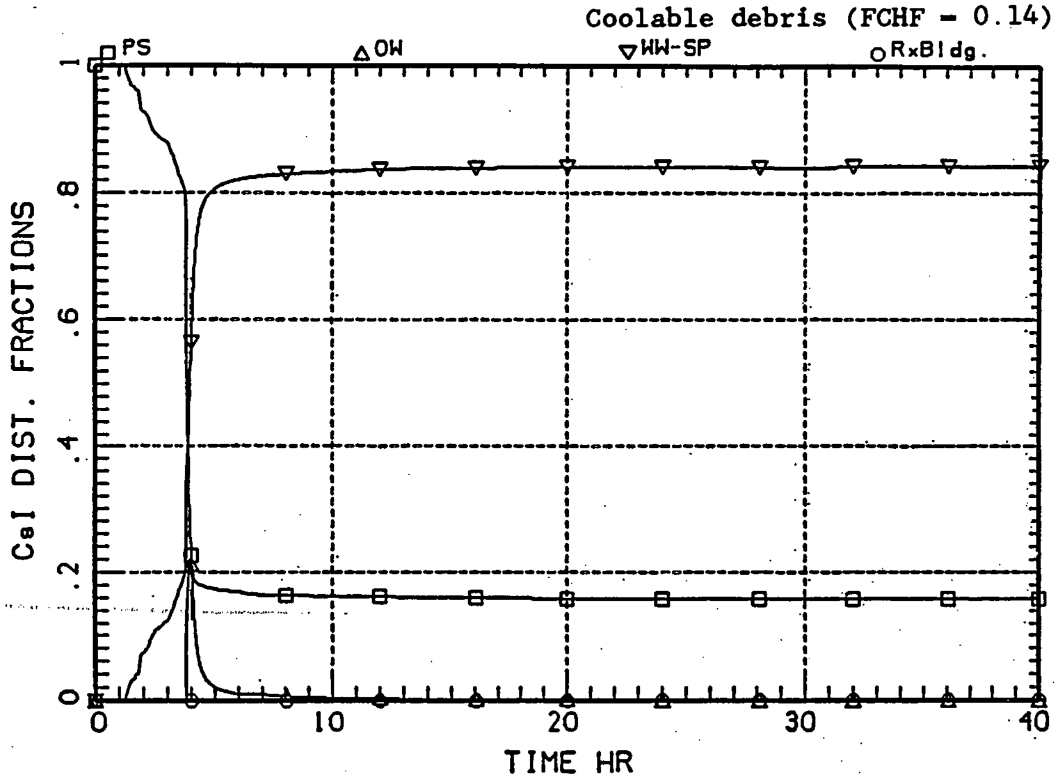
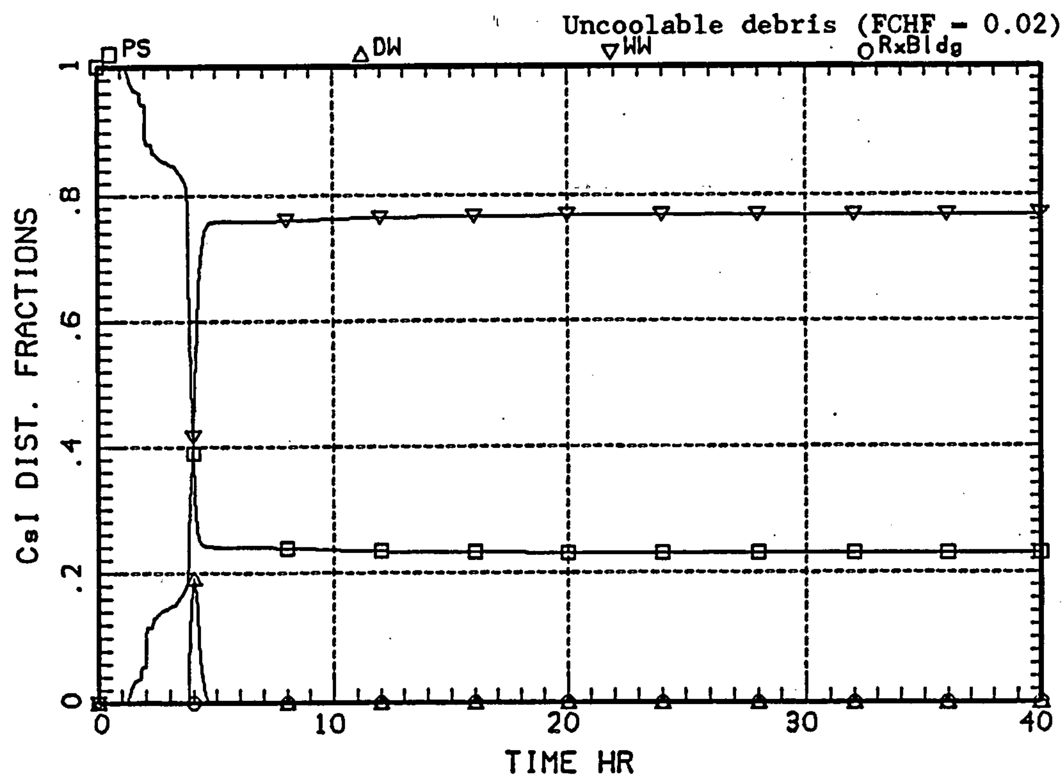


Figure 4.8-17

DEBRIS COOLABILITY SENSITIVITY - TQUX
CsI Distribution Fractions





5. UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE Program Organization

The organizational structure for the program is shown in Figure 5.1-1. The NSP Nuclear Analysis Department Manager has the overall review and approval responsibility. The NSP Superintendent Safety Analysis reports to the Manager of the Nuclear Analysis Department and is the NSP PRA/IPE program manager. The NSP Superintendent Safety Analysis is responsible for the details and overall project management for all PRA and IPE analysis at NSP. The NSP PRA staff working on the Monticello PRA/IPE was made up of five engineers. Two engineers are located at the Monticello site and the rest at the General Office. Having PRA staff at the site makes it easier to interface with the plant staff and to conduct walkdowns to ensure the PRA represents the as-built plant. Having PRA staff at the General Office makes it easier to interface with the other analysis groups, interface with management, and use the PRA staff for both Monticello and Prairie Island. The experience and training of the NSP PRA staff includes the following:

1. Two have had SRO licenses at Monticello. One was a shift supervisor. One Engineer is in SRO certification training.

2. The group has an average of 12.5 years experience in the nuclear field, with the maximum having 18.7 years and the minimum having 8.0 years.

3. All of the NSP staff are degreed engineers, which includes B. S. in Nuclear Engineering, M. S. in Nuclear Engineering, B. S. in Electrical Engineering, and B. S. in Chemical Engineering.

4. There is also experience in the following related areas: training, core transient analysis, operations, quality assurance, system engineering, plant technical staff, nuclear Navy and reactor physics.

5. The group is actively involved in industry committees and meetings. These include the steering committee of the MAAP users group, the BWROG Severe Accident Evaluation Committee, and the review team for the accident management Technical Bases Report.

6. Other activities include, being on the plant strategic planning committee, two ANS papers, and one ASME paper.

TENERA and Fauske & Associates Inc., which are part of IPEP (Individual Plant Evaluation Partnership), were used to help NSP develop the PRA/IPE. The IPEP program manager, from TENERA, reported directly to the NSP Superintendent Safety Analysis and provided NSP with a single point of contact for all IPEP activities.

The NSP PRA staff was involved with all aspects of the IPE. To ensure a complete understanding and to ensure the level 1 and 2 are properly integrated the same NSP PRA staff worked on both parts of the analysis. There was complete transfer of the technology to NSP including the use of the PRA computer codes, level 1 methodology, and level 2 methodology. The NSP PRA staff wrote the entire IPE report and is currently maintaining the PRA.

5.2 Composition of Independent Review Team

Four levels of review were done to ensure the correctness and that the NSP personnel are cognizant of the PRA/IPE.

The first review is the verification of the calculations to ensure the traceability of the input, correctness of the calculation, assumptions used, and that the results are correct. This was an independent review done by someone other than the preparer. Most of the calculations were prepared and verified by the NSP PRA staff with only a few calculations either prepared or verified by IPEP. In no case was a calculation both prepared and verified by IPEP. This was done to ensure a complete transfer of technology to the NSP PRA staff.

The second review is the review of other analyses performed in the industry. The Industry Degraded Core Rulemaking Individual Plant Evaluation (IDCOR IPEM) Methodology was developed initially and was then used as a starting point for the more detailed Probabilistic Risk Assessment (PRA) analysis. The IDCOR's IPEM used studies from Shoreham, Limerick, and Peach Bottom as sources for insights. NUREG-1150 was also reviewed for information specifically pertaining to Peach Bottom, since this plant most closely resembles Monticello. A representative from the Santa Maria de Garnora (NUCLENOR) plant in Spain visited Monticello. Since the Santa Maria de Garnora plant is very similar to Monticello, the PRAs were compared.

The third review is the review by the Senior Review Team (SRT). This is a team of three industry experts which reviewed the PRA/IPE to ensure correctness of the methodology and that the results are consistent with other PRAs in the industry. The team is made up of the following:

- Senior Vice-President, Fauske and Associates, Inc., who was the primary developer of the BWR IPE source term assessment methodology for IDCOR.

- Senior Project Manager, ERIN Engineering, who was the developer of the IDCOR BWR system analysis portion of the IPE methodology for IDCOR.

- Vice President, TENERA L.P., former utility manager to whom PRA and accident analysis personnel reported and currently a member of survival safety review boards.

- The forth review is the independent in-house review done by NSP personnel other than those on the NSP PRA staff. This is made up of NSP personnel not involved in the development of the PRA.

NSP plans to have a living PRA program to support the Monticello licensing, training, engineering and operations. The PRA input data, assumptions and models will be updated periodically to ensure the models reflect the current plant status. The NSP PRA staff is part of the modification process to ensure changes to the plant which could affect the PRA results are reviewed, and is on the strategic planning committee to help management determine the priority of proposed modifications.

The NSP PRA staff has already been involved with a significant number of support activities. Table 5-1 lists some of those activities.

FIGURE 5.1-1
PROGRAM MANAGEMENT STRUCTURE

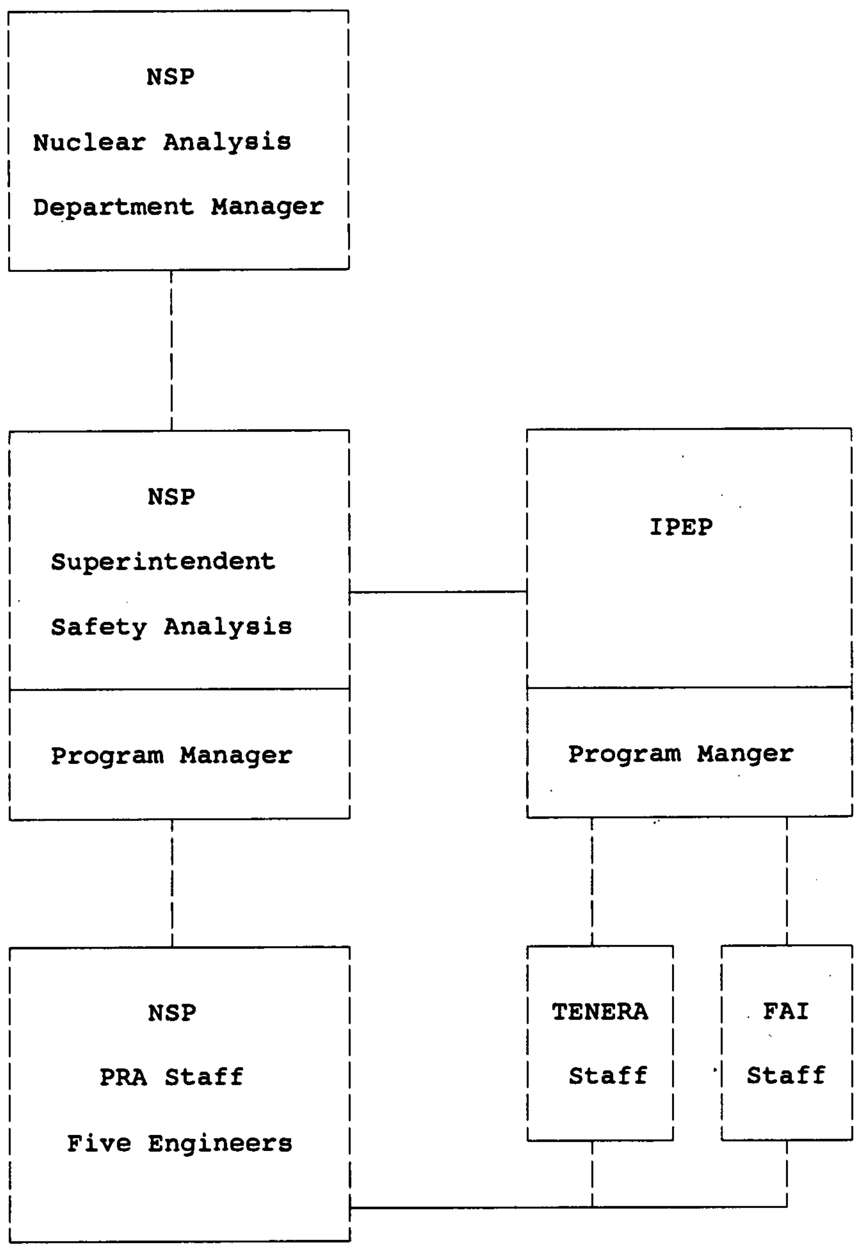


TABLE 5-1

MONTICELLO PRA APPLICATIONS

Study of LPCI flow diversion through torus cooling valve that fails to close.

Analysis of gas cylinder failure rates.

Analysis of drywell cooler failure rates.

Study of simultaneous LOCA and circuit breaker failure to open.

Feedwater pump lube oil pressure instrumentation study.

Design Basis Earthquake with check valve failure to close study.

Division II 250 VDC battery room heater seismic ruggedness study.

Turning gear oil pump study per TIL 968.

Containment vent line expansion bellows risk study.

Motor-Operated Valve importance ranking.

Feedwater pump discharge check valve stuck open importance.

Inflatable T-seal proposed modification benefit analysis.

Component and system rankings supplied to other NSP departments.

Recommendations to modify plant, improve training, and improve procedures.

Probability and consequences of HPCI flow diversion.

Evaluation of Fire/RHRSW crosstie modifications.

TIP Ball valve failure probability & consequences.

Component failure rates for the EDG ventilation system.

Evaluation of 2R transformer with and without CLiP installed.

Mod team member for hard-pipe vent.

Mod team member for 13 DG crosstie to safety load centers.

Probability of a LOCA, LOGP, and DC bus loss on the same day.

LPCI flow diversion probability.

Reliability of EDG with only one air start system working.

Benefit assessment of keep-full system modifications.

Assist with loose parts monitor safety evaluations.

TABLE 5-1 (continued)

MONTICELLO PRA APPLICATIONS

Advise on risk significance of inboard MSIV closure concerns.

Residual heat removal system reliability study.

Breaker coordination study.

480V AC diesel generator initial installation.

6. MONTICELLO IPE INSIGHTS AND RECOMMENDATIONS

6.1 Introduction

The purpose of this section is to present insights resulting from the IPE analysis. As noted in Section 2.0, an insight is defined as a unique design feature or operator action which drives risk either positively or negatively. Changes to plant design or operating procedures which may significantly lower risk are considered insights as well.

This section identifies those unique safety features at Monticello which are believed to impact risk from a severe accident. The following sections are broken down by damage classes, miscellaneous considerations, and containment performance improvement issues. The majority of the miscellaneous considerations come from discussions in Generic Letter 88-20 Supplement 2. The discussion includes:

1. Factors positively influencing the results.

2. Factors negatively influencing the results.

3. What can be done to improve plant safety, and how much the core damage frequency can reasonably be reduced where such an analysis has been performed.

6.2 Unique Safety Features of Monticello

This section identifies significant and unique safety features at Monticello which helped to minimize the risk from severe accidents. While presented by accident class, a number of features have a pervasive effect across many scenarios. These features limit the potential for challenges to core cooling and containment systems and assure the capability of these systems to cope with transients or accidents in general. A list of safety features follows:

1. The feedwater pumps are motor driven instead of turbine driven, which provides for reliable operation independent of the status of the MSIVs or main condenser which would be required if they were turbine driven pumps. The feedwater regulating valves will fail as is on loss of signal or air supply.

2. The condensate system is independent of all support systems with the exception of offsite power and DC power, if remote breaker operation is required. Plant operating experience has demonstrated that the condensate pumps can operate for extended periods without bearing cooling.

3. The offsite power switchyard has a highly reliable and diverse dual ring bus arrangement, minimizing the chance of loss of offsite power. Loads are normally operated from the 2R transformer which is not required to transfer on loss of the main generator.

4. Monticello has a variety of redundant and independent service water systems, minimizing the impact of loss of any single system. The ESW and EDG-ESW systems are backed up by the normal plant service water system. RHRSW is independent from the other service water systems. The RHRSW and fire system crossties can be emergency low pressure injection sources with the RHR system.

5. Equipment located in the Reactor Building does not require HVAC for extended periods. Plant tests and analyses have been performed demonstrate that the large rooms in the reactor building have sufficient heat capacity to significantly limit the temperature rise in the absence of room cooling.

6.3 Class 1A- Loss of High Pressure Injection and Failure to Depressurize

Analysis of the loss of feedwater sequence used plant specific experience regarding feedwater operation and recovery. Eight of the nine events initiated by a loss of feedwater have resulted in the early recovery of the system. On-line maintenance and valve failures have been the main causes of loss of feedwater events observed in the operating experience.

Factors positively influencing this damage class were the motor driven feed pumps, the offsite power configuration and the ability to easily recover feedwater if lost. In addition, other factors exist which positively influence this accident class and have not yet been credited in the PRA. These factors include a new digital feedwater control system and procedure changes dealing with avoiding flow diversion from the feedwater system when only one train is in service. These factors will reduce the possibility of a loss of feedwater event in the future.

Factors which have the potential to negatively influence this damage class were the condensate demineralizer bypass valve configuration and the dependency of long term operation of the SRVs on a key instrument panel. Both of these factors are discussed below.

The condensate demineralizers are provided with air-operated, fail closed isolation valves. A bypass line around the demineralizers would normally open on rising dp across the demineralizers. The normally closed condensate demineralizer bypass valve fails as is on sudden and complete loss of air. In this study, loss of air was assumed to result in loss of the condensate/ feedwater system because demineralizer valves will fail closed while the demineralizer bypass valve may not open due to the lack of pneumatic pressure. A modification was performed to assure faster operation of the bypass valve on loss of air.

Instrument panel Y20 powers AC solenoid valves which provide nitrogen to the SRVs for the purpose of depressurizing the reactor. Panel Y20 is not powered from essential AC (battery supplied), and the solenoid valves isolate on SBO as a result. Operation of SRVs, if required, is assured by accumulators until Y20 is repowered by a diesel or until offsite power is restored. In performing the PRA, the capacity of the accumulators is assumed to be limited and hence the operator action to restore power is important. Modifications are under consideration to supply power to the bottled nitrogen supply for the solenoid valves from an instrument panel that can be powered by an essential power supply or batteries. Reduction in CDF expected from this modification is 3.4E-6/yr.

A final factor which has both potentially positive and negative effects on this accident class is operator action to inhibit ADS on low-low reactor level. The PRA estimates that a large fraction of this accident class (approximately 80%) can be attributed to failure of operator action to depressurize the reactor. It is recognized that the EOPs instruct ADS inhibit to allow time for recovery of high pressure systems, permit low volume high pressure systems to recover level slowly, and to permit the maximum time possible to assuring low pressure systems are aligned and operating. The benefits of ADS inhibit and the importance of depressurization have been recommended for reinforcement in operator training.

In addition, a recommendation has been implemented to train the operators on key insights regarding this damage class. Besides reactor depressurization, operators have been trained on the impact of feedwater system recovery on reducing the risk of this damage class.

Because core damage initially occurs with an intact containment, the consequences associated with this class may not be as severe as some other classes for which containment failure is postulated. Factors positively influencing this accident class from a containment perspective include the ability to recover the core debris within the vessel, the sump configuration and the ability of low pressure systems to provide debris cooling on the drywell floor. Each of these factors is briefly discussed below.

As noted in Section 3.4.2.1, a large fraction of Class 1A events are a result of operator actions in depressurizing the reactor. Transient analyses have been performed to estimate the time available to initiate emergency depressurization in accordance with the EOPs. Using a Class A (TQUX) benchmark as an example, approximately 25 minutes are available to the operator from the time ADS is inhibited to the time that reactor level reaches the top of the fuel assuming no high pressure injection is available at all. Significantly more time is available if CRD is in operation or if decay heat levels are low. Assuming no recovery of high pressure injection, approximately 2 hours is available prior to the core slumping to the bottom of the vessel.

This sequence could be mitigated by recovery of high pressure injection; restoration of offsite power, if loss of offsite power is the initiator as it is for a large fraction of Class 1A sequences; or by operator action to depressurize the reactor. This assumes that the possibility for debris quenching and recovery within the vessel is high. Containment event tree quantification suggests that more than half of Class 1A sequences could be recovered in one of these ways.

If the scenario proceeds to the point of vessel penetration, blowdown of the reactor to the drywell at high pressure would occur. The suppression pool was assumed to absorb the energy of the blowdown limiting the containment pressure rise much in the way that it would for a LOCA. Debris exiting the vessel during the early stages of the blowdown may be dispersed in coolable form within the drywell. Debris that remains in the pedestal area was assumed to cool in the large equipment and drain sumps located beneath the vessel. These sumps are large enough to retain all of the debris expected to exit the vessel at the time of vessel penetration.

On vessel depressurization, low pressure systems such as core spray and LPCI will begin injection to the reactor providing cooling to the debris on the containment floor through the lower head.

The combination of the significant time for recovery in vessel, the retention of the debris in the sumps, and the systems available for long term cooling of the debris, provide substantial assurance that containment will remain intact for this accident class.

A factor which potentially has a negative influence on the outcome of this accident class is restrictions on use of the drywell sprays. The drywell spray initiation limit is provided in the EOPs to protect the containment from large differential pressures between the wetwell and the drywell on actuation of drywell sprays. Examination of drywell conditions during a Class 1A sequence demonstrates that at no time during the scenario is it expected to be within the drywell spray initiation limit. Early in the event, containment pressure and temperature are low and procedures do not call for actuation of sprays. This eliminates sprays as a potential source of debris cooling and aerosol scrubbing during this accident scenario. However, because of the availability of low pressure injection systems to the reactor and because the containment is likely to remain intact for this type of sequence, the importance of drywell sprays for this accident class is considered to be low. Pursuing relaxation of the drywell spray initiation limit through the BWR Owners Group Severe Accident Working Committee is the recommended course of action for this insight.

Beyond the SRV nitrogen supply solenoid power source modification, condensate demineralizer bypass line changes, and operator training, no significant reduction in core damage frequency or containment release is apparent for this accident class.

6.4 Class 1B- SBO and Failure of HPCI and RCIC

This damage class contributed the most to the total core damage frequency, and therefore has the greatest potential for recommendations to reduce risk. There were three dominant sequences involving SBO and failure of high pressure injection from various causes. The largest contributor to risk was HPCI and RCIC failure after 4 hours due to battery depletion. Another event involved random failure of HPCI and RCIC immediately after the SBO occurred. The last event involves a stuck open relief valve which causes depressurization below the HPCI and RCIC low pressure trip setpoints.

Factors which helped reduce the likelihood of this class of event are the switchyard configuration, the power recovery, the relatively high diesel generator reliability, and a battery capacity of reasonable duration, four hours. The Monticello emergency diesels have an overall reliability of >98% each. This reliability is based on several factors including an independent cooling system, ESW-EDG, backed up by non-safety service water, both of which are powered from the associated EDG. Maintenance provided only a limited contribution to system unavailability due to historical practice of not performing on-line preventive maintenance. The plant has also identified instrumentation that would be available during a SBO by placing a silver star next to it, or by engraving a star on the name tag as part of the control room panel upgrade.

Although the station batteries can last up to four hours, battery depletion after four hours with subsequent failure of HPCI and RCIC along with failure to recover AC power remains the most dominant sequence in the Monticello IPE.

Factors which negatively influence this event were:

1. There is a potential inability to depressurize the reactor vessel following battery depletion.

2. No AC independent low pressure injection source is available.

3. Procedures do not specifically provide actions to maximize the time the plant can survive this event.

The following recommendations were considered:

1. Operator training on station blackout was conducted which covered the following items:

   a) Shedding DC loads
   b) Operating HPCI/RCIC to minimize battery drain
   c) Diesel Repair
   d) Recovering diesel air start capability
   e) Breaker operation with degraded DC and no AC
   f) Restoring diesel field flash current
   g) Assuring adequate ventilation
   h) Aligning alternate diesel fuel supply
   i) Plant response for at least four hours

2. Procedure changes were drafted upgrading the steps to loadshed the station batteries in order to extend battery life if the diesels are not available. There is some guidance currently in the procedures. This procedure improvement will allow prolonged HPCI and RCIC operation, and extended SRV operation if pneumatic supplies are available with other modifications. This will allow more time for recovery of offsite power and the diesels. Station batteries are currently assumed to be unavailable after 4 hours without charging. Load shedding batteries to provide 2 extra hours of capacity is estimated to produce a reduction in CDF of 3.4E-6/yr.

3. A recommendation was made to add procedural steps to commence a controlled cooldown as soon as possible. Review of plant procedures concluded that this action is proceduralized, although not very detailed. It was stressed further in training as a result. This recommendation may extend the time available to use SRVs or prevent containment failure. The act of depressurizing during a station blackout could have an effect similar to extending battery life another 2 hours. As the vessel repressurizes after failure of station batteries, no loss of inventory occurs and water density will change, actually causing level to go up by swelling. After the vessel pressurizes to the point at which the relief valves lift, decay heat levels are lower and level will take more time to get to the top of active fuel.

Based on transient analysis results, core damage would result at 8 hours if the reactor was depressurized to 100 pounds before battery depletion. The estimated reduction in CDF from depressurizing during SBO is 3.4E-6 per year.

The effect of load shedding batteries and depressurizing were analyzed together. Assuming batteries lasted for 6 hours from load shedding and the reactor was depressurized below 100 pounds before battery failure, core damage would not be estimated to occur until about 10 hours. The reduction in the CDF for the combination of these two changes is estimated to be 5.0E-6 per year.

4. A recommendation was made to supply station battery chargers with an AC independent power supply to extend battery life. This would provide power to the station batteries for prolonged SRV, HPCI and RCIC operation. A 480V diesel generator is currently on-site which could be modified to allow an alternate power source for battery charging. RCIC and HPCI operability would then only be questionable for extended operation during SBO as a result of room cooling or high suppression pool temperature and pressure. The reduction in CDF was estimated at 8.2E-6 per year. This is under consideration by the plant.

5. The plant was modified to allow the diesel fire pump to be aligned as a reactor vessel injection source. The primary value of this modification from a risk standpoint was to prevent core damage during a SBO. The total reduction in CDF for the diesel fire pump was estimated at 4E-7/yr. The benefit of the fire pump by itself is limited because: a) the fire pump will not prevent core damage unless ADS capability is increased by extending battery life, and b) there is little impact on other accident classes due to the availability of numerous other low pressure systems such as LPCI, core spray and condensate.

6. A recommendation was made to modify the SRV pneumatic solenoid valves. This change is the same as that identified for Class 1A. Loss of all AC power will currently eliminate all long term pneumatic supplies to the SRVs with the exception of the accumulators. This would further reduce or eliminate any benefit associated with a fire system to RHR cross connect because the vessel may not remain depressurized for an extended length of time. Core damage would occur at high pressure in the reactor vessel for nearly all SBO cases as well. Changes to the power supplies of the drywell pneumatic supply valves are under consideration. This will allow continuous SRV pneumatic supply for most loss of offsite power events with only simple operator action. This change is also necessary in order to benefit from recommendations associated with operator actions to extend time to core uncovery during a SBO (see items 1 and 3).

Similar to Class 1A several items positively influence containment response to a station blackout.

1) Recovery in-vessel. Over eight hours is estimated to the point that the core would exit the bottom of the vessel as a result of a station blackout with battery depletion. Restoration of an offsite or onsite power source anytime during this period provides for the recovery of any of a number of AC powered injection systems.

2) Sump configuration. Upon exiting the vessel, a significant amount of core debris is expected to be retained within the sumps located in the pedestal area, effectively precluding debris contact with the containment liner.

3) Recovery in-containment. With most of the debris in the sump the most likely challenge to containment would be a result of over pressure or excessive drywell temperatures. Substantial time for recovery is available for these types of challenges. More than eighteen hours is estimated to the point that overtemperature failure of containment would occur even in the absence of debris cooling. Recovery of an AC power source at anytime up to this point provides for recovery of debris cooling, DHR systems, and protection of the containment.

Potential negative factors which influence the outcome of station blackout containment response at Monticello include:

1) No AC independent means of debris cooling.
2) No AC independent means of DHR.

The proposed 480V DG nodification, in addition to the completed diesel fire pump crosstie, would have a positive influence on the above factors. They would provide water to the debris on the containment floor. This source of debris cooling limits the temperature rise in the drywell and would also extend the time to containment failure on overpressure. An AC independent means of DHR could be considered in the form of the hard piped vent, planned in response to Generic Letter 89-16. The benefits of the vent are limited during SBO conditions because venting is not initiated in time to prevent overtemperature failure when no debris cooling exists and the power supply to the vent valves would be unavailable due to battery depletion. The benefits of the vent are further limited by the recovery of AC power during the time frame required to reach containment overpressure. This time frame is more than a day.

The most reduction in CDF that can be reasonably expected for this class IB event was 9.1E-6 per year considering improved ADS power supply and enhanced battery charging from the onsite 480V diesel. A fire crossconnect as an injection system was also considered useful to reduce the potential for core damage during non-battery depletion sequences and to maximize the time available for AC power recovery prior to containment challenge on overpressure or temperature.

6.5 Class 1D- Loss of High and Low Pressure Injection Systems

These events involved loss of high and low pressure injection sources with core damage occurring at low pressure with an intact containment. This class of sequences contributed little to the total core damage frequency because of the redundancy of coolant injection systems.

Factors which positively influenced this damage class were:

1. Reliable and redundant low pressure injection systems exist in the form of condensate, LPCI, and core spray.

2. The existence of a plant procedure dealing with loss of level indication (RPV Flooding C.5-2006) minimizes the uncertainty associated with this event.

No factors were observed which negatively influence this damage class in a significant way.

The following recommendations are noted based on qualitative insights:

1) While given limited or no credit in the IPE, improvements in the capability of manually aligned, backup low pressure injection systems could be made. These systems include:

   RHRSW through LPCI
   Condensate service water
   Service water to the hotwell

   While called out in the EOPs there are no procedures available that describe the alignment or operation of these systems in this manner. Development of procedures for the use of these systems, as injection sources would further assure the reliability of these backup injection systems.

2. Operators were trained on the limitations of the EOP pump curves, when the curves do not apply, and what to do when the curves do not apply. The EOP pump curves for RHR and core spray do not cover all possible operating conditions. The EOPs permit operation above the curves under certain conditions. Operators were trained on use of the pumps above the curves, particularly when cavitation is possible.

6.6 Class 2- Loss of Decay Heat Removal

This class of events contributed little to the total core damage frequency. These events involved loss of RHR, the main condenser, and failure to vent containment before containment failure. Core damage, if it were to occur, would occur with a failed containment so the consequences would be more severe than core damage sequences with an intact containment.

Factors which positively influence this accident class include the following:

1) Early recovery of DHR systems. Monticello plant operating experience suggests a large fraction of transient initiators which might lead to loss of one or more decay heat removal systems can be recovered quickly from the control room or with simple operator actions outside the control room. Initiators having these characteristics are:

   MSIV closure
   Loss of main condenser
   Loss of instrument air
   Loss of service water

2) Significant time for repair of DHR systems. Even if a total loss of all DHR systems occurs, the heat capacity provided by the suppression pool is significant. Two to three days is required to pressurize the containment to cause failure from decay heat levels. Repair of DHR systems during this period prevents further pressure rise and preserves the integrity of containment. It is recognized from the DHR analysis that recovering any of the three systems credited in the PRA is dependent on containment pressure, but all sequences provide significant time for recovery.

System	Containment Pressure	Limitation
Main Condenser	45 psig	MSIV operator dp
Vent	62 psig	Butterfly valve operation
RHR	62 psig	Personnel safety in reactor building
DHR Support Systems	103 psig	Containment capacity

3) Limited impact of DHR loss on vessel injection as containment pressurizes. Monticello has large CSTs with a normal inventory capable of making up for decay heat losses for several days. Makeup sources from the CSTs include all of the high pressure injection systems. As long as any of the high pressure systems continue to operate (Feedwater, CRD, RCIC or HPCI) makeup to the reactor can continue regardless of reactor or containment high pressure.

4) Limited impact of containment venting or containment failure on operation of reactor building or turbine building systems. The environment in the turbine building is not affected by the integrity of the containment. Evaluation of the corner rooms following containment venting or containment failure demonstrates that the environment remains below qualified temperature, humidity, and pressure.

Factors having potentially negative effects on this accident class:

1) Two of the three DHR systems credited in this study, the main condenser and containment venting, are dependent on common support systems, instrument air and service water. These dependencies are offset by the independence of RHR from these support systems and the ability of the operator to recover from loss of these support systems with simple actions outside the control room.

2) Operation of the main condenser, containment vent and one train of RHRSW are dependent on a single instrument panel, Y20.

The following recommendations are being considered for this accident class:

1. Operator training of recovery of a failed RHR system. Examples include:

   a) Pump cavitation
   b) Fouled heat exchanger

   Training on recovery of a failed main condenser with degraded or failed support systems. Examples include:

   a) Air ejectors
   b) Mechanical vacuum pump
   c) Condensate pumps
   d) Feedwater heaters
   e) Bypass valves
   g) MSIVs
   h) Circulation Water
   i) Service Water
   j) Service air system including circulation water pump trip, feed regulating valves, condensate demineralizer bypass valve, and circulation water valve operation.
   k) Gland seal system

2. Air receiver tank discharge valves for the plant air system were formerly locked open. They are no longer locked, so they could easily be closed to isolate a receiver tank with a stuck open relief valve to prevent a possible loss of air scram. A stuck open air receiver tank relief valve has been the cause of the only loss of instrument air initiator experienced at Monticello.

3. Write a procedure for the emergency replenishment of the CSTs. The action to refill the CSTs was initially considered in the Monticello IPE analysis. It was determined that over 200,000 gallons were normally present in the CSTs which would be adequate for approximately 3 days of normal decay heat makeup injection flow rate. Because the volume of water

would last several days, makeup is available from the demin water storage tank or radwaste, and the fire system was available as a refill source, failure to refill the CSTs was not considered a likely failure mode of the CSTs.

4. A hardpipe containment vent from the torus to outside the containment building is planned. This will provide another independent means to vent the containment and allow for heat removal. A hardpipe vent can improve the probability that containment integrity can be maintained and maintains a habitable environment in the reactor building during venting operation. Because of the ability of systems in the turbine building to provide adequate makeup and because of the limited impact on core cooling systems provided by the existing vent, the impact of this modification on the core damage frequency calculated in the IPE was not significant.

Given the multiple and diverse means of providing DHR that exist at Monticello, and the low potential for core damage even if a loss of DHR were to occur, few additional recommendations can be identified to further reduce the significance of this accident class.

6.7 Class 3- LOCA

The low probability of this class of events is strongly influenced by low initiating event frequencies and high ECCS reliabilities.

One potential factor which negatively impacted the results was the LPCI loop selection logic. This four pump, two loop system can only be credited as a single loop system because of the loop selection logic. Even then the accident class as a whole does not significantly contribute to risk.

Because of reliable operation of multiple and diverse means of reactor makeup, specific insights for Monticello are not easily determined. No means for significant reduction in core damage frequency was evident for this event class.

6.8 Class 4- Anticipated Transient Without SCRAM

The dominant ATWS sequences were composed of transient initiating events followed by a mechanical failure of control rods to scram. These events were highly influenced by the failure probability of the control rods to mechanically insert and the operator failing to inject boron before containment failure. These are relatively fast acting events, postulated to result in containment failure and core damage on the order of one hour.

Factors positively influencing ATWS quantification for Monticello include:

1) Relatively limited demands on RPS. Monticello has historically experienced 3 to 4 reactor trips per year from power. A relatively large fraction of these have been manually initiated or have been due to spurious off normal conditions, about one per year. For these spurious events, operation of the plant would continue if a failure to trip occurred, allowing significant time for operator action to insert rods or actuate SLC.

2) RPS reliability. The generic value of 3E-5 per demand was used for a RPS failure rate. The presence of ARI effectively eliminates two thirds of this failure rate by providing an effective diverse rod insertion signal.

3) Monticello has a large SRV capability at >70% rated steam flow. This is significantly greater than the steam flow expected following an ATWS even if only one of the recirculation pumps were tripped.

4) SLC enriched boron allows for shutdown of the reactor within 12 minutes of initiation using only a single pump.

One factor has a potentially negative effect on the results:

1) The turbine bypass capacity is relatively small at Monticello. As a result, a turbine trip with bypass could result in heatup of the containment at a rate similar to MSIV closure until the operator takes action to initiate power/level control as specified in the EOPs.

Recommendations were made for the following items which are either approved or under evaluation by plant staff:

1. Operator training was conducted on the significant insights regarding ATWS.

2. Remove the actions for mechanically bound CRDs to a contingency procedure in the EOPs, so that the operator will focus on reactor shutdown with SLC.

3. Test the CRD boron injection hoses. The hoses could collapse when used, or deteriorate, which would eliminate this boron injection option. Alternate boron injection has a very small impact on overall plant CDF. It is still worth testing the injection path to make sure the hoses and pump will work if needed.

Because of system reliabilities for SLC, RPS, control rods, and the initiating event frequencies, no hardware modifications could be identified that would result in a significant reduction of CDF for this damage class. Continued emphasis on ATWS in operator training is the best means to minimize the significance of this accident class.

6.9 Class 5- Unisolated LOCA outside Containment

This damage class contributes less than 1% to the total core damage frequency. Factors positively influencing the results of this accident class include:

1) Operator actions to depressurize the reactor on indications associated with LOCA outside containment.

2) The reactor building environment is such that only equipment located in the vicinity of the break is expected to exceed environmentally qualified limits.

Because of the low likelihood for the event and low potential for core damage given the event, no additional recommendations were identified.

6.10 Containment Performance Improvement Issues

1. Alternate water supply for drywell spray/vessel injection. Monticello currently has a connection from RHRSW to RHR. A modification has also

been completed to connect the fire system to this same line. Crediting the RHRSW and fire systems as a vessel injection source not associated with SBO has limited value because a dominant failure mode of RHR and core spray is the common failure of the injection valves.

2. Enhanced ADS system. Plans are under consideration to modify the power supply for the nitrogen bottles to the SRVs so that the SRVs will remain available during a loss of offsite power.

3. Implementation of revision 4 of the EPGs. The primary differences between revisions 3 and 4 of the EPGs was examined. No specific sensitivity studies were performed because it was determined that none of the changes significantly change any action modelled in the IPE. They also did not add or take away the option to use a system for core or containment cooling which was not already included in the IPE.

6.11 Lowest Core Damage Frequency With Modifications

The total CDF after recommended changes are made, was calculated by considering the new level 1 CDF with automatic ADS (no operator action to inhibit), and AC crossconnect to essential power during SBO. With these changes the new estimated CDF value is 9.3E-6/yr. This value did not include the effect of the diesel fire pump crosstie.

7. TRANSIENT ANALYSIS

A number of transient events and accident sequences were analyzed (a) to establish the minimum equipment required to bring certain events to successful endstates; (b) to establish the relative timing of key events during various classes of accident; and (c) to determine whether the conditions in the reactor building following a containment failure are severe enough to disable the equipment located there. These analyses were done using the Modular Accident Analysis Program (MAAP) 3.0B, BWR Revision 7.0, with best-estimate, Monticello-specific input parameters. MAAP was also used for source term assessment as described in section 4.7.

7.1 Success Criteria

The success criteria cases give best-estimate determinations for Monticello of the minimum equipment needed in order to bring various accidents to successful end states.

7.1.1 LPCI Success Criteria

7.1.1.1 LPCI Success for a Large-break LOCA

In this case, the purpose is to determine the minimum number of LPCI pumps needed to prevent core damage during a large-break loss-of-coolant accident, assuming that no other source of injection is available. The LOCA modeled is a 3.7 ft^2 break at the junction of the reactor vessel with a recirculation pump suction pipe.

When this break is modeled with no injection available except one LPCI pump, the actual water level reaches the top of active fuel in 17 seconds, and LPCI injection begins in 21 seconds. The water level drops briefly to about 1/3 core height; the LPCI flow then raises level to the top of the jet pumps (about 2/3 core height) and maintains it for the remainder of the event. No damage to fuel or cladding occurs. [Figures 7.1-2 and 7.1-3.]

It is concluded that one LPCI pump is sufficient to prevent core damage during a large-break LOCA.

7.1.1.2 **LPCI Success for Loss of Injection with ADS**

The purpose of this case is to determine how many LPCI pumps are needed to prevent core damage if all other injection is lost and ADS is initiated when indicated reactor level drops to the top of active fuel.

This event is initiated by a loss of feedwater, followed by the failure of all high-pressure injection systems. The automatic depressurization system (ADS) is inhibited initially, so the reactor vessel remains near its normal operating pressure of 1000 psig while the water in the vessel gradually boils off. After 8.8 minutes, the indicated level has dropped to the top of active fuel, and the operator opens all three ADS valves to depressurize the vessel and enable the low-pressure injection systems; for this case, no low-pressure injection is available except for a single LPCI pump. The actual water level reaches the top of active fuel at 11 minutes, and at 12.2 minutes the vessel pressure is low enough that LPCI flow is established. The actual level is recovered to the top of active fuel at 20 minutes, and to +48" at 24 minutes. In all, the actual level is below the top of active fuel for less than 9 minutes, and no damage to the fuel or cladding occurs during this event. [Figures 7.1-4 and 7.1-5.]

It is concluded that one LPCI pump is sufficient to prevent core damage during a loss of all other injection if ADS is initiated when indicated level reaches the top of active fuel.

7.1.2 Core Spray Success Criteria

This case is to determine the minimum number of core spray pumps needed to prevent core damage during a large break LOCA. As with the LPCI LOCA case (section 7.1.1.1), the LOCA modeled is a 3.7 ft^2 break in one of the recirculation pump suction lines. With no injection available during this LOCA except for 1 core spray pump, the actual water level reaches the top of active fuel in 17 seconds, and core spray flow is established within 21 seconds. Level is quickly recovered, and the boiled up level in the core is maintained slightly above the top of the jet pumps for the remainder of the event. No damage is done to either the fuel or the cladding. [Figures 7.1-6 and 7.1-7.]

It is concluded that one core spray pump is sufficient to prevent core damage during a large-break LOCA.

7.1.3 Control Rod Drive (CRD) Success Criteria

The CRD success criteria cases were done to determine the extent to which flow from the CRDs into the reactor can prevent damage to the core when other injection sources are limited or unavailable.

The CRD system has two pumps which can be operated concurrently, but under normal plant operating conditions only one pump is in use. The amount of flow these pumps deliver to the reactor vessel increases when the reactor is scrammed; this increased flow continues until the scram is reset. For all of the following cases, the event begins with only one CRD pump operating, delivering the flow which is expected from one pump when the reactor is scrammed, the scram has not been reset, and the system is aligned normally. At normal operating pressures, this is about 100 gpm.

The operator can increase CRD flow by either realigning certain valves in the system piping, starting the second CRD pump, or both. This is considered in some of the following cases by increasing the CRD flow after first allowing a lapse of time in which the operator may take action.

7.1.3.1 1 CRD Pump, No Operator Action

This event is initiated with a loss of feedwater, followed by a loss of all injection except for 1 CRD pump, which provides its normal post-scram flow with the scram not reset and the system in its normal configuration. ADS is inhibited throughout this sequence. After 21 seconds, an indicated reactor level of -48" causes MSIV closure and trips the recirculation pumps. The indicated level drops to the top of active fuel in 16 minutes, though the actual boiled-up level in the core does not reach that level until 1.1 hours. [Figures 7.1-8 and 7.1-9.]

The actual core level reaches a minimum of about 1/2 core height at 1.9 hours; by that time, decay power has decreased to the point that the boil-off rate is less than the CRD flow, and the level therefore begins to rise, regains the top of active fuel at 6.2 hours, and continues rising for the remainder of the analysis.

As long as the core is more than 2/3 covered, the peak core temperature remains near 550°F. When the actual level drops below 2/3 core height, the peak temperature in the core rises quickly, reaching 1300°F about 2 hours into the event, and then ranges between 1300°F and 1400°F until 3.1 hours. At that time the core level is back up to 2/3 core height, and the peak core temperature drops steadily back to 550°F. Although the temperature of the fuel is elevated during this 1.1-hour period when the level is below 2/3 core height, no damage occurs to the fuel or the cladding.

It is concluded that in this sequence, the flow from a single CRD pump is sufficient to prevent core damage without addition of water from other systems or operator action to maximize flow.

It should be noted that this conclusion differs from the assumptions made about the adequacy of CRD injection in the quantification of the PRA sequences. As noted in section 3.4, core damage was defined as an extended period of time with reactor level below 2/3 core height. However, the analysis described above shows that although reactor level is less than 2/3 core height for approximately two hours, the peak cladding temperatures remain less than that at which significant oxidation occurs (approximately 2200°F). It appears that simply by changing the definition of core damage into terms of clad temperature rather than reactor level, a number of accident sequences could be reclassified as having no core damage, notably in accident classes 1A and 1D.

7.1.3.2 1 CRD Pump, Operator Realigns System

This event is the same as the event described in 7.1.3.1, except that after 30 minutes the operator realigns the CRD system in order to maximize flow to the reactor vessel.

The event begins with a loss of feedwater, followed by a loss of all injection except for a single CRD pump. ADS is inhibited, so the reactor vessel remains near its normal operating pressure. After 30 minutes, the operator increases flow to the vessel by realigning valves in the CRD system.

The first 30 minutes of this event are the same as in 7.1.3.1 - at 21 seconds, an indicated level of -48" causes MSIV closure and trips the recirculation pumps, and at 16 minutes the indicated level has dropped to the top of active fuel. At

30 minutes, when the CRD system is realigned, the actual level is still roughly two feet above the top of active fuel. With the extra flow from the second pump, the reactor water level begins to rise, reaching +48" at 3.5 hours. [Figures 7.1-10 and 7.1-11.]

The core remains covered throughout this event, and no damage occurs to the fuel or the cladding. It is concluded that if the operator successfully realigns the CRD system within 30 minutes, the flow from a single CRD pump is sufficient to keep the core covered and to prevent core damage.

This analysis indicates that operator action to realign CRD flow might lower the frequency of class 1A and 1D sequences; this is currently not credited in the PRA. It is noted, however, that a strong degree of coupling may exist between action taken to realign CRD flow and actuation of the ADS, as these are both manually initiated actions based on falling reactor water level. This is the basis for not crediting both actions in the current sequence quantification.

7.1.3.3 2 CRD Pumps, Operator Realigns System

This event is the same as the event described in 7.1.3.1, except that (1) ADS is initiated when the indicated level reaches the top of active fuel, and (2) after 30 minutes the operator starts the second CRD pump and realigns valves in the CRD system to maximize flow to the reactor vessel. The purpose of this analysis is to determine if CRD can prevent core damage during sequences in which ADS is initiated.

The event begins with a loss of feedwater, followed by a loss of all injection except for a single CRD pump. ADS is inhibited initially, so the reactor vessel remains near its normal operating pressure. At 21 seconds, an indicated level of -48" causes MSIV closure and recirculation pump trip. At 16 minutes, the indicated level reaches the top of active fuel, and the operator depressurizes the vessel by initiating ADS. With the ADS valves open, the actual water level falls to the top of active fuel at 18 minutes and continues to drop. The fuel temperature drops initially during depressurization, then begins to rise as the core becomes uncovered. [Figures 7.1-12 and 7.1-13.]

At 30 minutes the operator starts the second CRD pump and realigns the system in order to maximize flow to the vessel. With this increased CRD flow, the actual level in the core begins to rise from its minimum of about 1/4 core height, and the maximum fuel temperature falls from a peak of 1350°F at 40 minutes to 250°F at 1.2 hours. The indicated level, which is measured outside the core shroud, remains at the bottom of the jet pumps until 1.3 hours, since this region does not begin to fill until the core level is high enough to spill over the tops of the jet pumps. The core is completely covered at 2.4 hours, without having sustained damage to the fuel or cladding.

It is concluded that the CRD system can provide sufficient flow to prevent core damage following ADS actuation if the operator succeeds in starting the second CRD pump and realigning the CRD system.

Similar to the case described in section 7.1.3.1, reactor level remains below 2/3 core height for more than an hour in this analysis; this situation is defined as causing core damage in the sequence quantification. For this reason, flow from the CRD system alone following ADS actuation is not given credit for preventing core damage. However, redefining core damage in terms of cladding temperature may result in reduction of the frequency of core damage for certain accident classes, as in this class 1D accident.

7.1.3.4 CRD/HPCI Success with 1 SORV

This sequence begins with a single stuck-open relief valve (SORV) and a loss of feedwater. All injection fails except HPCI and one CRD pump. The indicated level drops to -48" in 21 seconds, causing the MSIVs to close, the recirculation pumps to trip, and initiating HPCI. At 32 minutes, HPCI fails due to low vessel pressure. The CRD pump continues to inject, and at this stage of the event, with the vessel depressurized and the decay heat reduced, CRD flow is sufficient to make up for the water lost through the SORV. Actual level is maintained above the -48" level throughout the event, and the core and cladding remain undamaged. [Figures 7.1-14 and 7.1-15.]

It is concluded that HPCI and a single CRD pump are sufficient to prevent core damage in the event of a single stuck-open relief valve without the need for ADS or operation of low pressure systems.

7.1.4 Safety/Relief Valve (SRV) Success Criteria

Two MSIV closure cases were used to determine how the peak pressure in the reactor vessel is affected by the number of SRVs available. In the first case, all eight SRVs operate normally in response to the MSIV closure; in the second case, only one SRV opens while the other seven remain closed.

During an MSIV isolation, a scram signal is generated when the valves are 10% closed. Besides scramming the reactor, the scram signal causes the opening setpoints of three of the SRVs to be lowered to 1052, 1062, and 1072 psig, respectively. The other five SRVs open at their mechanical setpoints, which range from 1100 psig to 1120 psig in this analysis.

For the first case, the MSIV closure initiates the scram and lowers the SRV setpoints; each of the eight SRVs opens when its setpoint is reached. The maximum pressure inside the reactor vessel during this event is 1120 psig. For the second case, one SRV is allowed to open when vessel pressure reaches 1120 psig; all others remain closed. The maximum vessel pressure in this case is 1297 psig. This is well within ASME code limits for vessel and piping design.

It is concluded that one operable SRV is sufficient to prevent a serious overpressure challenge to the reactor vessel during this type of event. This is different from the assumption used in the level 1 sequence quantification that two SRVs are required.

7.1.5 Vapor Suppression Success Criteria

Vapor suppression during a LOCA is accomplished at Monticello by channeling steam from the drywell through eight large vent pipes into a vent header inside the torus air space; from this header the steam is piped into the suppression pool water through 96 downcomers, each about two feet in diameter. In the vent header, there are also eight 18" vacuum breakers which, when open, provide a direct pathway from the drywell into the wetwell airspace. A study was done to find the number of vacuum breakers which must stick open in order to fail vapor suppression during a LOCA blowdown, causing drywell failure on overpressure.

7.1.5.1 **One or two stuck-open vacuum breakers:**

Successful vapor suppression depends on both the number of open vacuum breakers and the LOCA break size. It was found that with one or two vacuum breakers stuck open, no break size was large enough to cause drywell failure during the blowdown, up to and including a double-ended guillotine break of the 28" recirculation pump suction line. This appears to be due to two different considerations. First, when the break is larger than about 0.35 ft^2 (8" diameter), there is enough driving force that even with two vacuum breakers open, much of the steam is still forced through the 96 downcomers and into the suppression pool, where it is condensed. The larger the break, the more condensed steam, so among breaks larger than 0.35 ft^2, a larger break actually results in a lower drywell pressure. Second, with the vacuum breakers open, the volume to be pressurized is the combined volume of the drywell and the wetwell airspace, rather than the drywell alone. This combined volume is roughly double the volume of the drywell, and break sizes less than about 1 ft^2 cannot pressurize this huge volume enough to fail the containment even with all vacuum breakers open (see below). It was therefore found that with two open vacuum breakers, the highest containment pressures occur for breaks of about 0.35 ft^2. The peak pressure with this break size and two stuck-open vacuum breakers is 99.4 psia, significantly less than the estimated containment failure pressure of 117.7 psia.

It is concluded that with two vacuum breakers open, vapor suppression is successful for all LOCAs.

7.1.5.2 **Eight stuck-open vacuum breakers:**

A further study was done to determine how large a LOCA must be to overpressurize the containment when all eight vacuum breakers are open. A break of 1.07 ft^2 (14" diameter) produced a peak drywell pressure of 109.5 psia; a 1.77 ft^2 break (18" diameter) reached 117.8 psia, sufficient to fail the containment.

It is concluded that open vacuum breakers do not lead to containment failure during LOCAs less than about 14" in diameter.

These results differ slightly from the success criteria assumed for class 3-D sequences in the PRA quantification. As noted in section 3.4.2.5, failure of more than one vacuum breaker conservatively was assumed to result in loss of vapor suppression, yet this did not contribute significantly to core damage.

Figure 7.1-1: Reactor Elevations from Sea Level.

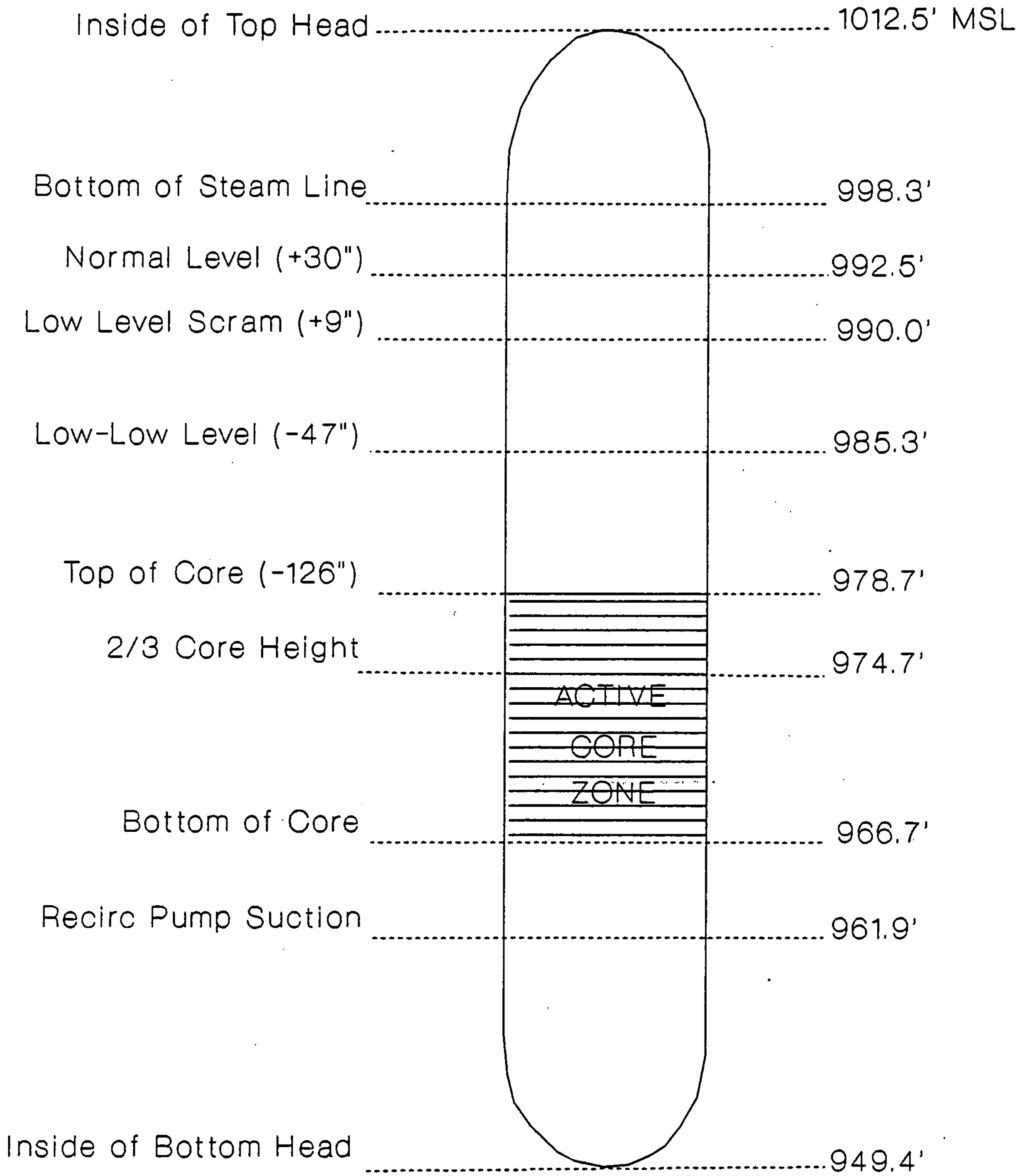
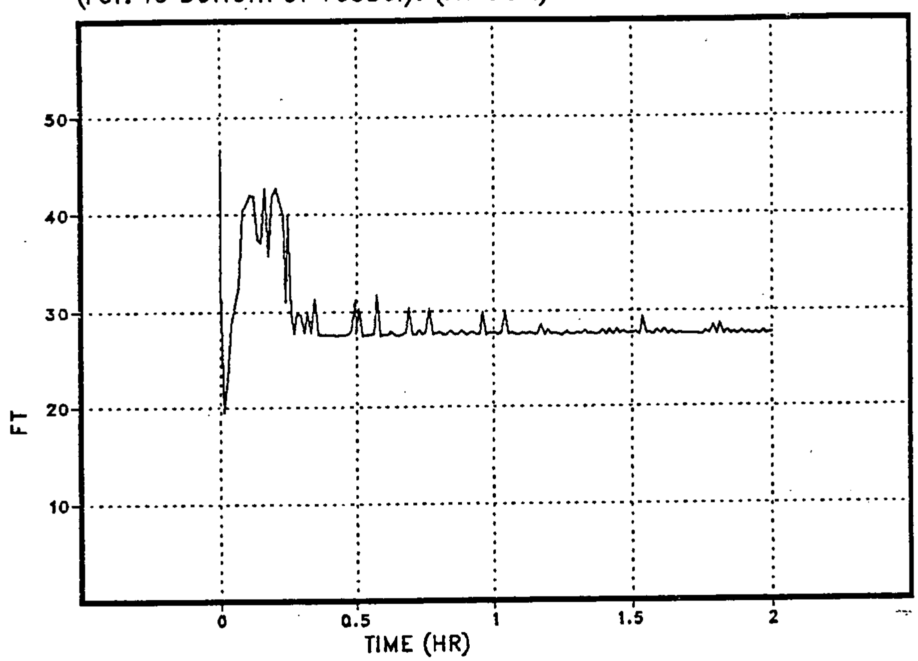


Figure 7.1-2

MAAP MNT LARGE LOCA W/ 1 LPCI PUMP ONLY; NO ADS MPB019/91

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

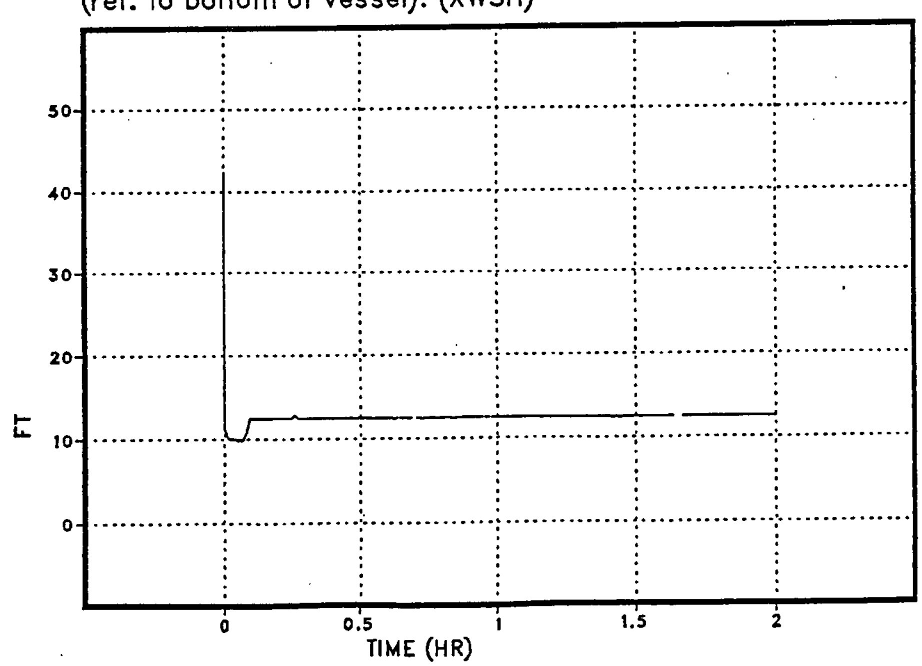


Figure 7.1-3

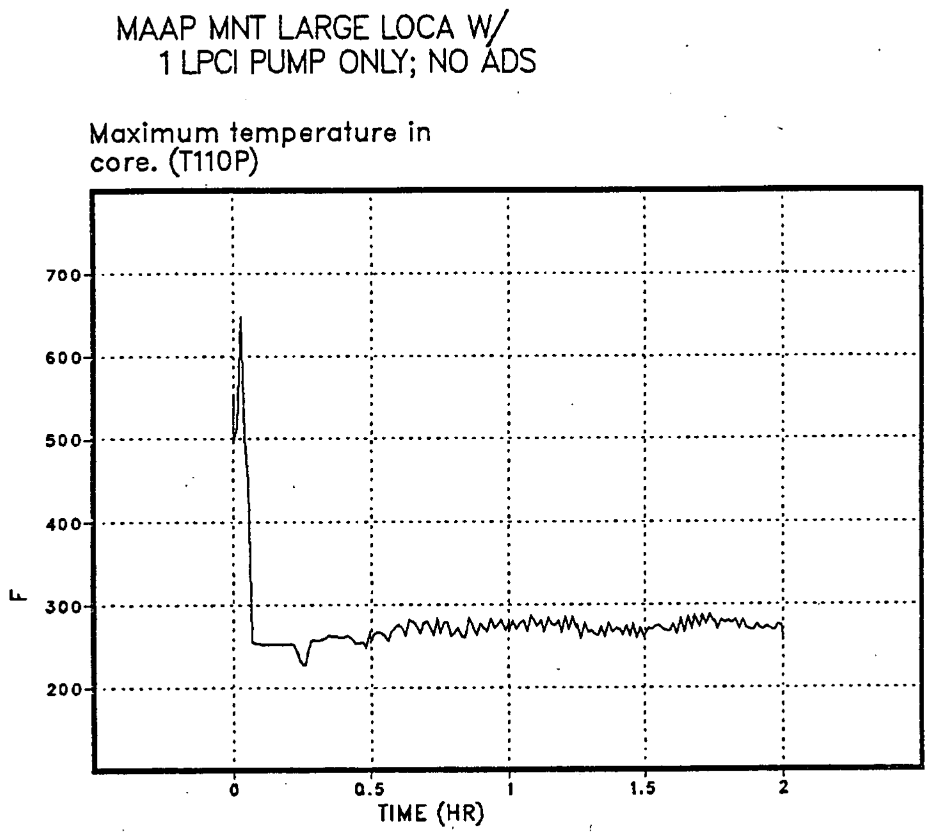
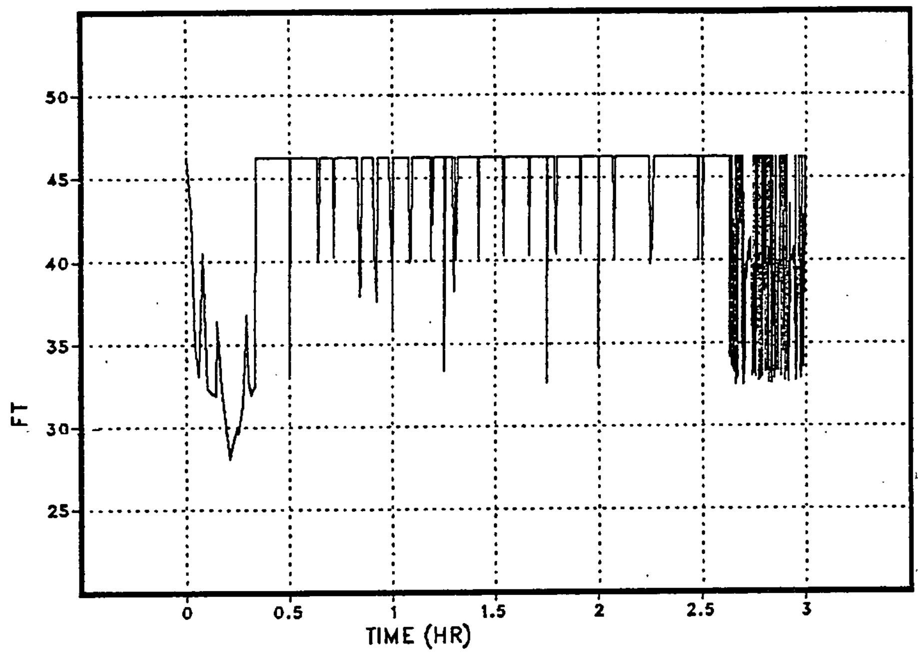

MAAP MNT LARGE LOCA W/
1 LPCI PUMP ONLY; NO ADS
Maximum temperature in
core. (T110P)
700
600
500
400
300
200
F
0
0.5
1
1.5
2
TIME (HR)

Figure 7.1-4

MNT 1 LPCI PUMP ONLY
ADS AT TAF

Height of water in core (ref. to vessel bottom). (XWCOR)



ECCS cycles on and off as relief valves open and reclose.

Height of water in shroud (ref. to vessel bottom). (XWSH)

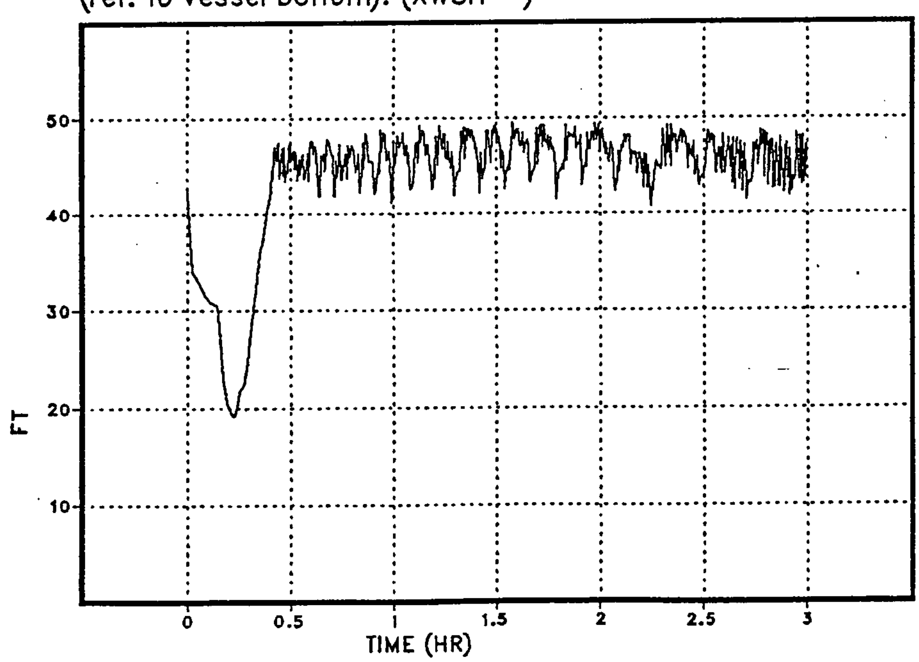


Figure 7.1-5

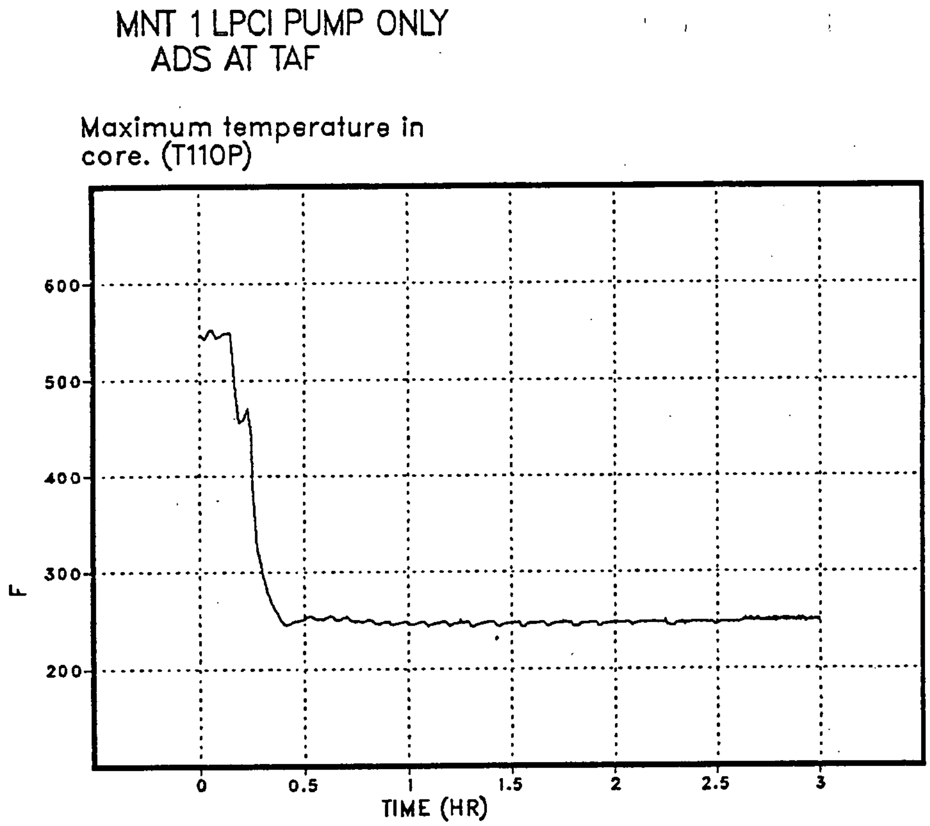
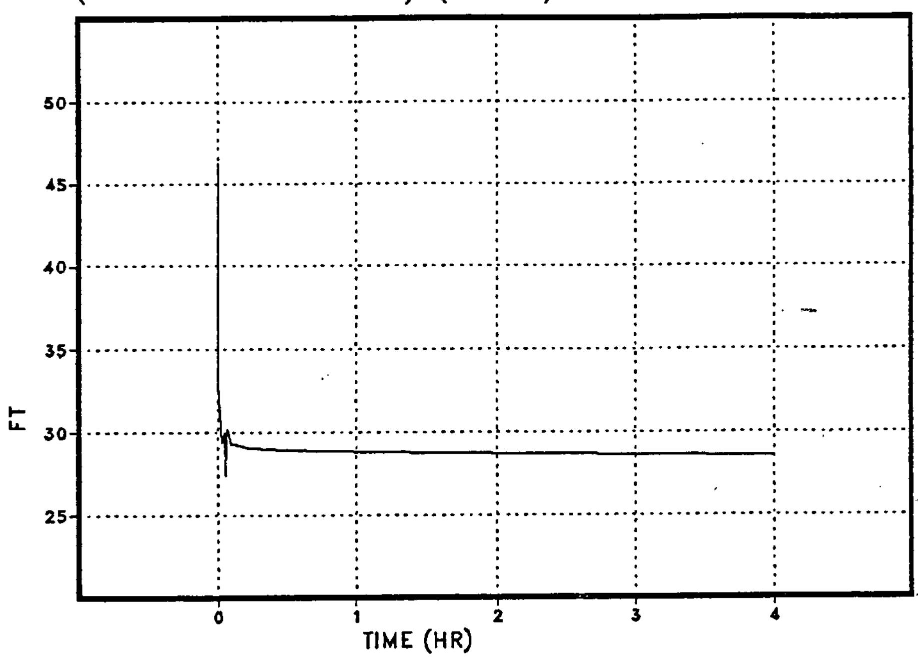

MNT 1 LPCI PUMP ONLY
ADS AT TAF
Maximum temperature in
core. (T110P)
600
500
400
300
200
F
0
0.5
1
1.5
2
2.5
3
TIME (HR)

Figure 7.1-6

MAAP MNT LARGE LOCA W/
1 LPCS PUMP ONLY; NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

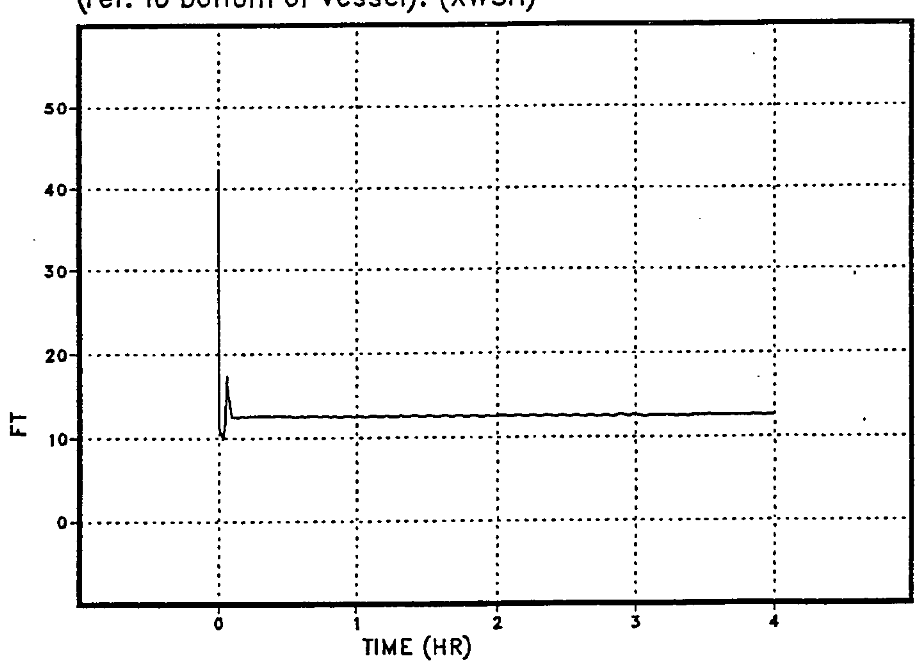


Figure 7.1-7

MAAP MNT LARGE LOCA W/
1 LPCS PUMP ONLY; NO ADS

Maximum temperature in
core. (T110P)

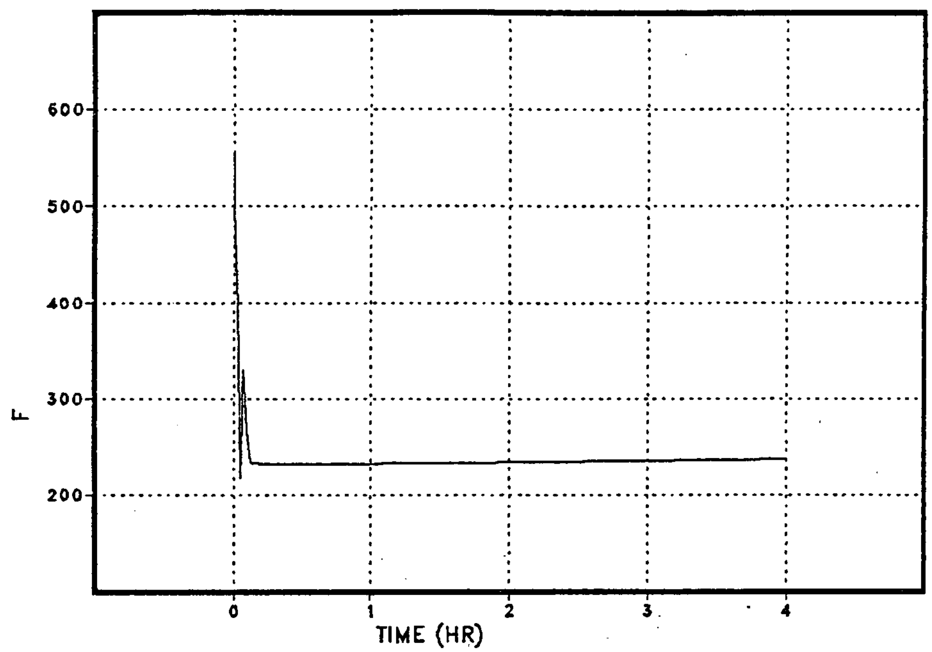
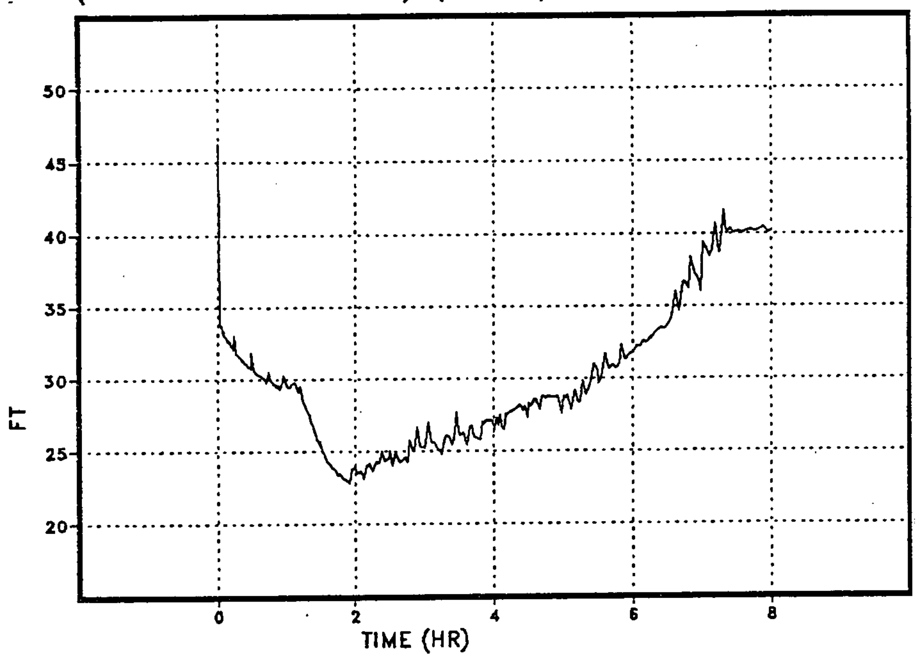


Figure 7.1-8

MNT 1 CRD PUMP ONLY
NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

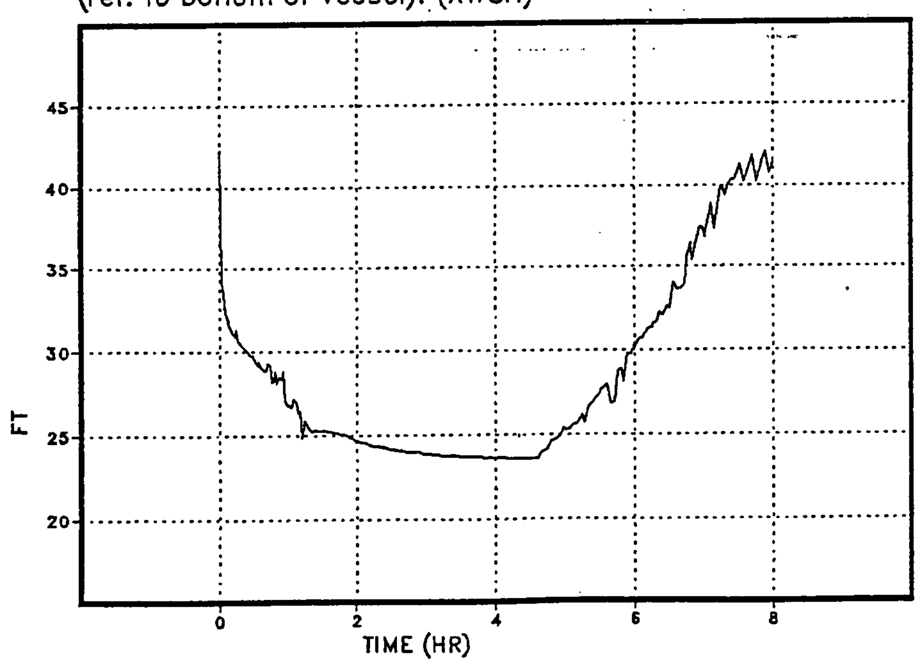
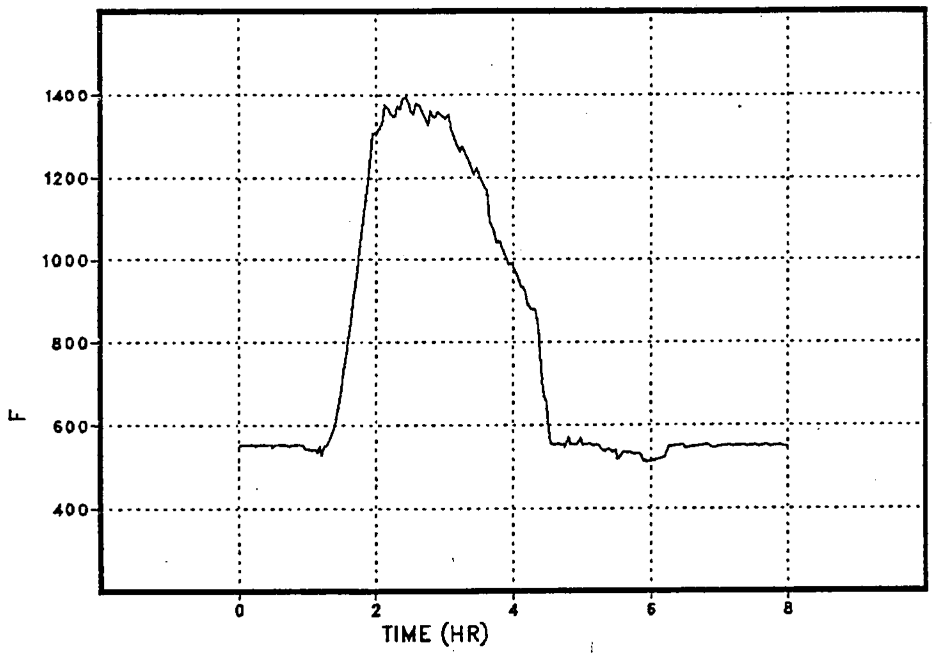


Figure 7.1-9

MNT 1 CRD PUMP ONLY
NO ADS

Maximum temperature in
core. (T110P)



Average core temperature.
(T15P)

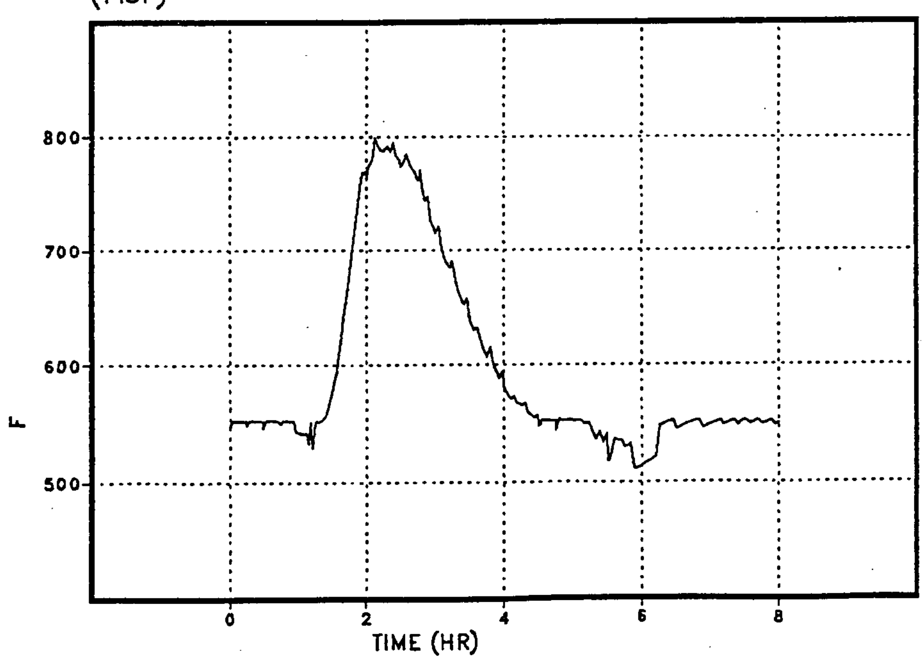
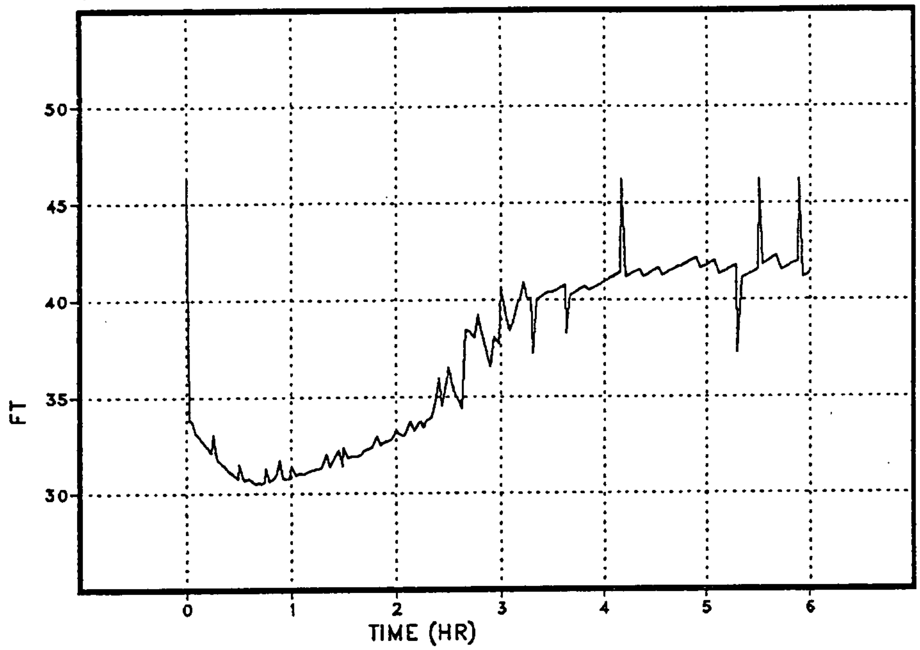


Figure 7.1-10

MNT 1 CRD PUMP ONLY; NO ADS
REALIGN @ 30 MIN TO MAXIMIZE FLOW

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

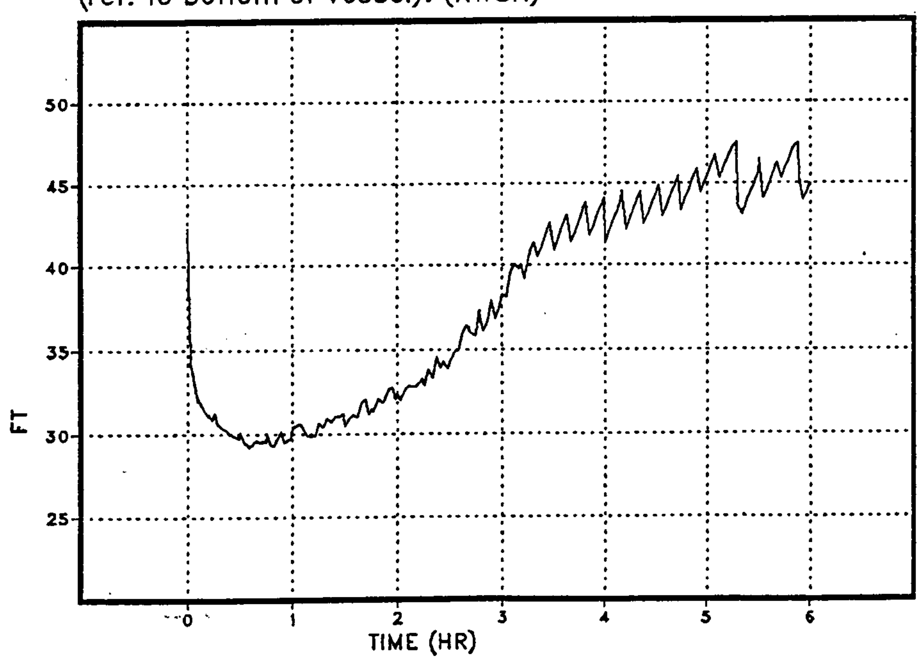
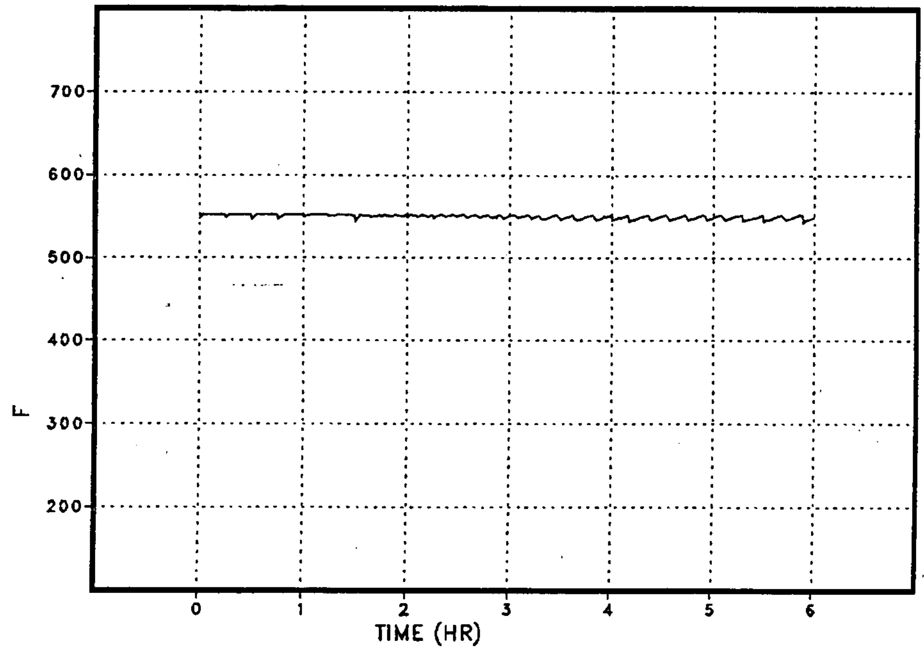


Figure 7.1-11

MNT 1 CRD PUMP ONLY; NO ADS
REALIGN @ 30 MIN TO MAXIMIZE FLOW

Maximum temperature in core. (T110P)



Average core temperature. (T15P)

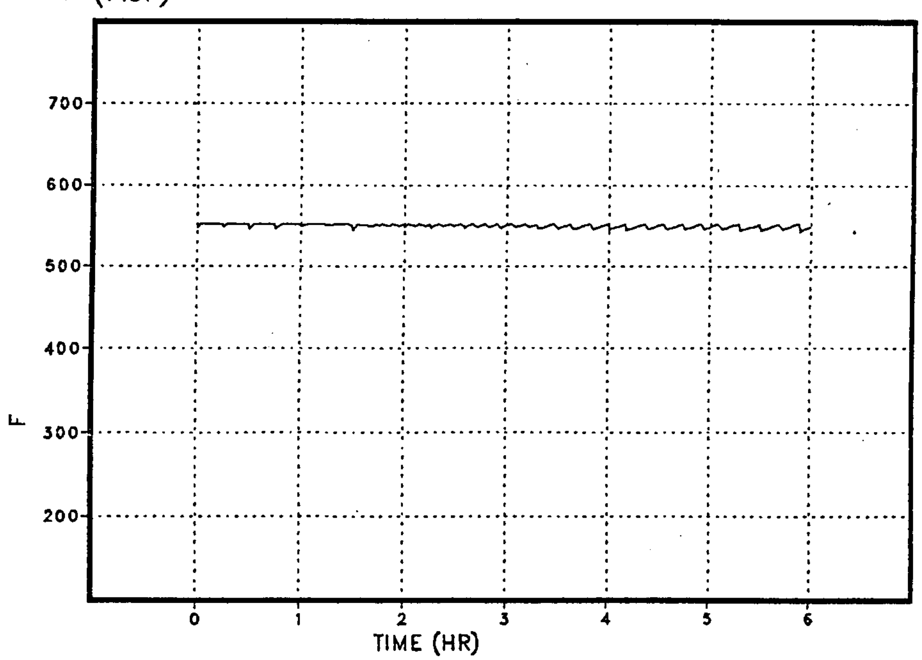
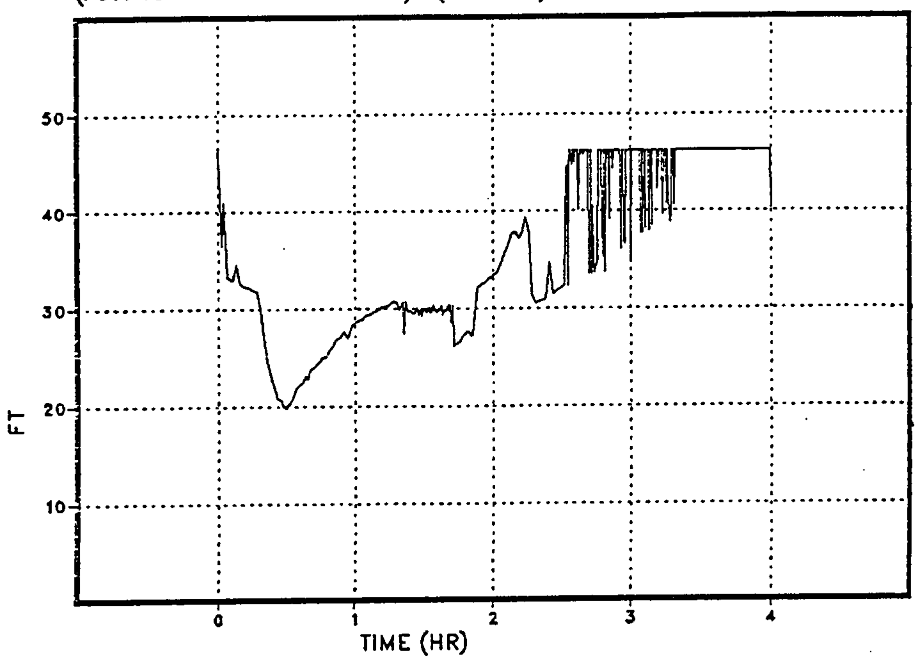


Figure 7.1-12

MNT 1 CRD PUMP; ADS @ TAF
START 2ND PUMP & REALIGN SYS @ 30 MIN

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

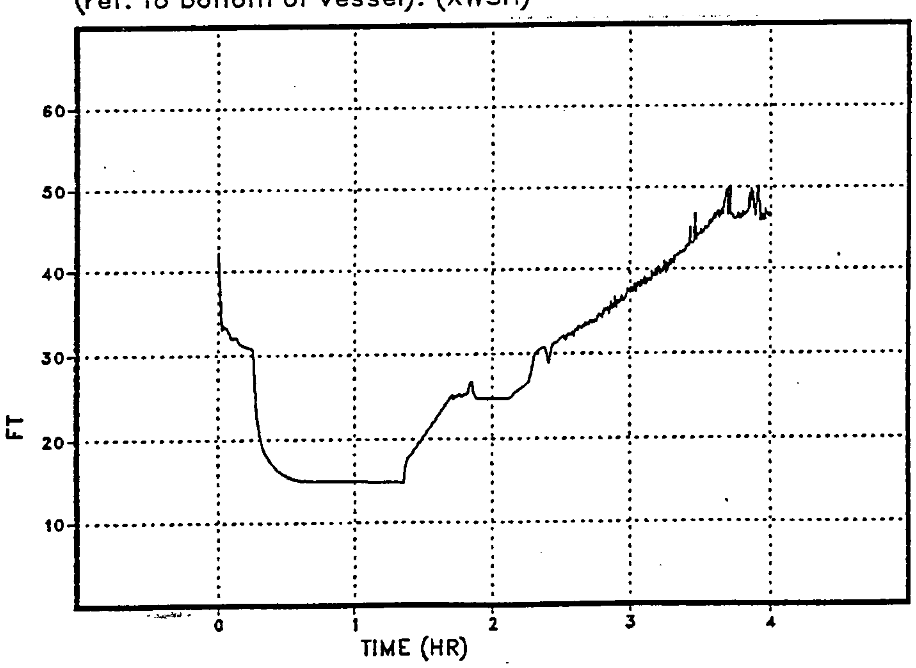
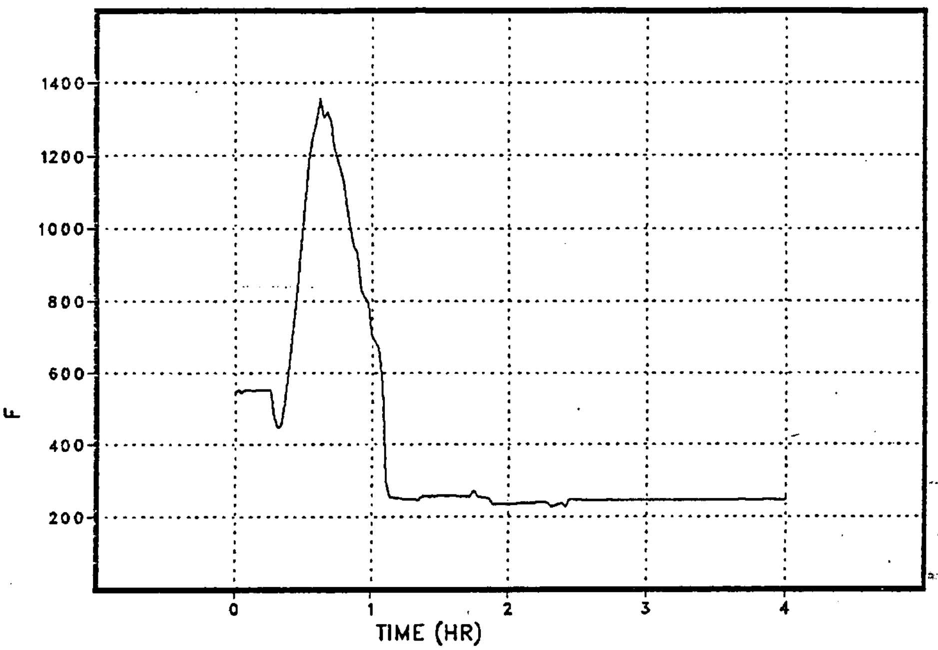


Figure 7.1-13

MNT 1 CRD PUMP; ADS @ TAF
START 2ND PUMP & REALIGN SYS @ 30 MIN

Maximum temperature in core. (T110P)



Average core temperature. (T15P)

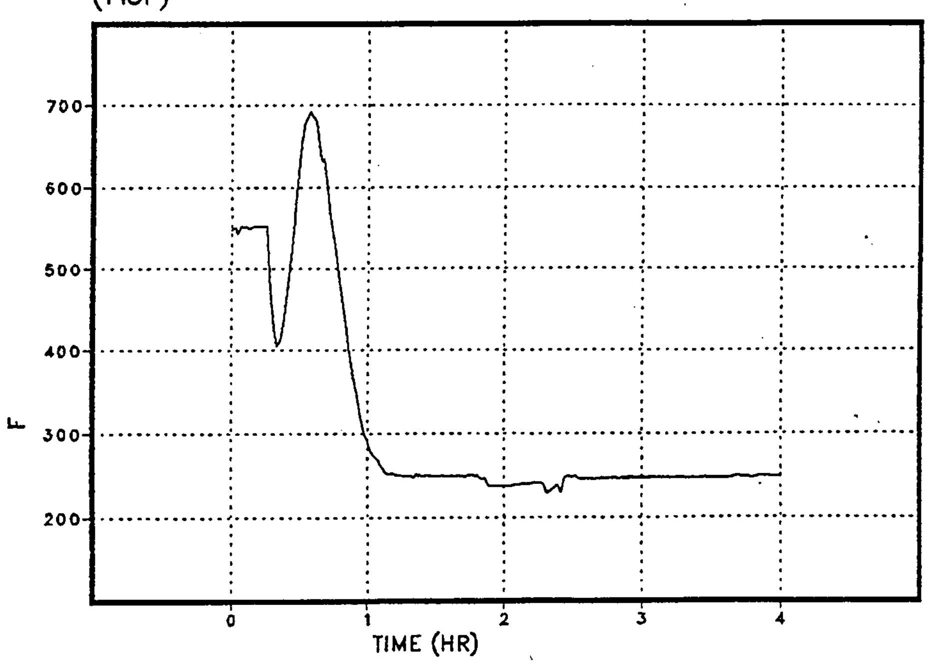
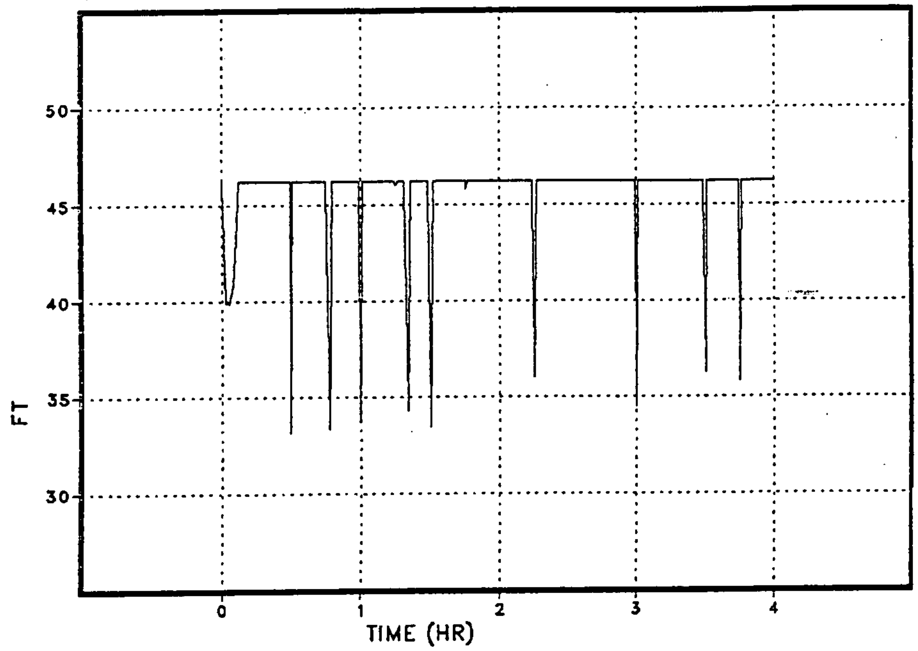


Figure 7.1-14

MNT 1 SORV W/
HPCI & 1 CRD PUMP ONLY; NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

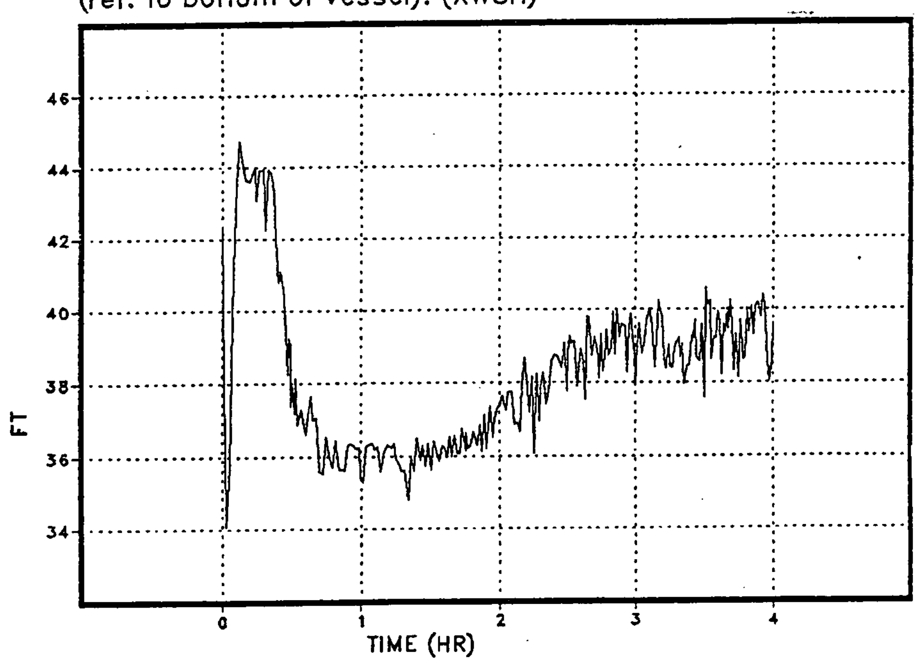
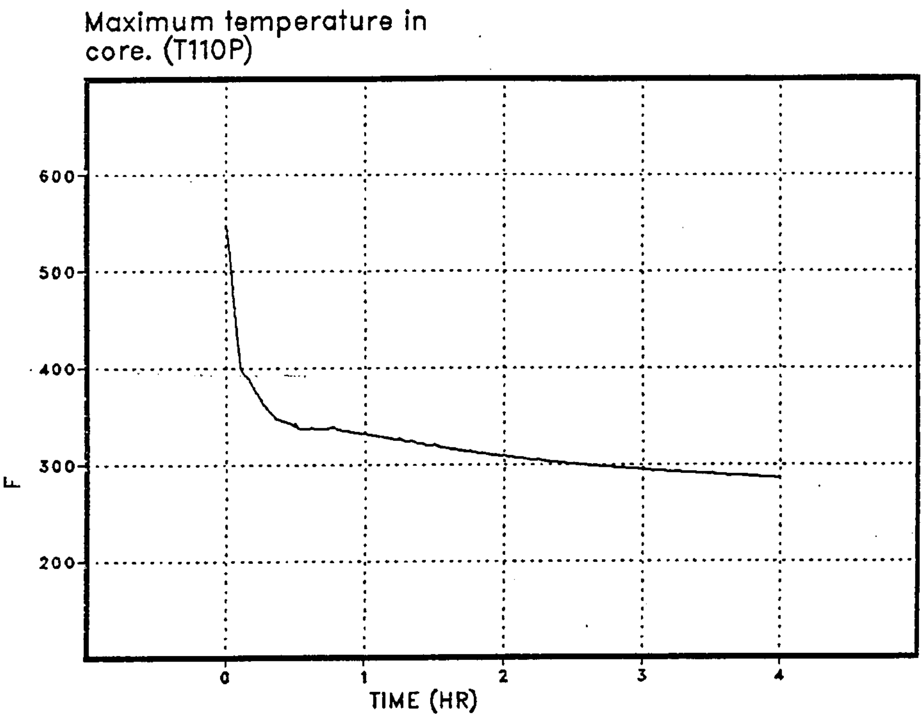
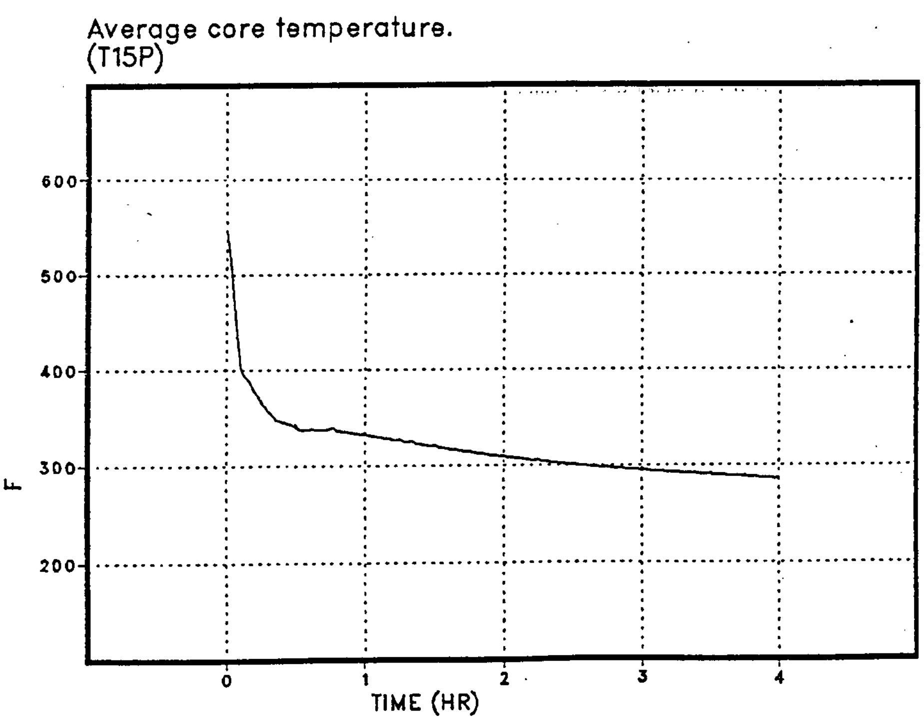


Figure 7.1-15

MNT 1 SORV W/
HPCI & 1 CRD PUMP ONLY; NO ADS





7.2 Accident Class Benchmarks

A series of MAAP cases was run to benchmark various accident classes by analyzing representative sequences for these classes. These benchmarks describe the progress of each accident for the Monticello plant, showing the relative timing and order of key events, plant conditions during the accident, the time available for operator actions, and the impact of those actions.

The benchmarks used to establish the timing of events for the level 1 analysis were run using revision 7.0 of MAAP 3.0B BWR. The behavior of drywell gas temperature in some accident classes was characterized using a later version of the code in order to eliminate certain conservatisms in the treatment of energy transfer between the drywell and the wetwell and in the heat transfer between the drywell airspace and the surface of a water pool on the drywell floor.

7.2.1 Class 1A: Loss of High-Pressure Injection, Fail to Depressurize

7.2.1.1 Loss of All Injection and Failure to Depressurize

This benchmark begins with a loss of feedwater and the failure of all high pressure injection systems; ADS is inhibited, so the reactor vessel remains pressurized. When the torus water temperature reaches 110°F, a single loop of torus cooling is initiated. Without high pressure injection, the core eventually melts through the reactor vessel bottom head and is ejected at high pressure into the pedestal. With the vessel depressurized, a single LPCI pump begins injecting; the LPCI flow pours through the hole in the vessel bottom and onto the core debris in the pedestal, providing debris cooling. [Figures 7.2-1 through 7.2-3.]

Timing of key events:

0 sec	feedwater trip, fail high pres injection, fail to depressurize
21 sec	indicated level = -48" → MSIV closure, recirc pump trip signal
9 min	indicated level = -126" = top of active fuel
26 min	actual water level = -126" = top of active fuel
56 min	pool temp = 110°F → start suppression pool cooling
1.1 hr	max core temp = 4040°F → fuel melting begins
2.0 hr	vessel failure; LPCI flow begins

Drywell pressure peaks at about 92 psia when the reactor vessel fails, then levels out near 86 psia for the remainder of the sequence.

7.2.1.2 Loss of Offsite Power with 1 LPCI Pump Available, Fail to Depressurize

This benchmark is used to characterize drywell gas temperature response in this accident class. It begins with a loss of offsite power in which no high pressure injection is available. ADS is not initiated, so the reactor vessel remains near operating pressure. One loop of suppression pool cooling is initiated when the pool temperature reaches 90°F. Without injection, the core melts through the vessel bottom and is ejected into the pedestal at high pressure. A single LPCI pump then begins to inject water through the failed vessel and onto the debris in the drywell. [Figures 7.2-4 and 7.2-5.]

Timing of key events:

0 sec	LOOP, recirc pump trip, fail high pressure injection, fail to depressurize
1 min	indicated level = -48" → MSIV closure
14 min	pool temp = 90°F → start suppression pool cooling
24 min	indicated level = -126" = top of active fuel
30 min	actual water level = -126" = top of active fuel
1.3 hr	max core temp = 4040°F → fuel melting begins
3.8 hr	vessel failure; LPCI flow begins

The drywell temperature spikes to 560°F when the vessel fails, drops back to 330°F, and rises slowly thereafter, reaching 430°F 40 hours into the accident.

7.2.2 Class 1D: TQUV - Loss of All Injection

7.2.2.1 Loss of All Injection; ADS at 2/3 Core Height

This benchmark begins with the failure of all injection systems. ADS is inhibited initially, so the reactor vessel remains near its normal operating pressure of 1000 psig while the existing water in the vessel boils off. When the indicated level reaches 2/3 core height, the operator opens a single ADS valve; once the pressure drops to 700 psig, the operator opens two more ADS valves; when the pressure reaches 100 psig, the operator begins cycling the ADS valves to maintain that pressure. The core eventually melts through the vessel bottom head and pours into the pedestal at low pressure, where it remains uncooled. [Figures 7.2-6 through 7.2-9.]

Timing of key events:

0.0 sec	feedwater and CRD pumps trip; no other injection available
21 sec	indicated level = -48" → MSIV closure, recirc pump trip signal
9 min	indicated level = -126" = top of active fuel
26 min	actual level = -126" = top of active fuel
27 min	indicated level = -174" = 2/3 core height → open 1 ADS valve
29 min	vessel pressure = 700 psig → open all 3 ADS valves
34 min	vessel pressure = 100 psig; maintain this pressure
1.8 hr	vessel failure

The drywell pressure spikes to 42 psia when the reactor vessel fails, drops back to 35 psia, and rises steadily thereafter, reaching 56 psia at 12 hours.

7.2.2.2 MSIV Closure, Loss of All Injection; ADS Initiated

This benchmark begins with closure of the MSIVs, followed by the failure of all injection systems. ADS is allowed to operate automatically to depressurize the vessel. With no injection, the core melts through the vessel bottom head and pours into the pedestal at low pressure. [Figures 7.2-10 through 7.2-12.]

Timing of key events:

0.0 sec	MSIV closure, failure of all injection
10 min	automatic ADS actuation
11 min	indicated level = -126" = top of active fuel
14 min	actual level = -126" = top of active fuel
1.6 hr	vessel failure
34 hr	drywell overtemperature failure (700°F)

The gas temperature in the drywell spikes to 250°F at vessel failure, drops back to 220°F, and then rises slowly throughout the accident, eventually reaching the drywell failure temperature of 700°F after 34 hours.

7.2.3 Class 1B: Station Blackout

7.2.3.1 SBO with HPCI and RCIC Available

This benchmark begins with a station blackout, disabling all injection except HPCI and RCIC, which operate on battery power. ADS is inhibited. The batteries are depleted after four hours, failing HPCI and RCIC and leaving the reactor with no injection. The core eventually melts through the vessel bottom and is ejected into the pedestal at high pressure. [Figures 7.2-13 through 7.2-15.]

Timing of key events:

0.0 sec	station blackout
31 sec	indicated level = -48" → MSIV closure and initiate HPCI & RCIC
4 min	indicated level = +48"
4 hr	battery depletion → HPCI & RCIC fail
5.2 hr	indicated level = -126" = top of active fuel
5.8 hr	actual level = -126" = top of active fuel
7.2 hr	max core temp = 4040°F → fuel melting begins
8.2 hr	vessel failure
35.4 hr	drywell gas temperature = 700°F → drywell failure

Drywell pressure shows a spike of 97 psia when the reactor vessel fails, considerably less than the 118 psia assumed to cause drywell failure. It drops back to 90 psia and rises slightly thereafter, reaching 104 psia at 35.4 hours.

7.2.3.2 SBO with No Injection; Manual ADS at 2/3 Core Height

In this benchmark, HPCI and RCIC are not available when the station blackout occurs, leaving no injection system available. ADS is inhibited until the indicated water level in the reactor reaches 2/3 core height. At that time the operator opens all 3 ADS valves, depressurizing the reactor vessel. The core eventually melts through the vessel bottom head and is ejected into the pedestal at low pressure. All of this occurs before the batteries fail at 4.0 hours. [Figures 7.2-16 through 7.2-21.]

Timing of key events:

0 sec	station blackout; no injection available
31 sec	indicated level = -48" ➡ MSIV closure
13 min	indicated level = -126" = top of active fuel
31 min	actual level = -126" = top of active fuel
32 min	indicated level = -174" = 2/3 core height ➡ ADS on
1.3 hr	max core temp = 4040°F ➡ fuel melting begins
1.9 hr	vessel failure
4.0 hr	battery failure

7.2.3.3 SBO with No Injection and No Depressurization

In this benchmark, HPCI and RCIC are not available when the station blackout occurs, leaving no injection system available. ADS is inhibited, and the core melts through the vessel bottom head and is ejected into the pedestal at high pressure. The drywell temperature and pressure continue to rise slowly due to decay heat; after 22 hours, the drywell gas temperature reaches 700°F, and the containment fails. [Figures 7.2-22 through 7.2-24.]

Timing of key events:

0 sec	station blackout; no injection available; ADS inhibited
1 min	indicated level = -48" ➡ MSIV closure
24 min	indicated level = -126" = top of active fuel
30 min	actual level = -126" = top of active fuel
1.2 hr	max core temp = 4040°F ➡ fuel melting begins
3.7 hr	vessel failure
4.0 hr	battery failure
22 hr	drywell overtemperature failure (700°F)

7.2.3.4 SBO with one SORV - HPCI Available Only

In this benchmark, a station blackout occurs at the same time that one safety/relief valve (SRV) sticks open. No injection is available except HPCI, which operates until the SORV lowers vessel pressure below the HPCI trip setpoint. The existing water in the vessel boils off, and the core eventually melts through the bottom head and is ejected at low pressure into the pedestal.

The drywell then heats up, reaching 700°F at 13.7 hours; for this case, the drywell was assumed not to fail at 700°F, but was allowed to remain intact until the failure pressure of 103 psig was reached at 17.5 hours. [Figures 7.2-25 through 7.2-30.]

Timing of key events:

0 sec	station blackout & 1 SORV
44 sec	indicated level = -48" → MSIV closure & HPCI actuation signal
6 min	indicated level = +48"
38 min	HPCI trips on low reactor vessel pressure
1.1 hr	indicated level = -126" = top of active fuel
2.0 hr	actual level = -126" = top of active fuel
4.0 hr	battery failure
4.2 hr	vessel failure
13.7 hr	drywell gas temperature = 700°F
17.5 hr	drywell overpressure failure (103 psig)

The drywell pressure is 93 psia at 13.7 hours.

7.2.3.5 SBO with one SORV - RCIC Available Only

This benchmark involves a station blackout coincident with a single safety/relief valve sticking open, with only the RCIC system available for injection. RCIC flow is insufficient to make up the inventory lost through the open SRV, and reactor level drops. RCIC fails when vessel pressure falls below its trip setpoint. The water in the vessel boils off, and the core melts through the vessel bottom head and is ejected at low pressure into the pedestal. The drywell then heats up, eventually failing due to high temperature. [Figures 7.2-31 through 7.2-36.]

Timing of key events:

0 sec	station blackout & 1 SORV
42 sec	indicated level = -48" ➡ MSIV closure & RCIC actuation signal
5 min	indicated level = -126" = top of active fuel
59 min	RCIC trip due to low reactor vessel pressure
1.3 hr	actual level = -126" = top of active fuel
3.5 hr	vessel failure
4.0 hr	battery depletion
26 hr	drywell gas temperature = 700°F ➡ drywell failure

Drywell pressure is 105 psia at the time of drywell failure.

7.2.4 Anticipated Transient Without SCRAM (ATWS)

7.2.4.1 ATWS: MSIV Closure; HPCI, RCIC, & 1 CRD Pump Only

This benchmark begins with the closure of all MSIVs; the scram function fails and the standby liquid control system is not actuated. Feedwater fails, but the single CRD pump which is already in operation continues to run. Within seconds, HPCI and RCIC are initiated and the recirculation pumps trip due to low level in the reactor; no other injection is available. The combined flow is sufficient to maintain indicated level about two feet below the top of active fuel, and core power drops to 28%. With the MSIVs closed, the steam from the reactor is released through the SRVs and into the suppression pool. When the pool becomes saturated, the containment begins to pressurize and eventually fails on high pressure, disabling all injection systems. The core melts through the vessel bottom and is ejected at high pressure into the failed containment. [Figures 7.2-37 through 7.2-40.]

Timing of key events:

0 sec	MSIV closure ATWS; feedwater fails, 1 CRD pump injects
25 sec	indicated level = -48" ➡ recirc pump trip; actuate HPCI & RCIC
1 min	indicated level = -126" = top of active fuel
46 min	drywell overpressure failure (118 psia) ➡ fail all injection
50 min	actual level = -126" = top of active fuel
2.3 hr	vessel failure

The time allotted in the level 1 analysis for the operator to initiate standby liquid control was based on suppression pool temperature. The system had to be initiated in time to achieve reactor shutdown before the suppression pool reached 260°F. This temperature conservatively accounts for uncertainties in vapor suppression in a saturated pool with high steam flow rates and SRV loads at high pool temperatures. In this case, the operator has fifteen minutes to initiate standby liquid control.

7.2.4.2 ATWS: MSIV Closure; Feedwater Initially Available; HPCI, RCIC, and 1 CRD Pump Available

This benchmark begins with the closure of all MSIVs; the scram function fails and the standby liquid control system is not actuated. Feedwater and one CRD pump remain in operation, maintaining normal level. Both recirculation pumps trip on high vessel pressure, lowering core power to 37%; because the MSIVs are closed, the steam from the core must be relieved to the suppression pool. At five minutes, HPCI is initiated by high pressure in the drywell. At 11 minutes, feedwater is assumed to fail due to low level in the condenser hotwell; HPCI and one CRD pump provide insufficient makeup for this core power, and the water level begins to drop. RCIC is initiated when the indicated level reaches -48", and this additional flow is enough to prevent any further drop. With the level maintained near -48", the core power is maintained at 25% until the drywell fails on overpressure, disabling all injection. Without injection, the core eventually melts through the vessel bottom and is ejected at high pressure into the failed containment. [Figures 7.2-41 through 7.2-44.]

Timing of key events:

0 sec	MSIV closure ATWS; feedwater & 1 CRD pump operating
15 sec	recirculation pump trip
5 min	HPCI actuation on high drywell pressure (+2 psig)
11 min	feedwater failure
14 min	indicated level = -48" → RCIC actuation signal
43 min	drywell overpressure failure (118 psia) → fail all injection
44 min	indicated level = -126" = top of active fuel
48 min	actual level = -126" = top of active fuel
2.1 hr	vessel failure

Standby liquid control must be initiated within fifteen minutes in this case in order to achieve reactor shutdown before the suppression pool temperature reaches 260°F. Note that the operation of feedwater in this case does not alter the time available for operator action.

7.2.4.3 ATWS: Turbine Trip with Bypass; Feedwater Available; Maintain Indicated Reactor Level at +48"

This benchmark begins with a turbine trip in which the scram function fails and the standby liquid control system is not actuated. Within seconds, both recirculation pumps trip on high vessel pressure, lowering core power to 39%. The turbine bypass valves, which are sized for 15% of rated flow, remain open; the rest of the steam is relieved through the SRVs to the suppression pool. Feedwater is available and is used to maintain indicated level at +48". When the suppression pool becomes saturated after 20 minutes, the containment begins to pressurize and eventually fails on high pressure. This releases steam into the reactor building, and all injection is assumed to fail as a result. Without injection, the water in the vessel quickly boils off, and the core melts through the bottom head and is ejected at high pressure into the failed containment. [Figures 7.2-45 through 7.2-48.]

Timing of key events:

0 sec	turbine trip ATWS
8 sec	recirculation pump trip; power drops to 39%
13 sec	indicated level = +48"
38 min	drywell overpressure failure (118 psia) → all injection fails
39 min	indicated level = -48" → MSIV closure
40 min	indicated level = -126" = top of active fuel
44 min	actual level = -126" = top of active fuel
2.1 hr	vessel failure

The time available for the operator to inject standby liquid control in this sequence is also fifteen minutes. For Monticello, the rate of containment heatup during an ATWS does not depend significantly on whether the condenser is available; this is because of the relatively small turbine bypass capacity, about 15% of rated steam flow.

7.2.4.4 ATWS: Turbine Trip with Bypass; Feedwater Available; Maintain Indicated Reactor Level at Top of Active Fuel

This benchmark is identical to the preceding case except that when the suppression pool temperature reaches 110°F, the operator throttles feedwater flow to maintain indicated level at the top of active fuel. The event begins with a turbine trip in which the scram function fails and the standby liquid control system is not actuated. Both recirculation pumps trip on high vessel pressure; the turbine bypass valves, which are sized for 15% of rated flow, remain open; the excess steam is relieved through the SRVs to the suppression pool. After three minutes the torus water reaches 110°F and the operator throttles feed flow to maintain indicated level at the top of active fuel, thereby reducing core power to 26%. When the suppression pool becomes saturated, the containment begins to pressurize and eventually fails on high pressure. This releases steam into the reactor building, and all injection is assumed to fail as a result. Without injection, the core melts through the bottom head and is ejected at high pressure into the failed containment. [Figures 7.2-49 through 7.2-52.]

Timing of key events:

0 sec	turbine trip ATWS
13 sec	recirculation pump trip
3 min	pool temperature = 110°F → operator throttles feedwater
5 min	indicated level = -126" = top of active fuel; power = 26%
1.8 hr	drywell overpressure failure (118 psia)

By lowering the reactor water level, and therefore reactor power, in this case as compared to the case in section 7.2.4.3, the drywell failure is delayed from 38 minutes to 1.8 hours, a gain of 1.2 hours. The time of vessel failure was not determined for this case, but would be expected to occur around 3.5 hours, assuming that the elapsed time from drywell failure to vessel failure is approximately the same for these two cases.

Significant time can be made available to initiate alternate boron injection by lowering the reactor water to the top of active fuel, thereby lowering core power. This action was credited for sequences in which the main condenser was available but standby liquid control failed due to mechanical or electrical causes.

7.2.5 TW - Loss of Decay Heat Removal

7.2.5.1 Loss of Decay Heat Removal - High and Low Pressure Injection from Suppression Pool

This benchmark represents a loss of all containment heat removal in which both high-pressure (RCIC) and low-pressure (LPCI) injection systems are available, but only to inject water from the suppression pool; no source of water from outside the containment is used. [Figures 7.2-53 through 7.2-62.]

This case begins with the closure of all MSIVs and failure of feedwater and CRD injection. RCIC is initiated by low level in the reactor, and the SRVs relieve steam to the suppression pool, maintaining reactor pressure near normal operating pressure; ADS is inhibited. When the pool temperature reaches 142°F, the operator begins using one SRV to gradually depressurize the reactor vessel per procedure, to stay within the suppression pool's heat capacity temperature limit curve. When reactor pressure becomes low enough, the LPCI system also can inject. As the pool temperature reaches 194°F, the reactor is brought to 70 psig and maintained at that pressure.

RCIC becomes inoperable when the pool temperature reaches 200°F; after that time, makeup flow is supplied to the vessel by the LPCI system. The pressure and temperature in the drywell and wetwell continue to rise. After many hours, the drywell pressure reaches 70 psig; pneumatic control of the SRVs is lost, and they reclose. The reactor vessel repressurizes and LPCI can no longer inject to the vessel. Without makeup flow, the water level in the reactor begins to boil down, and it now becomes a race to see whether the core will melt through the vessel before the rising drywell pressure causes the containment to fail.

The drywell reaches its failure pressure of 103 psig when the actual reactor level is still one foot above the top of active fuel. The drywell failure depressurizes the containment and, for this analysis, all injection systems are assumed to fail upon drywell failure. Pneumatic control of the SRVs is regained as the drywell pressure drops, and the reactor vessel again depressurizes to 70 psig. Considerable inventory is lost through the SRVs during blowdown, and the core becomes uncovered. By this time nearly two days have elapsed, and decay heat is low enough that 5 hours pass before the core finally melts through the vessel bottom and is ejected at low pressure into the failed containment.

Timing of key events:

0 sec	MSIV closure; feedwater and CRD pumps fail
36 sec	indicated level = -48" → recirculation pump trip & RCIC actuation signal
4.6 hr	suppression pool temp = 142°F → begin vessel depressurization
7.1 hr	suppression pool temp = 194°F; vessel pressure = 70 psig
7.6 hr	suppression pool temp = 200°F → RCIC fails; LPCI injects
1.3 days	drywell pressure = 70 psig → SRVs close
1.4 days	LPCI fails on high vessel pressure
1.8 days	drywell overpressure failure (103 psig) → all injection fails & SRVs reopen; core becomes uncovered
1.9 days	max core temp = 4040°F → fuel melting begins
2.0 days	vessel failure

The drywell gas temperature reaches a peak of 345°F at the time of drywell failure.

7.2.5.2 Loss of Containment Heat Removal - High and Low Pressure Injection from CST

This benchmark represents a loss of all containment heat removal in which feedwater remains available to inject water from the condensate storage tanks (CSTs). No other injection source is used. [Figures 7.2-63 through 7.2-72.]

This case begins with the closure of all MSIVs and failure of CRD injection; feedwater continues to operate, and the recirculation pumps are tripped. Steam from the vessel is released through the SRVs to the suppression pool to maintain reactor pressure near normal operating pressure; ADS is inhibited. When the suppression pool water temperature reaches 142°F, the operator begins using one SRV to gradually depressurize the reactor vessel in order to stay within the suppression pool's heat capacity temperature limit curve. As the pool temperature reaches 194°F, the reactor is brought to 70 psig and maintained at that pressure.

The containment pressure and temperature continue to rise due to decay heat, and eventually the drywell pressure reaches 70 psig. This disables pneumatic control of the SRVs; they reclose, and the reactor vessel repressurizes to the SRV mechanical setpoint pressure of 1120 psig. Feedwater and CRD continue to provide makeup flow to the vessel. The drywell fails when it reaches 103 psig, nearly three days after the beginning of the accident. MAAP was not used to analyze this sequence beyond drywell failure; injection through systems external to the reactor building would be expected to maintain adequate cooling even after containment failure.

Timing of key events:

0 sec	MSIV closure; feedwater continues to operate
36 sec	recirculation pumps trip
4.7 hr	suppression pool temp = 142°F ➡ begin vessel depressurization
8.7 hr	suppression pool temp = 194°F ➡ vessel pressure = 70 psig
2.0 days	drywell pressure = 70 psig ➡ SRVs close
2.2 days	vessel pressure = SRV mechanical setpoint (1120 psig)
2.7 days	drywell overpressure failure (103 psig)

The drywell gas temperature reaches a peak of 360°F when the drywell fails.

7.2.6 Class 3: Loss of Coolant Accidents (LOCA)

7.2.6.1 SBLOCA, 1 LPCI Pump Only, Fail to Depressurize

In this small break LOCA benchmark, a break of one square inch occurs in the suction line to one of the recirculation pumps, and the containment isolates. All high pressure injection systems fail, but the operator inhibits ADS and does not depressurize the reactor vessel. The vessel remains at operating pressure without injection until the core melts through the vessel bottom and is ejected at high pressure into the pedestal. A single LPCI pump then begins injecting water from the suppression pool into the failed vessel, through the opening in the vessel bottom, and onto the core debris in the pedestal. As the water overflows back into the suppression pool, LPCI continues to recirculate it through the vessel to keep the debris cooled. The containment pressure and temperature rise due to the decay heat until in time the drywell fails on high pressure. [Figures 7.2-73 through 7.2-78.]

Timing of key events:

0 sec	SBLOCA (1 in^2); ADS inhibited, high pressure injection fails
21 sec	indicated level = -48" → MSIV closure, recirc pump trip signal
5 min	indicated level = -126" = top of active fuel
14 min	actual level = -126" = top of active fuel
47 min	max core temp = 4040°F = fuel melting begins
1.3 hr	vessel failure; LPCI flow begins
22 hr	drywell overpressure failure (118 psia)

The drywell gas temperature is 600°F at the time of containment failure.

7.2.6.2 SBLOCA, No Injection; ADS at 2/3 Core Height

In this small break LOCA benchmark, a break of one square inch occurs in the suction line to one of the recirculation pumps, causing the containment to isolate; all injection systems fail. The operator inhibits ADS until the indicated water level drops to 2/3 core height, then opens all three ADS valves and depressurizes the vessel. The loss of inventory through the ADS valves causes the core to become uncovered; the core melts through the bottom of the vessel and is ejected at low pressure into the pedestal. There is still no injection available to cool the debris, and the containment temperature and pressure rise until the drywell fails on high temperature. [Figures 7.2-79 through 7.2-85.]

Timing of key events:

0 sec	SBLOCA (1 in^2); ADS inhibited and all injection fails
21 sec	indicated level = -48" → MSIV closure, recirc pump trip signal
5 min	indicated level = -126" = top of active fuel
11 min	indicated level = -174" = top of active fuel → actuate ADS
12 min	actual level = -126" = top of active fuel
50 min	max core temp = 4040°F = fuel melting begins
1.3 hr	vessel failure
8.5 hr	drywell overtemperature failure (700°F)

The drywell pressure is 63 psia at the time of containment failure.

7.2.6.3 Medium Break LOCA, No Injection, No Depressurization

In this LOCA benchmark, a break of 9.6 in^2 occurs in the suction line to one of the recirculation pumps, causing the containment to isolate. All injection systems fail. ADS is inhibited, but the reactor vessel depressurizes through the break. The water in the vessel boils off quickly, and the uncovered core melts through the bottom of the vessel and pours at low pressure into the pedestal, where it remains uncooled. The temperature and pressure in the containment continue to rise, and in time the drywell fails on high temperature. [Figures 7.2-86 through 7.2-92.]

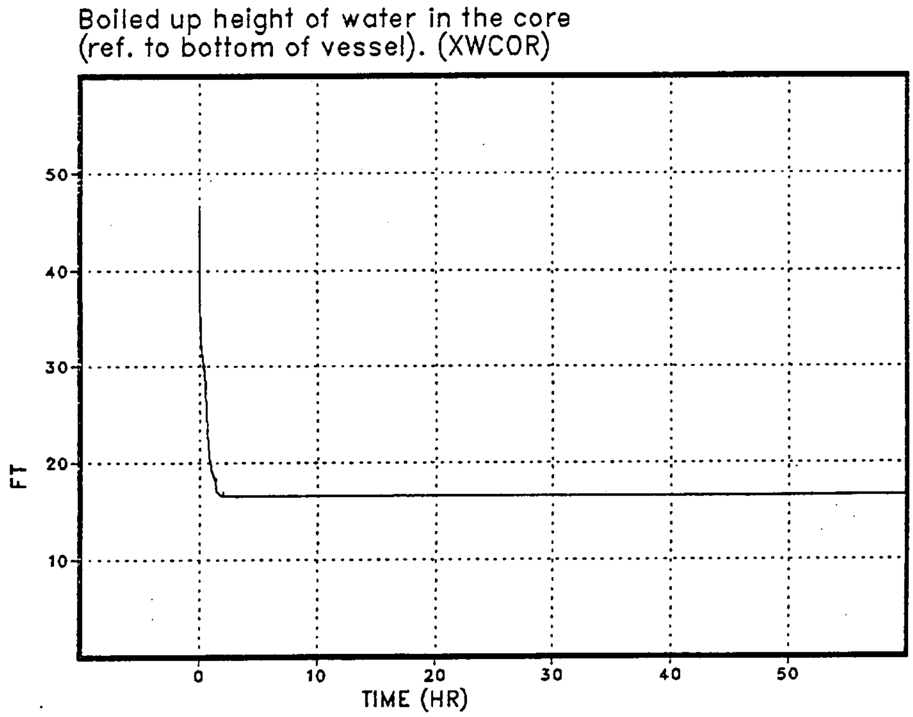
Timing of key events:

0 sec	medium LOCA (9.6 in^2); ADS inhibited and all injection fails
16 sec	indicated level = -48" → MSIV closure, recirc pump trip signal
2 min	indicated level = -126" = top of active fuel
5 min	actual level = -126" = top of active fuel
28 min	max core temp = 4040°F = fuel melting begins
35 min	vessel failure
5.5 hr	drywell overtemperature failure (700°F)

The drywell pressure is 63 psia at the time of containment failure.

Figure 7.2-1

MNT NO INJ EXCEPT 1 LPCI PUMP
1 RHRHX @ TORUS = 110F; NO ADS



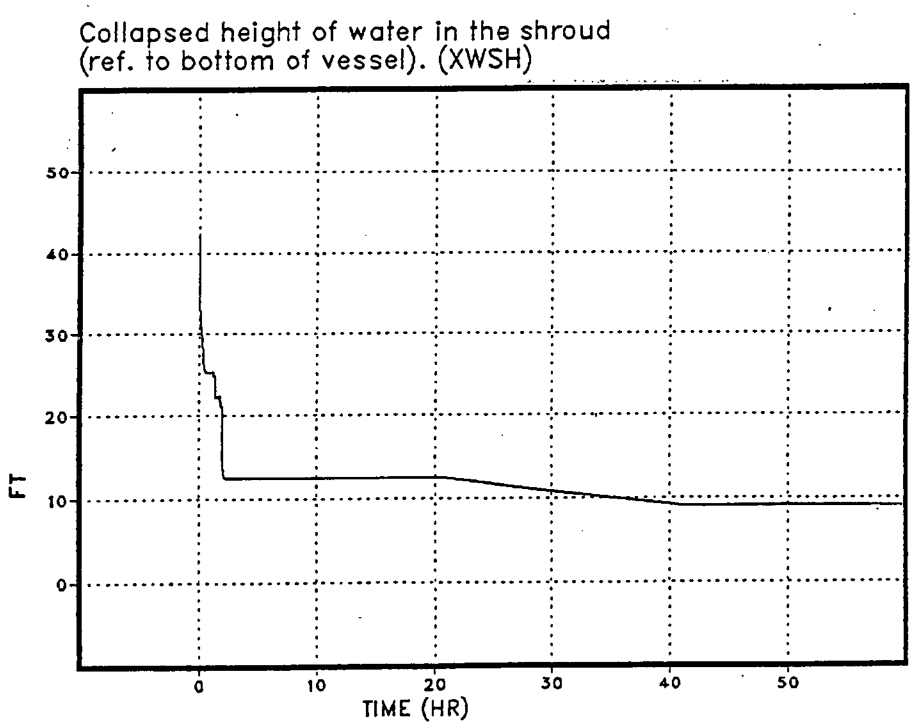
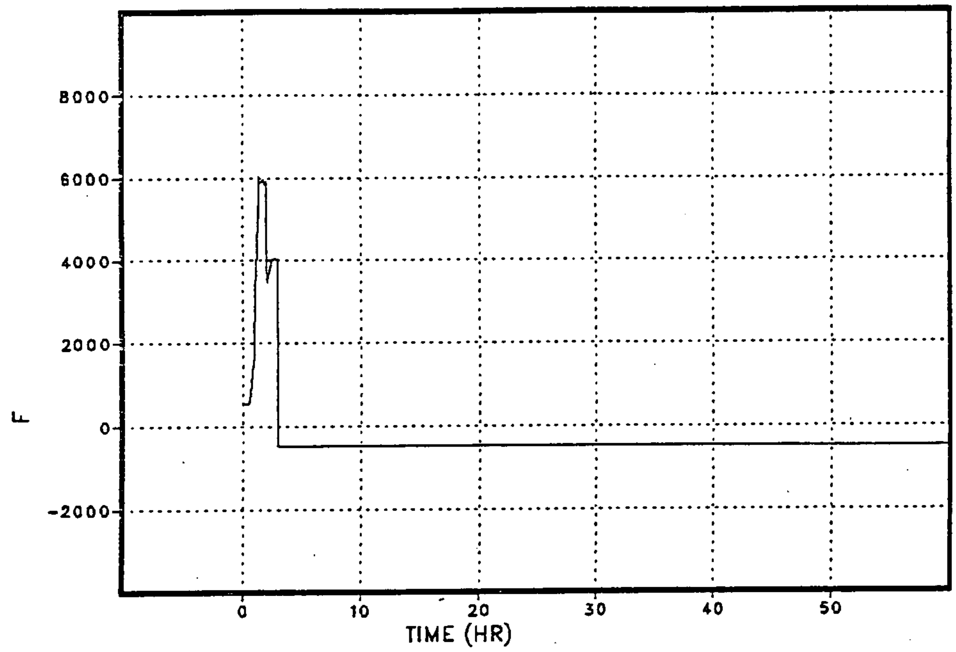


Figure 7.2-2

MNT NO INJ EXCEPT 1 LPCI PUMP
1 RHRHX @ TORUS = 110F; NO ADS

Maximum temperature in core. (T110P)



Pressure in the drywell. (PDW)

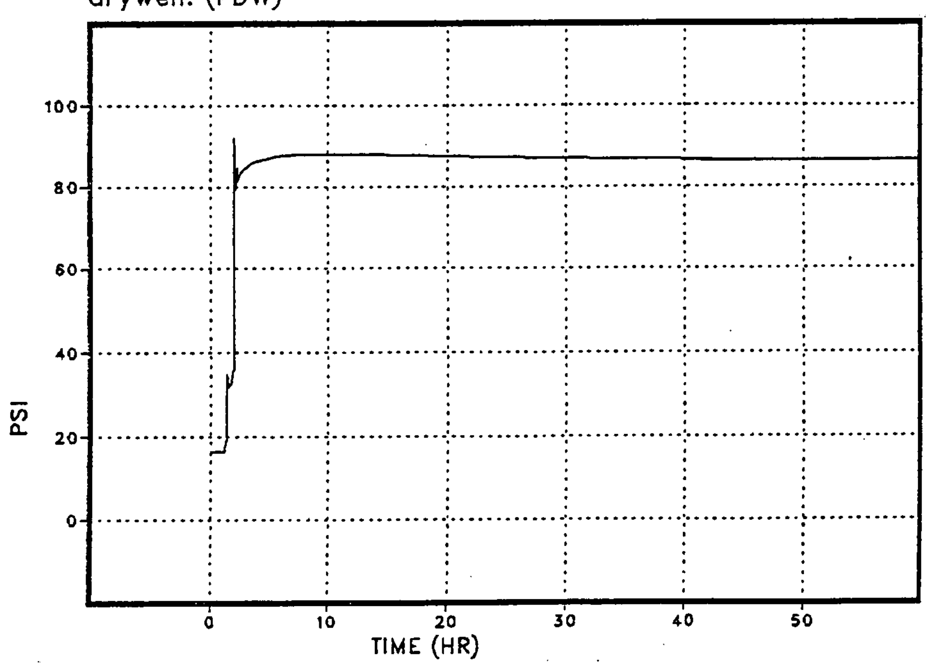
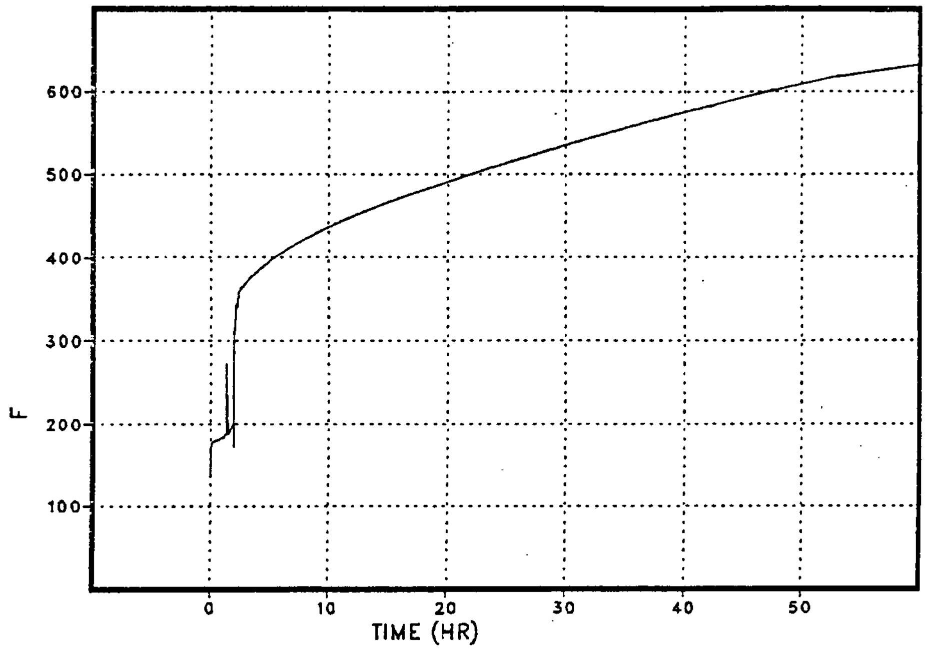


Figure 7.2-3

MNT NO INJ EXCEPT 1 LPCI PUMP
1 RHRHX @ TORUS = 110F; NO ADS

Temperature of gas
in the drywell. (TGDW)



Torus water temperature.
(TWSP)

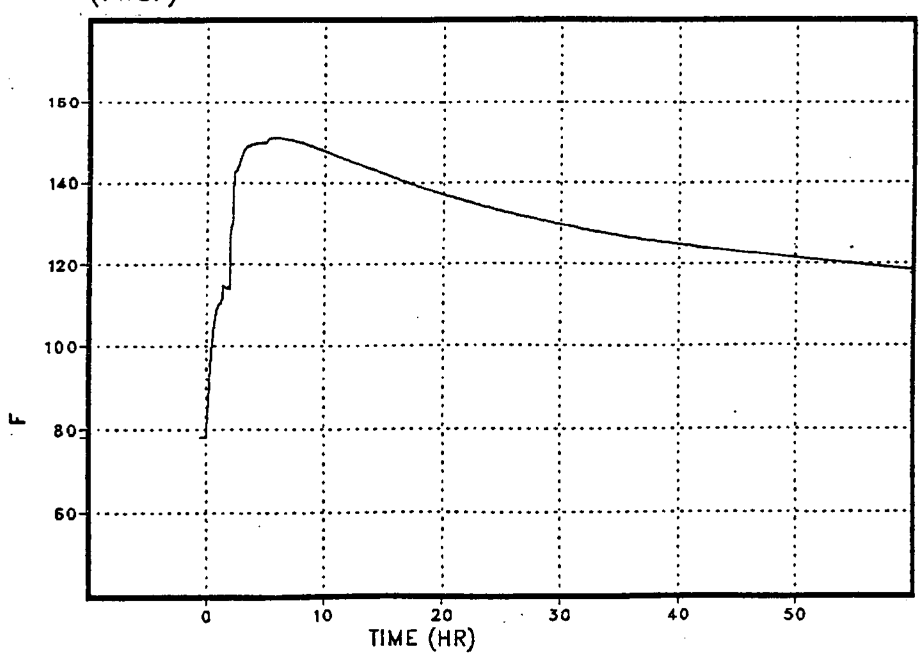
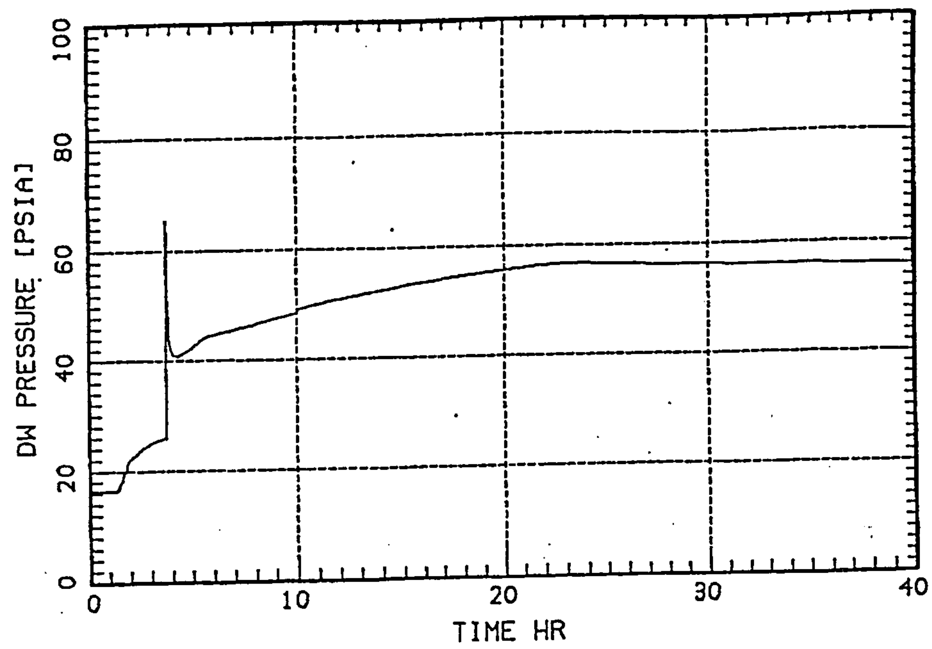


Figure 7.2-4

Loss of Offsite Power with 1 LPCI Pump Available
Failure to Depressurize



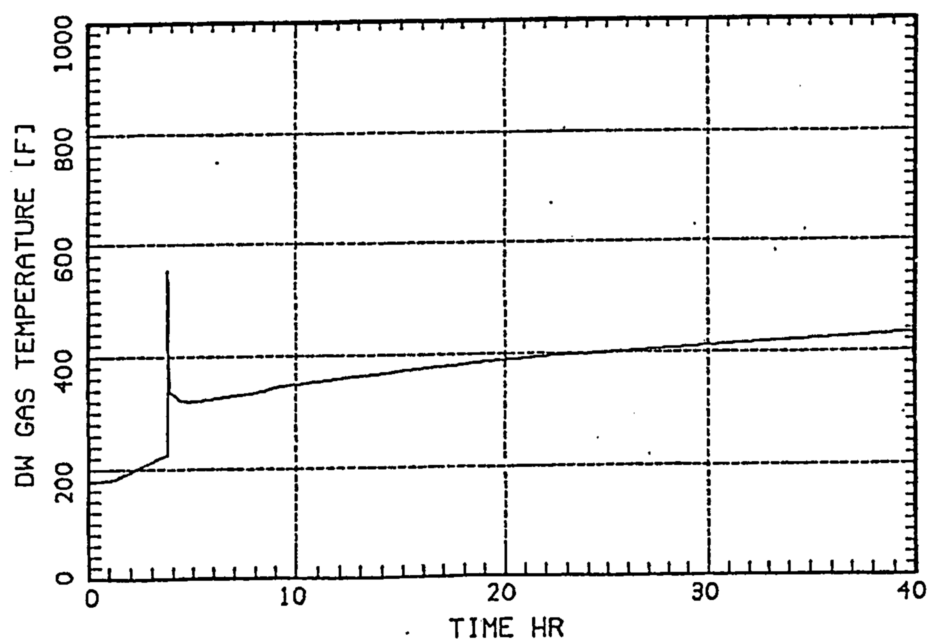


Figure 7.2-5

Loss of Offsite Power with 1 LPCI Pump Available
Failure to Depressurize

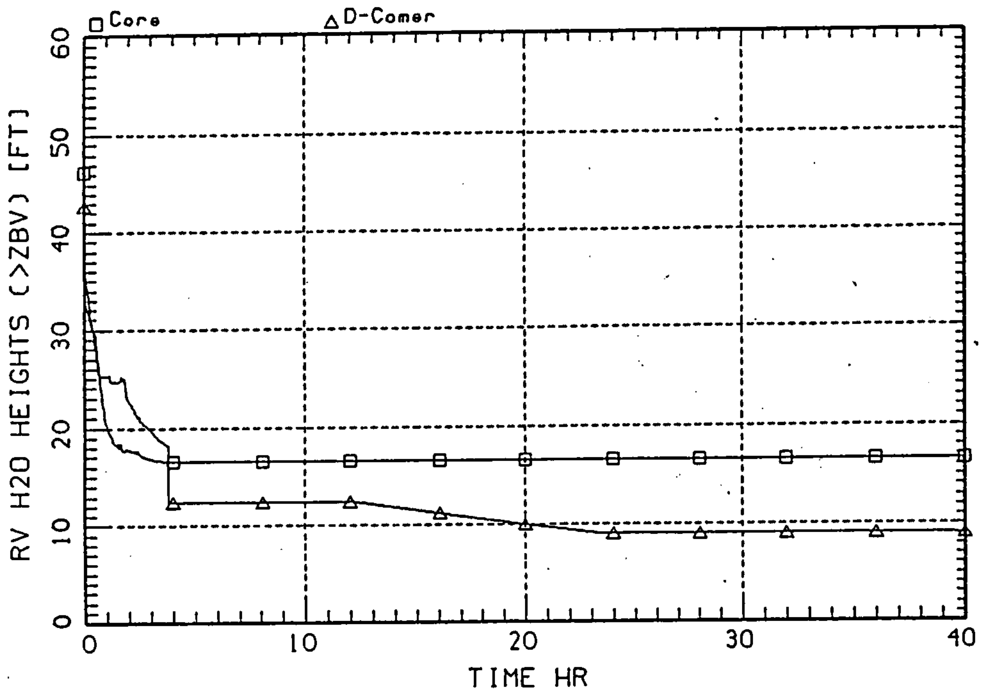
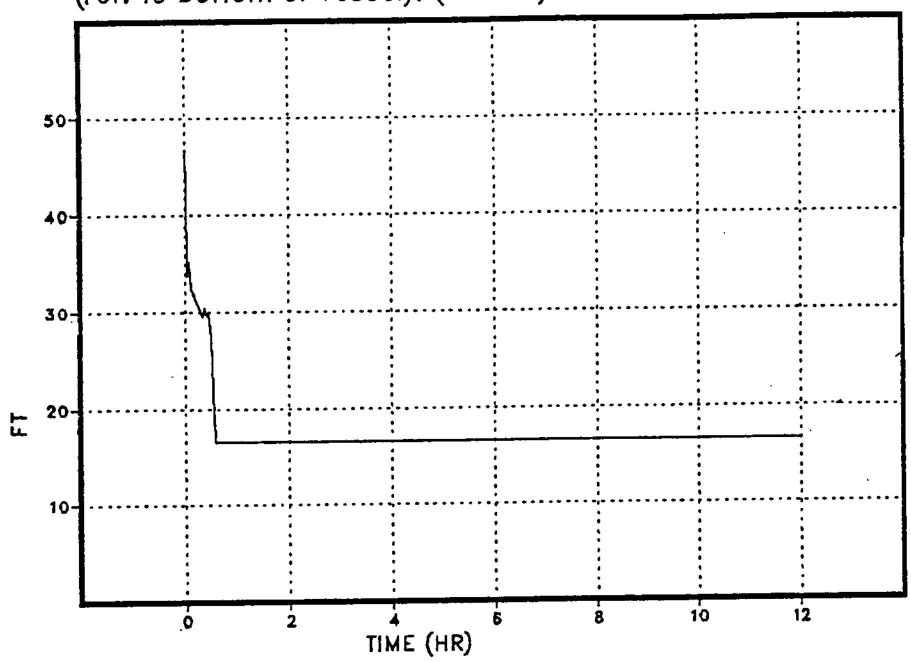


Figure 7.2-6

MNT LOFW NO INJ, 1 ADS
@ 2/3 CORE 3 ADS @ 700#

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

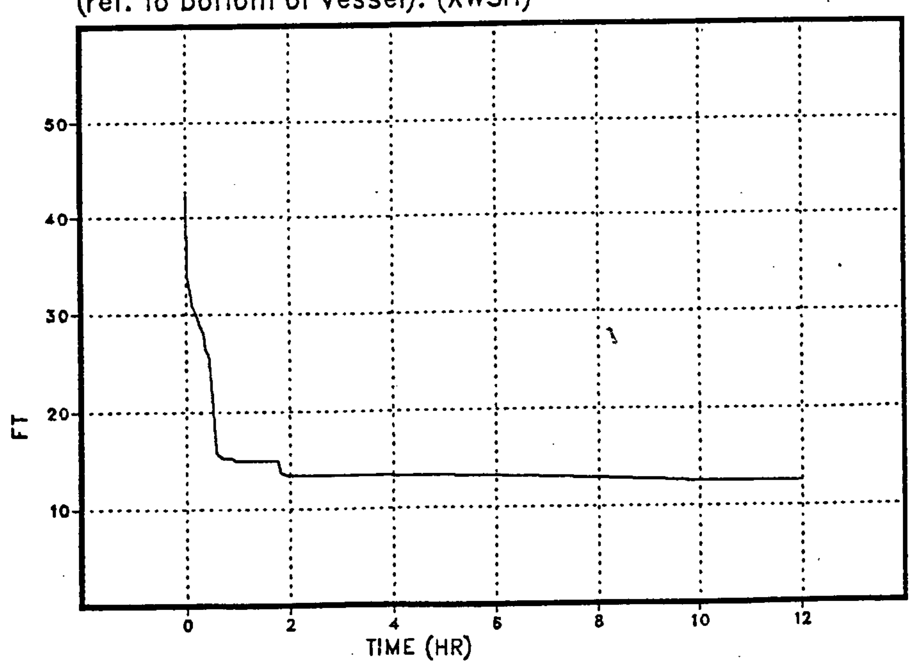
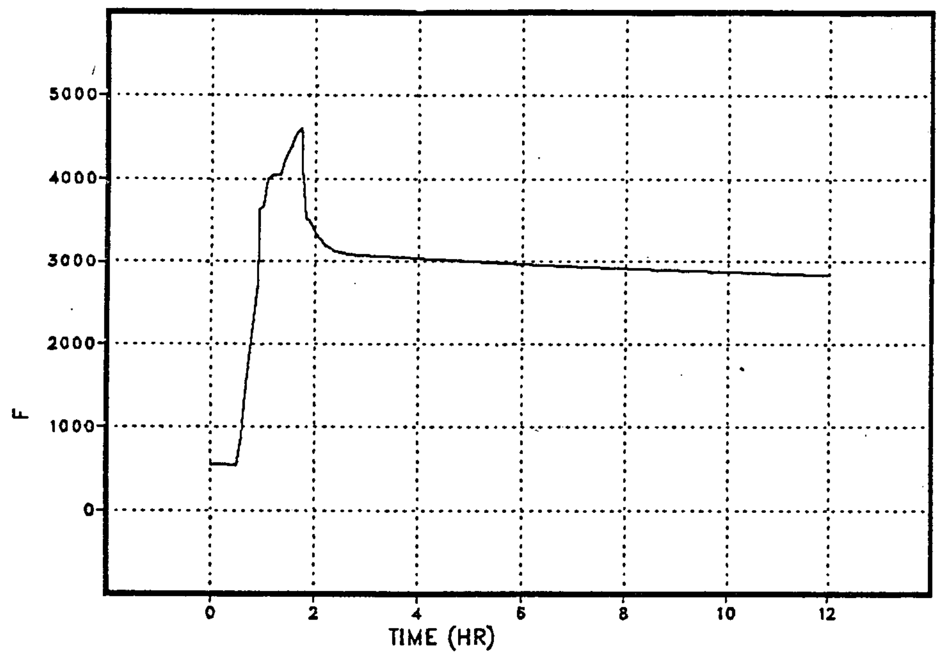


Figure 7.2-7

MNT LOFW NO INJ, 1 ADS
@ 2/3 CORE 3 ADS @ 700#

Maximum temperature in
core. (T110P)



Pressure in the
drywell. (PDW)

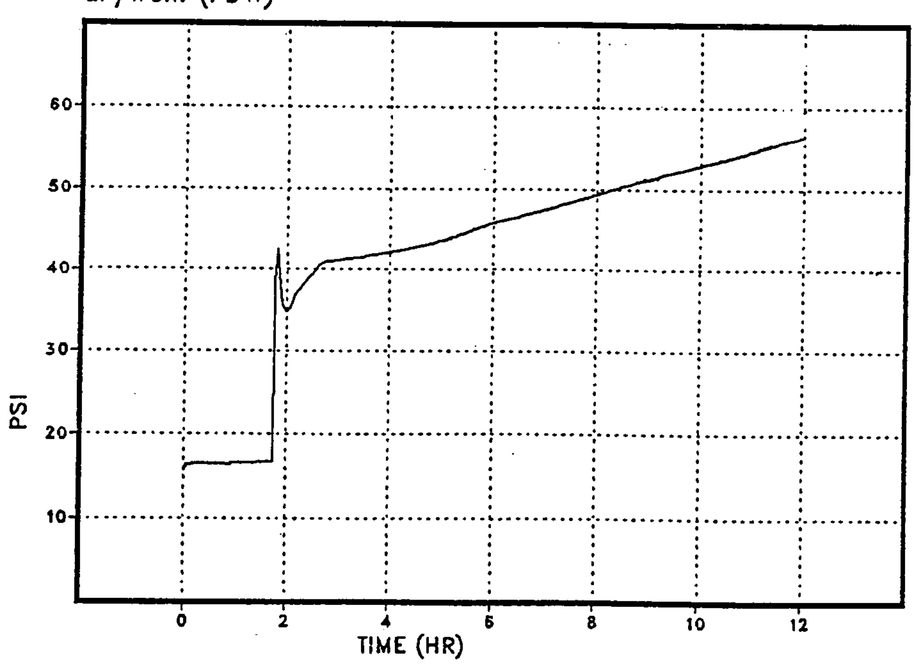
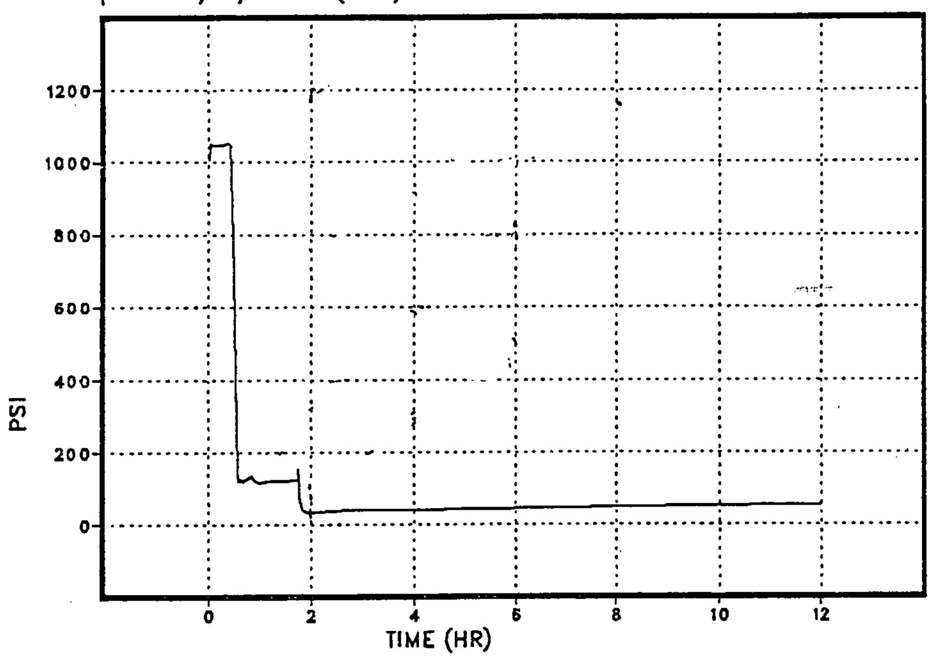


Figure 7.2-8

MNT LOFW NO INJ, 1 ADS
@ 2/3 CORE 3 ADS @ 700#

Pressure in the
primary system. (PPS)



Concrete ablation thickness in
the pedestal (XCNPDP)

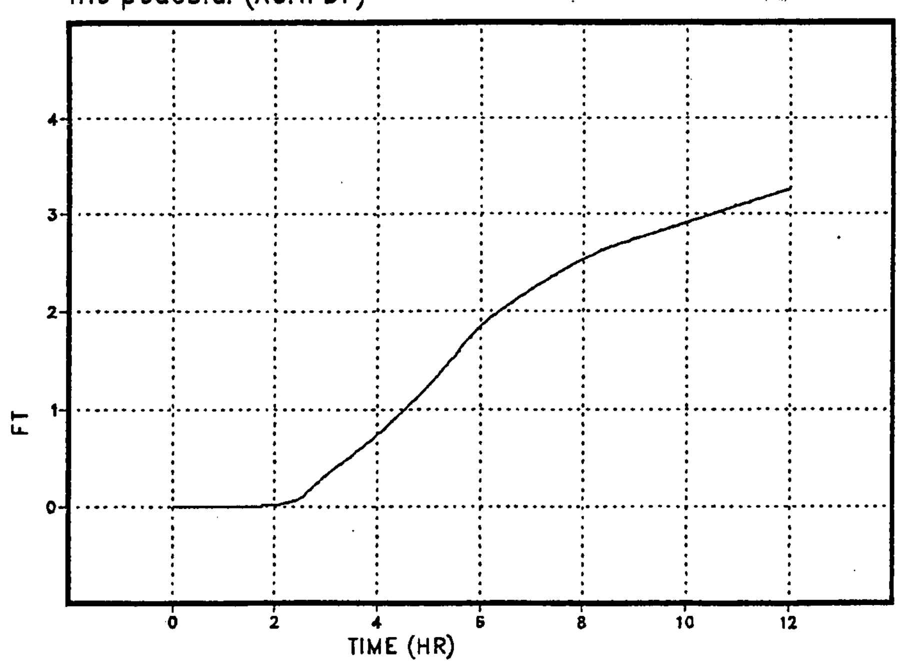
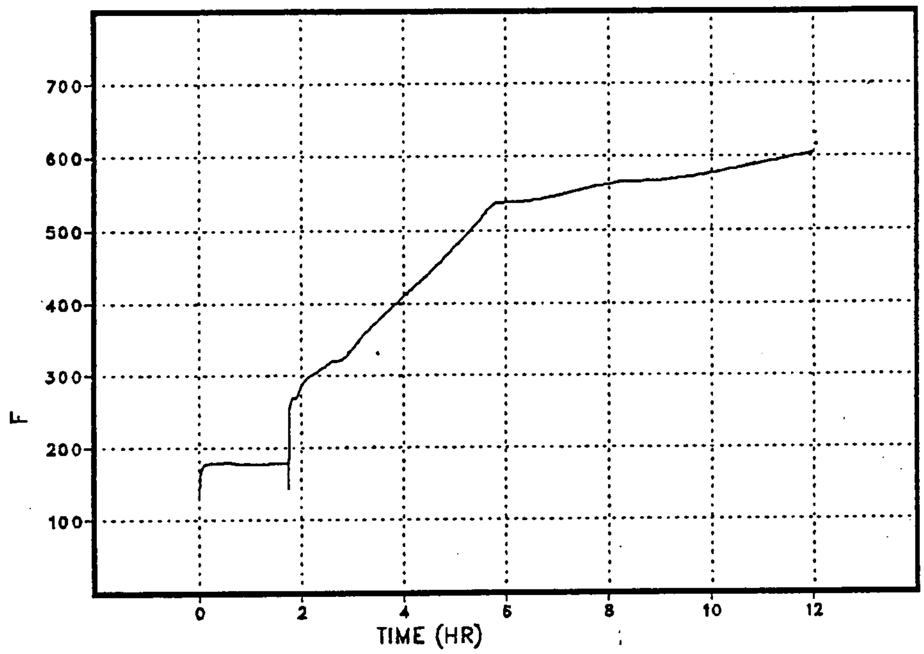


Figure 7.2-9

MNT LOFW NO INJ, 1 ADS
@ 2/3 CORE 3 ADS @ 700#

Temperature of gas
in the drywell. (TGDW)



Torus water temerature.
(TWSP)

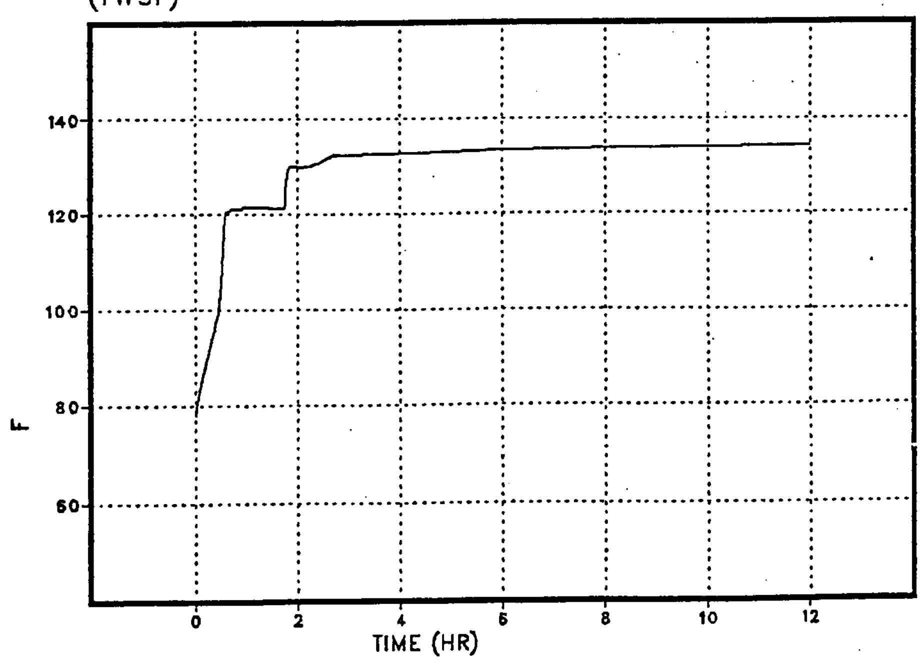
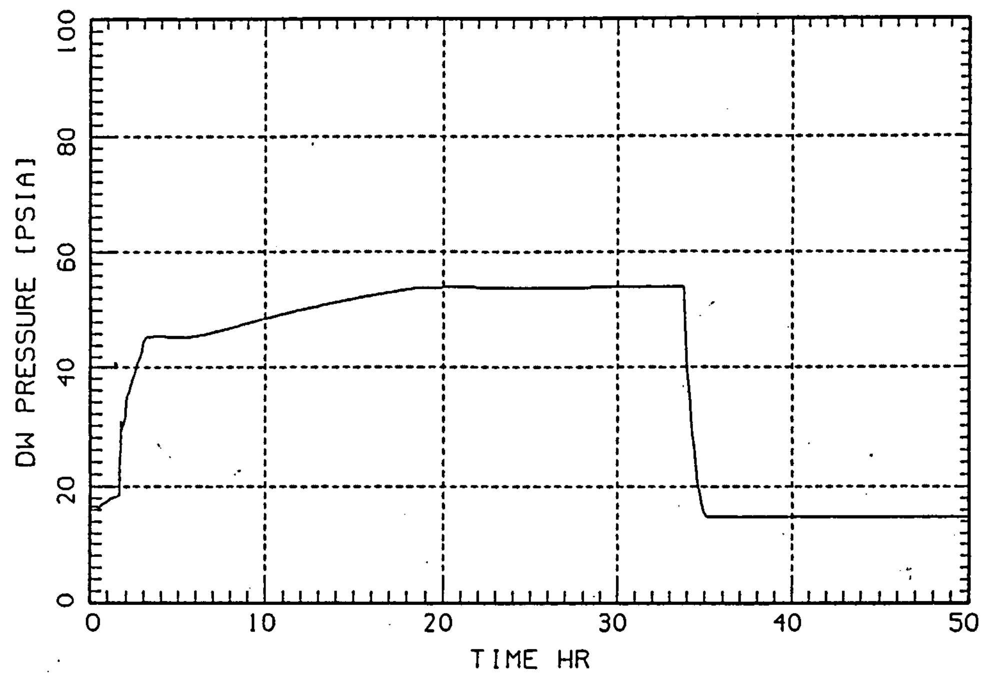


Figure 7.2-10

MSIV Closure with Loss of All Injection; ADS Initiated



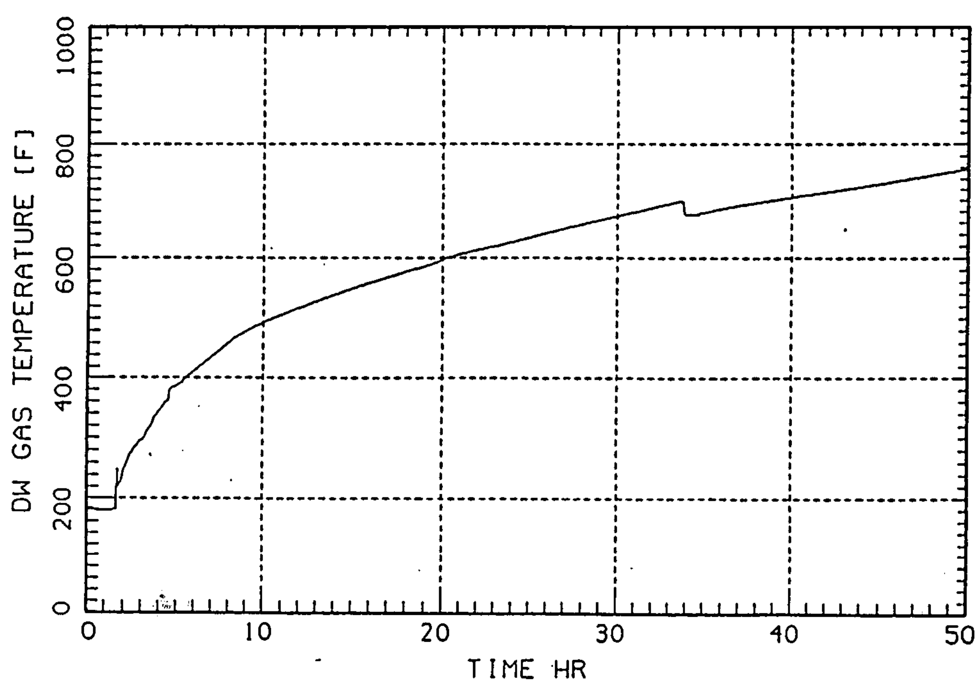
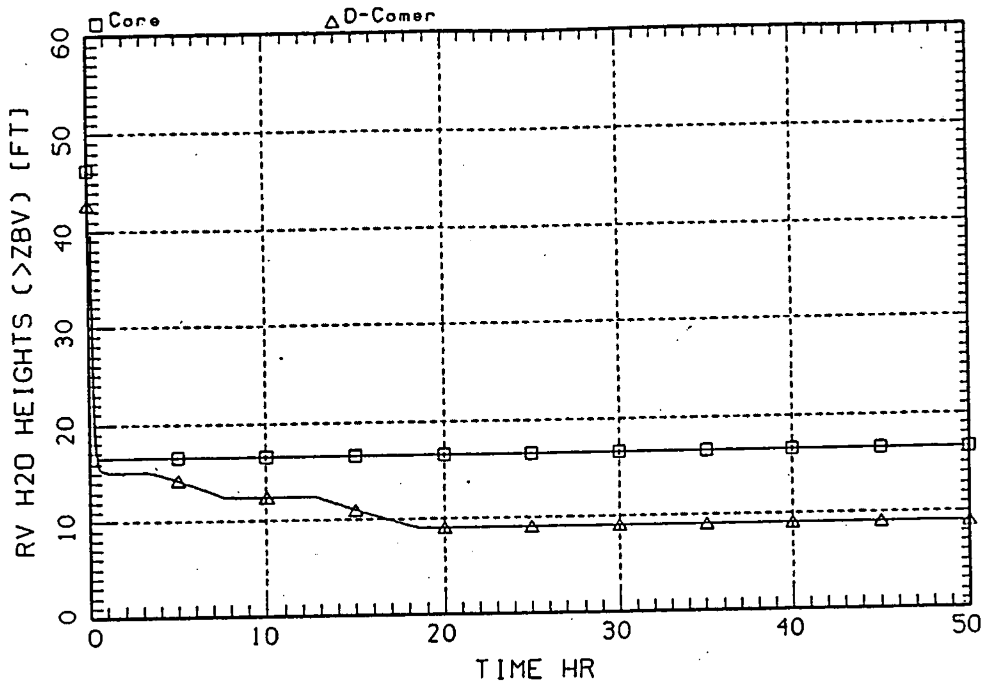


Figure 7.2-11

MSIV Closure with Loss of All Injection; ADS Initiated



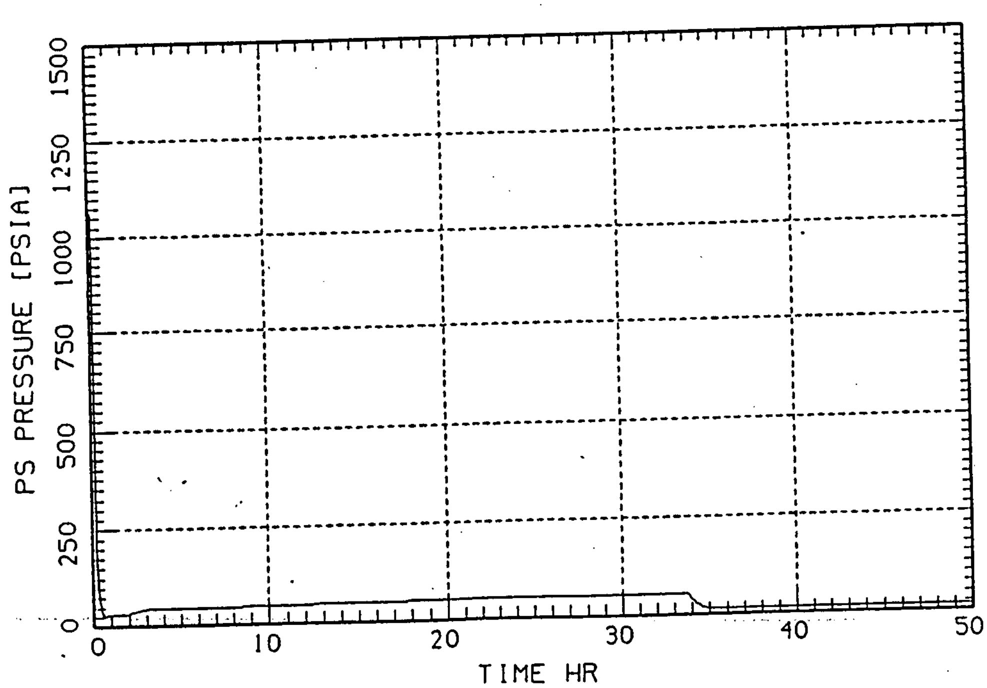


Figure 7.2-12

MSIV Closure with Loss of All Injection; ADS Initiated

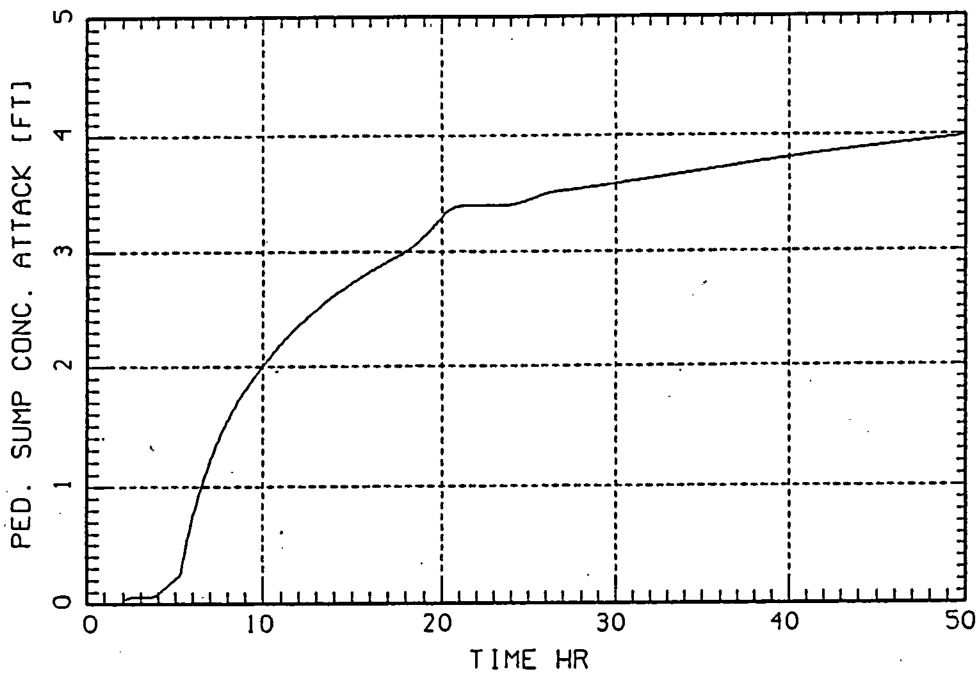
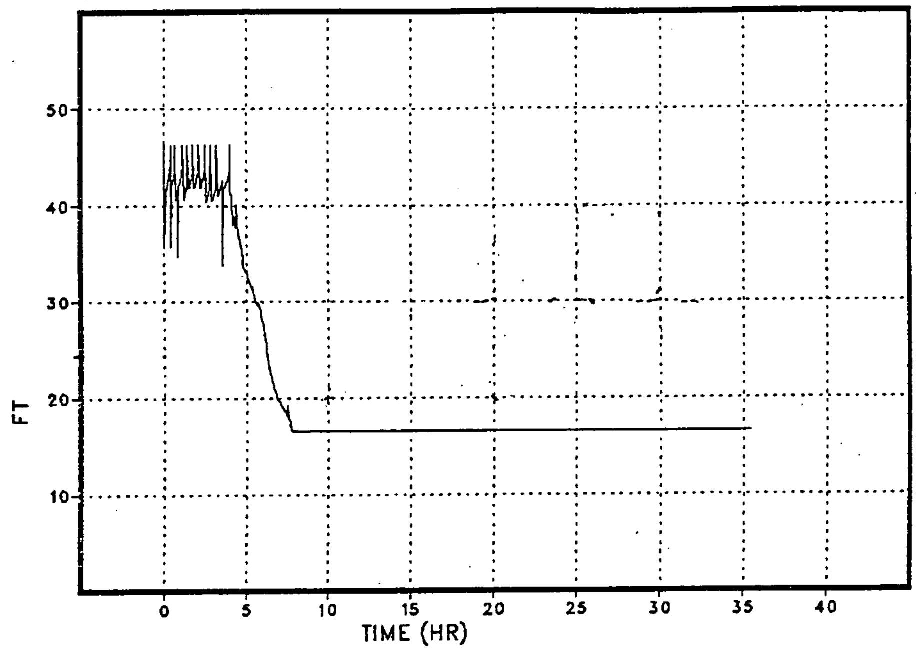


Figure 7.2-13

MAAP MNT SBO W/ HPCI & RCIC
4 HR BATT, NO ADS, NO OP ACTION

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

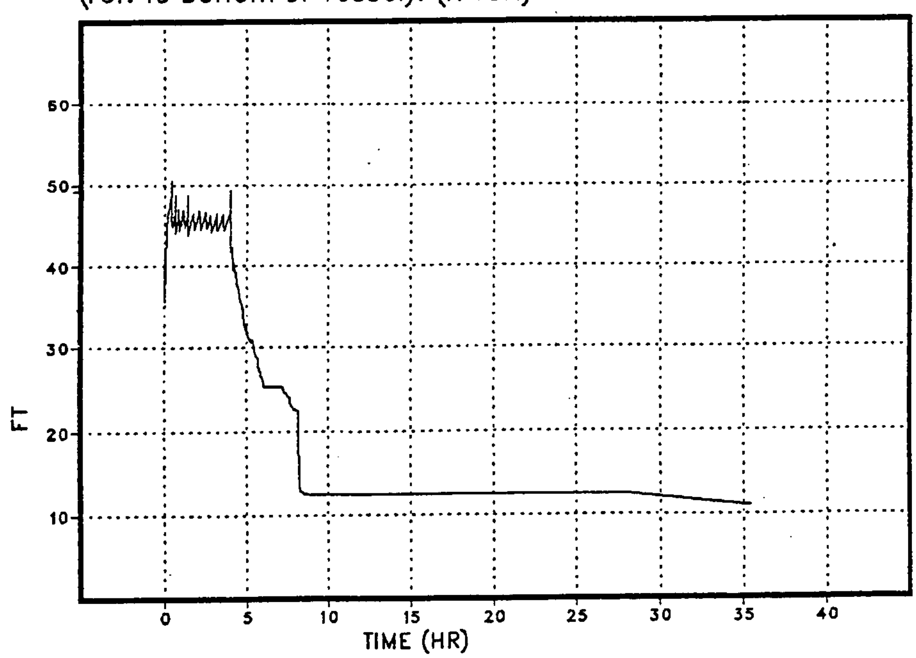


Figure 7.2-14

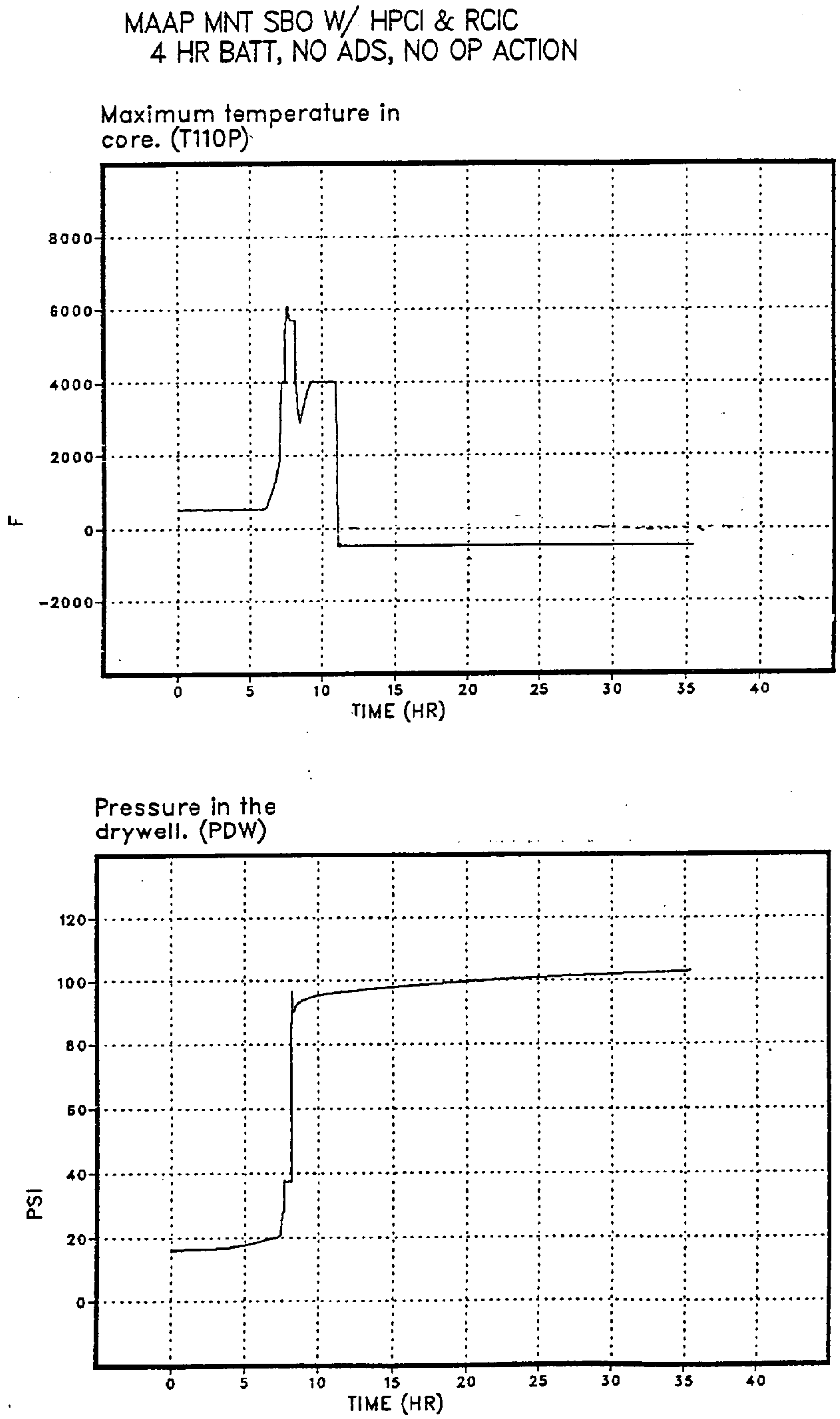
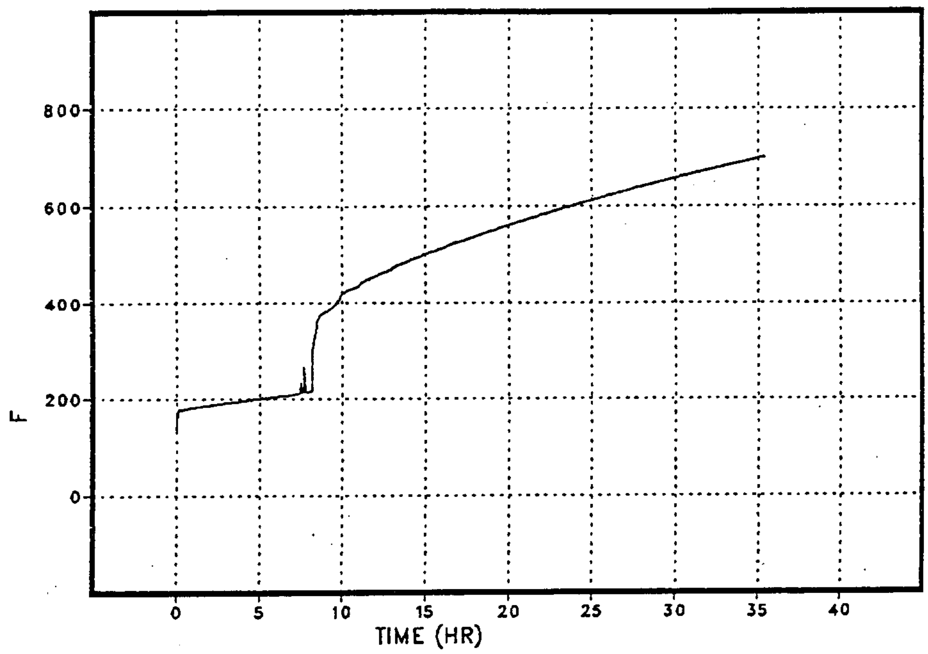

MAAP MNT SBO W/ HPCI & RCIC
4 HR BATT, NO ADS, NO OP ACTION
Maximum temperature in
core. (T110P)
8000
6000
4000
2000
0
-2000
F
0
5
10
15
20
25
30
35
40
TIME (HR)
Pressure in the
drywell. (PDW)
120
100
80
60
40
20
0
PSI
0
5
10
15
20
25
30
35
40
TIME (HR)

Figure 7.2-15

MAAP MNT SBO W/ HPCI & RCIC
4 HR BATT, NO ADS, NO OP ACTION

Temperature of gas
in the drywell. (TGDW)



Torus water temperature.
(TWSP)

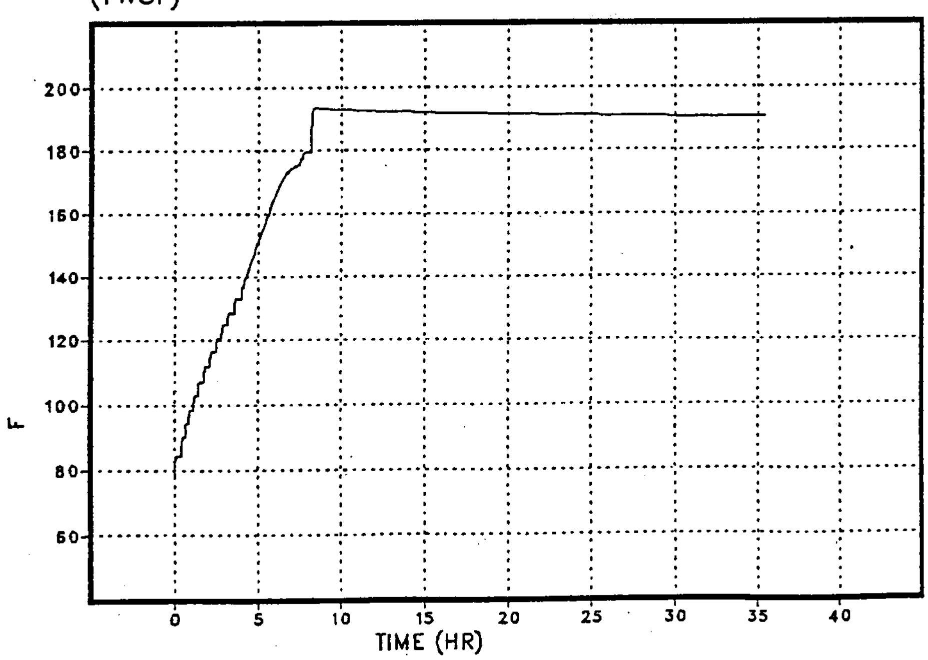
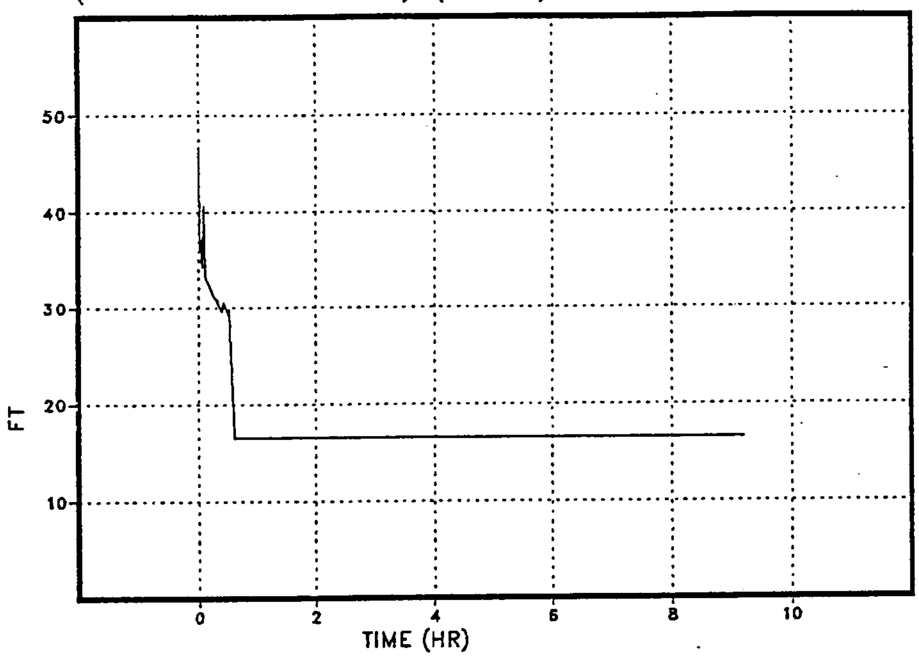


Figure 7.2-16

MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

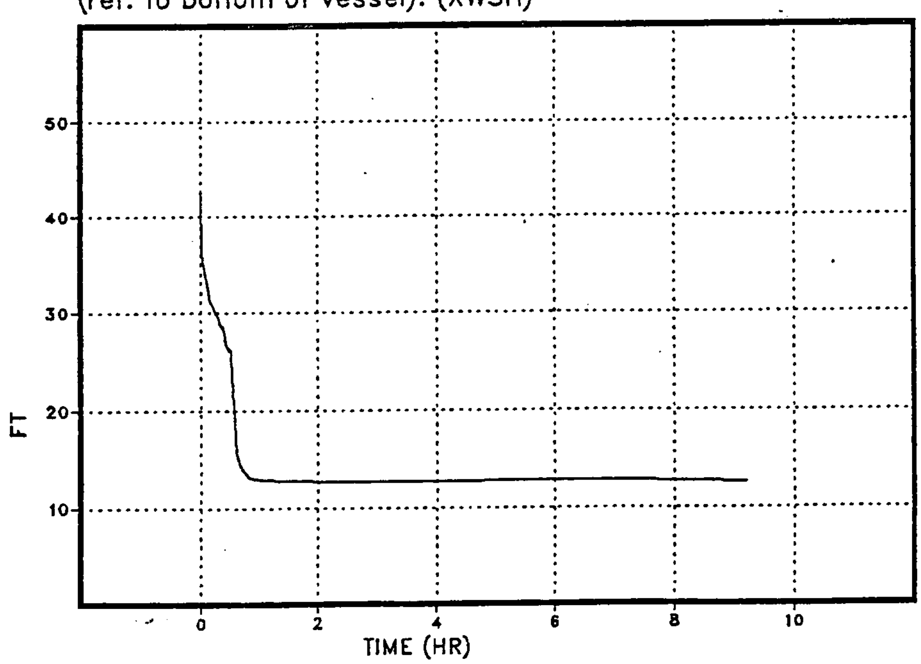
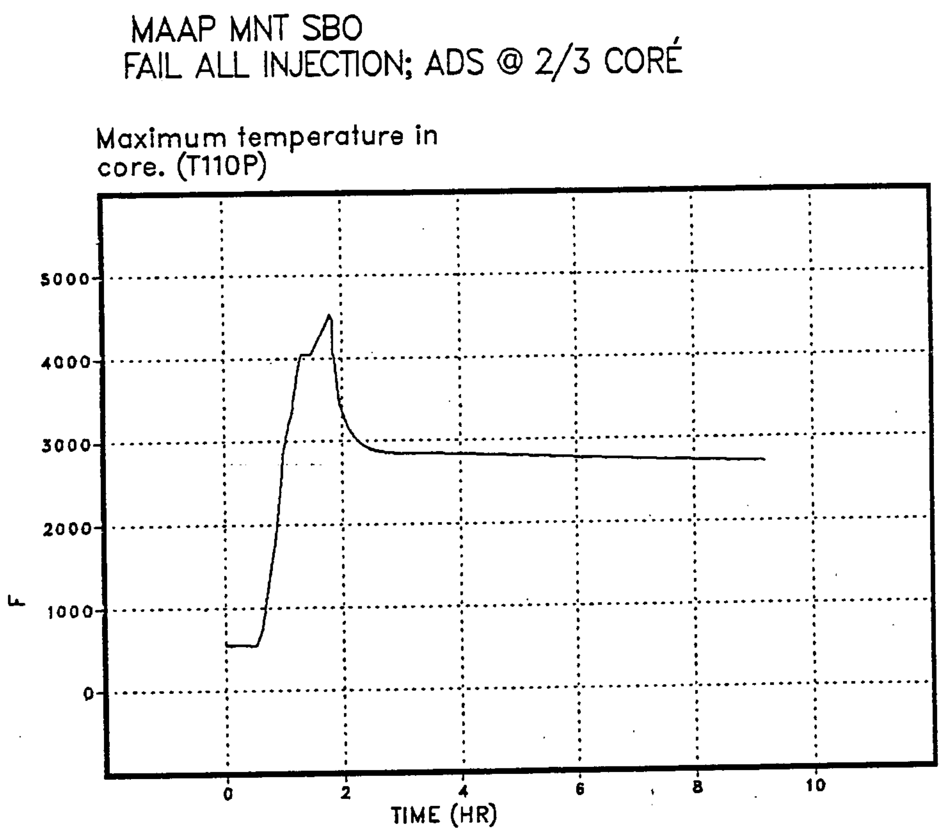
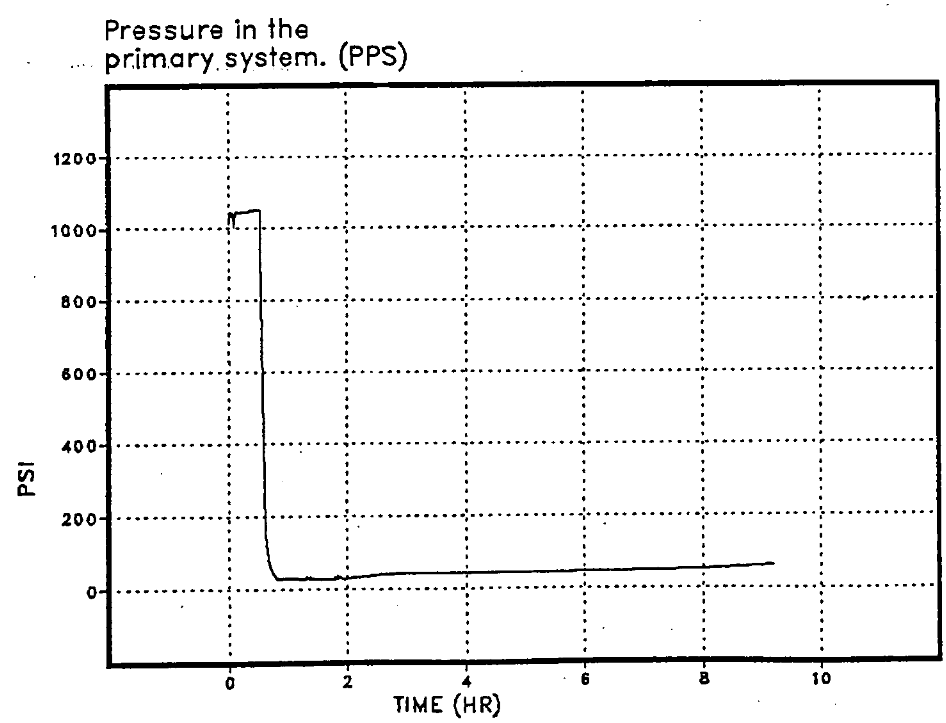


Figure 7.2-17



MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE
Maximum temperature in core. (T110P)
5000
4000
3000
2000
1000
0
F
0
2
4
6
8
10
TIME (HR)

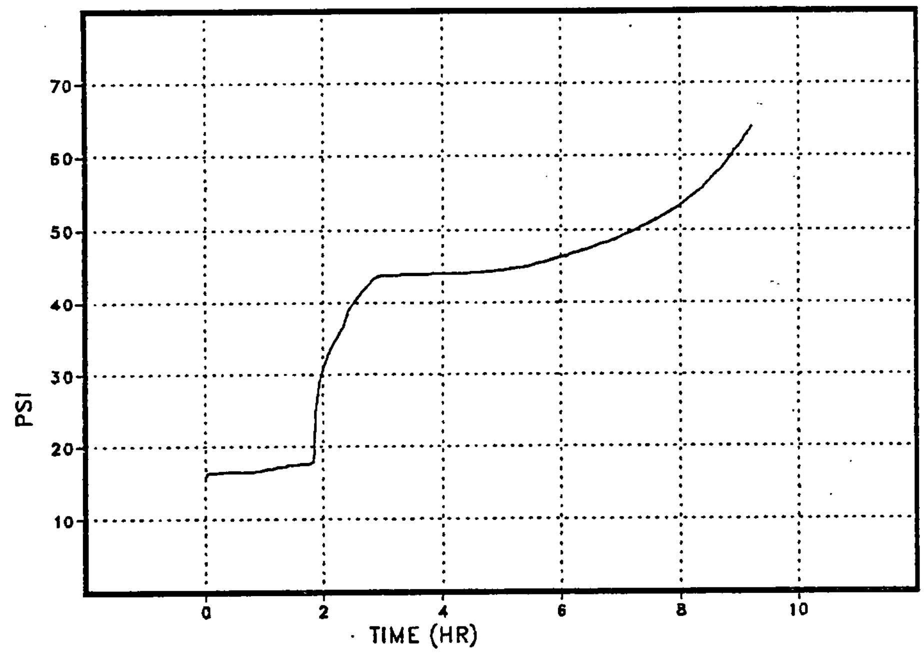


Pressure in the primary system. (PPS)
1200
1000
800
600
400
200
0
PSI
0
2
4
6
8
10
TIME (HR)

Figure 7.2-18

MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE

Pressure in the drywell. (PDW)



Concrete ablation thickness in the pedestal (XCNPDP)

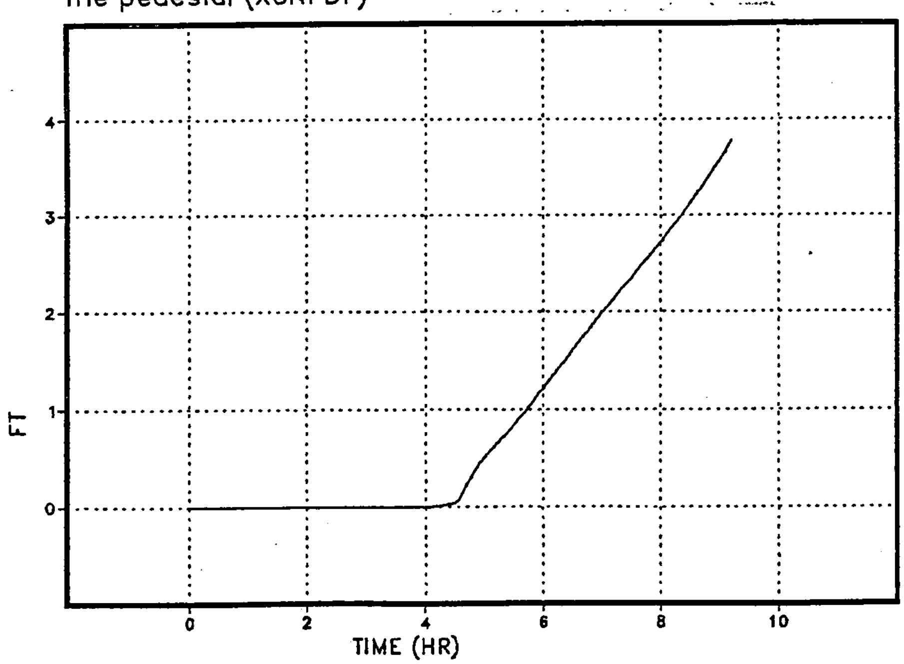
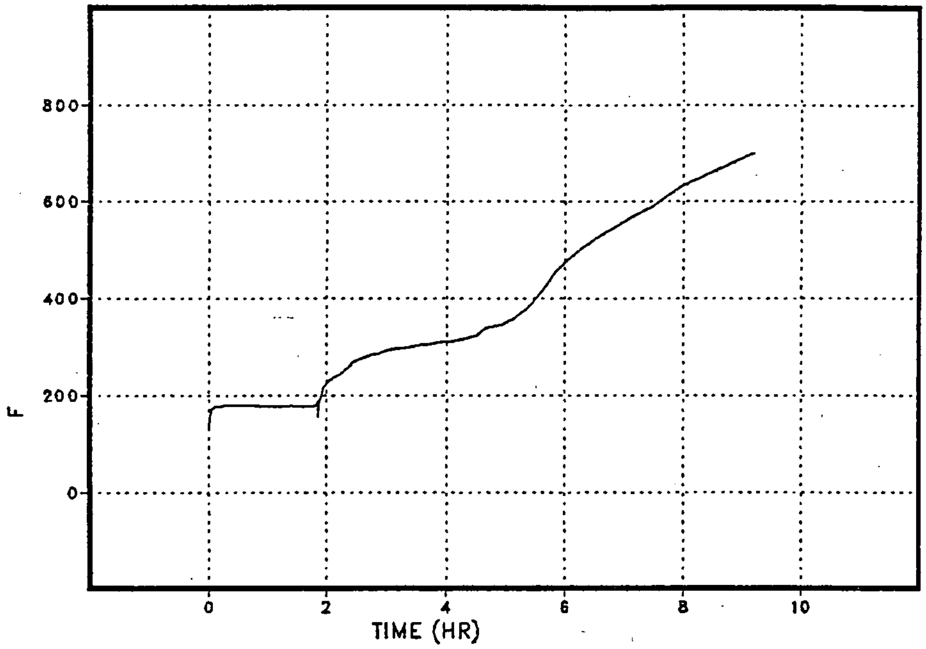


Figure 7.2-19

MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE

Temperature of gas
in the drywell. (TGDW)



Torus water temerature.
(TWSP)

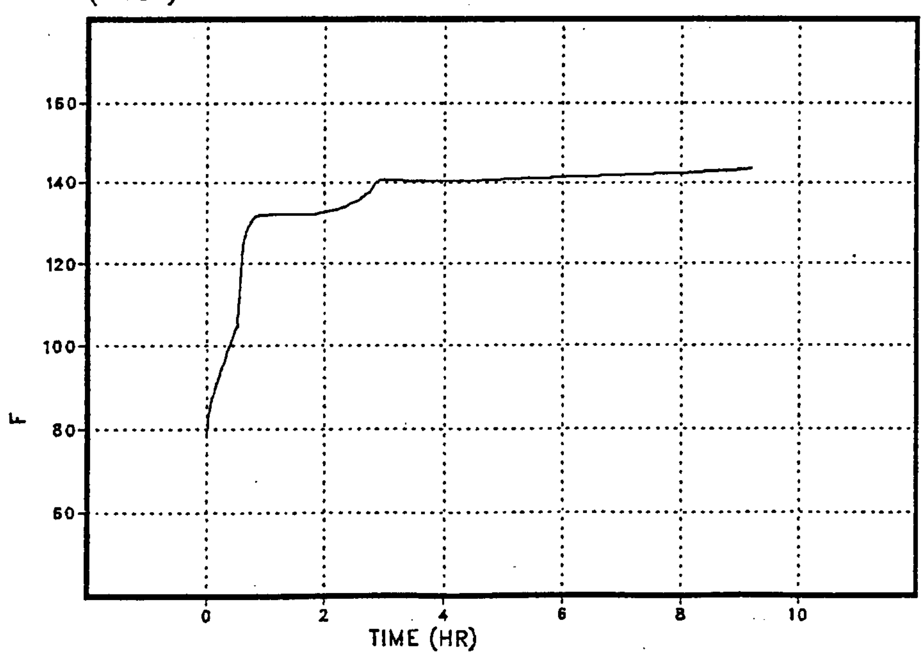
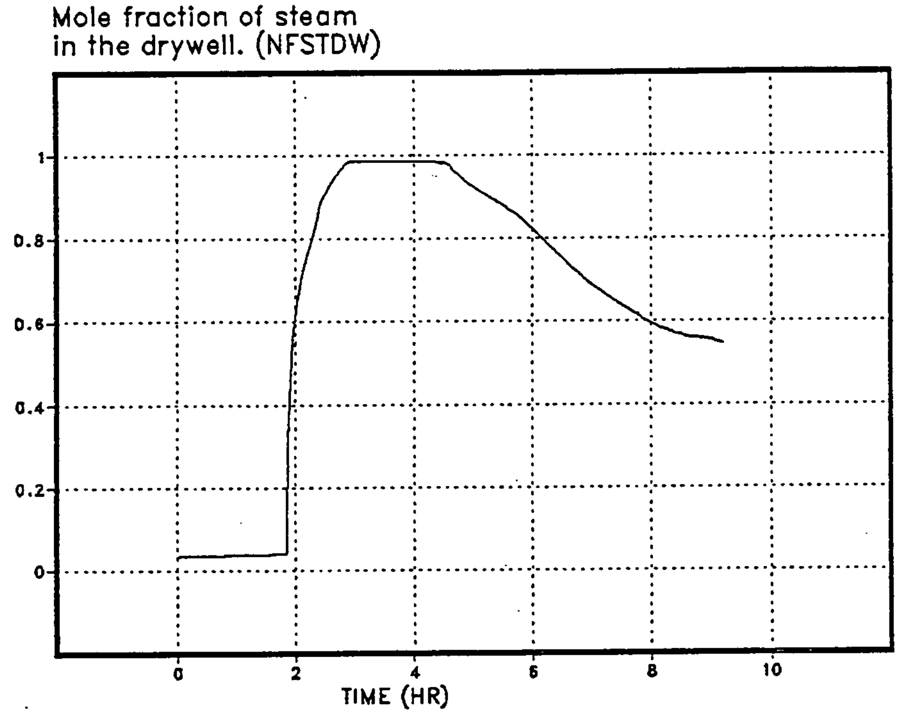


Figure 7.2-20

MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE



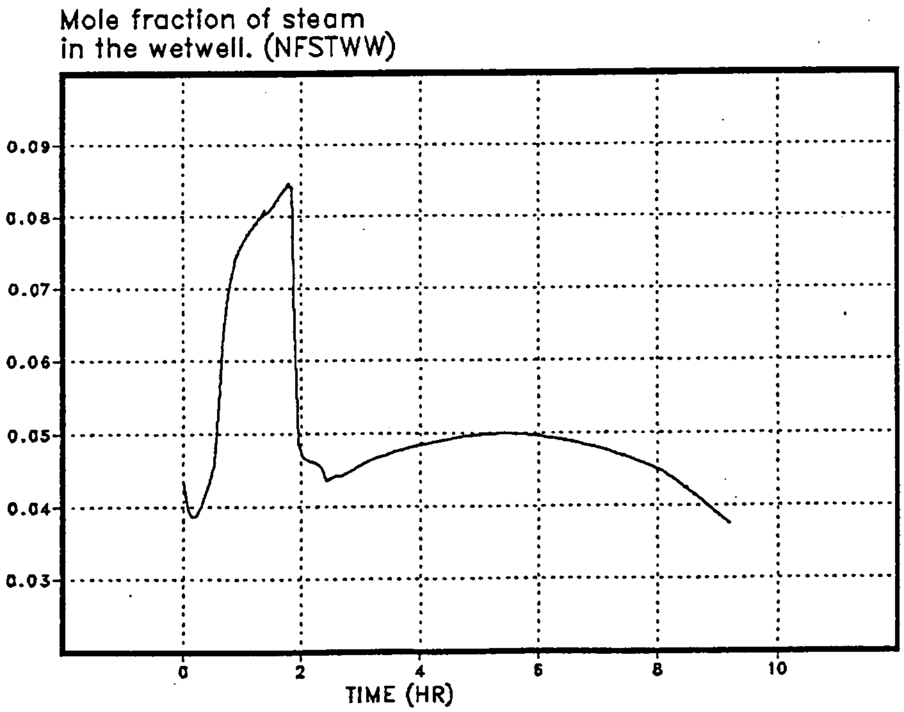
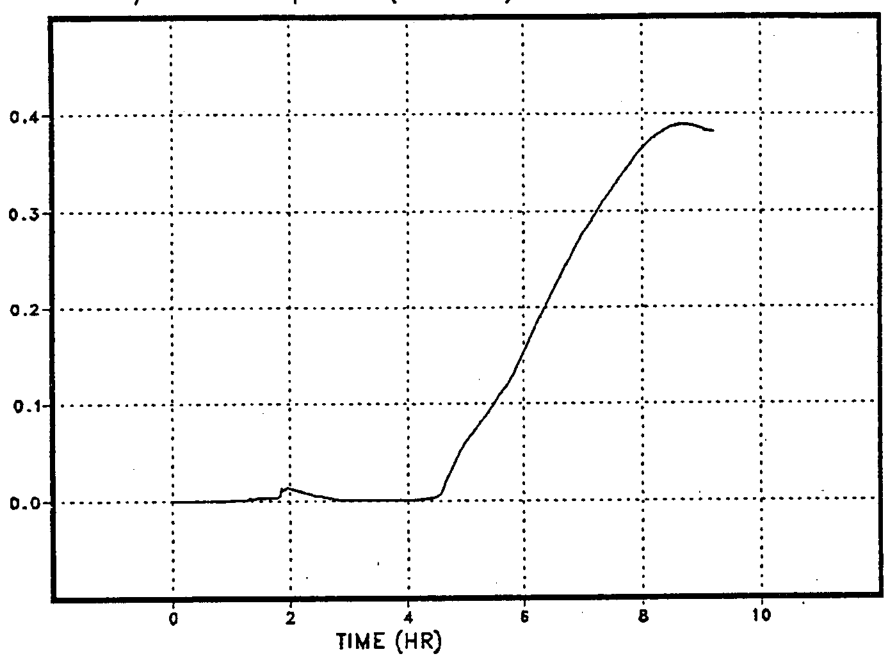


Figure 7.2-21

MAAP MNT SBO
FAIL ALL INJECTION; ADS @ 2/3 CORE

Mole fraction of hydrogen in
the drywell atmosphere. (NFH2DW)



Mole fraction of H2
in the wetwell. (NFH2WW)

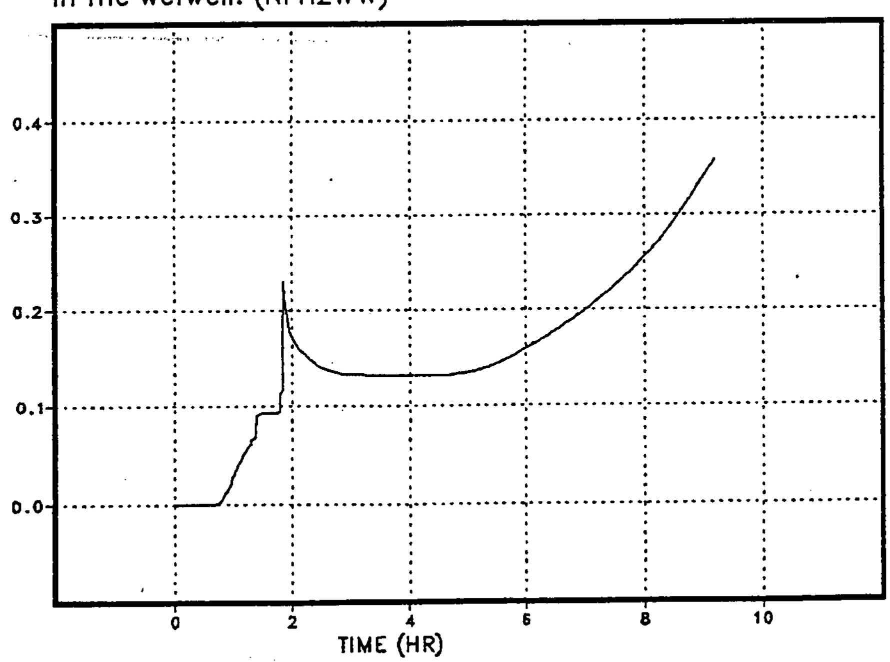
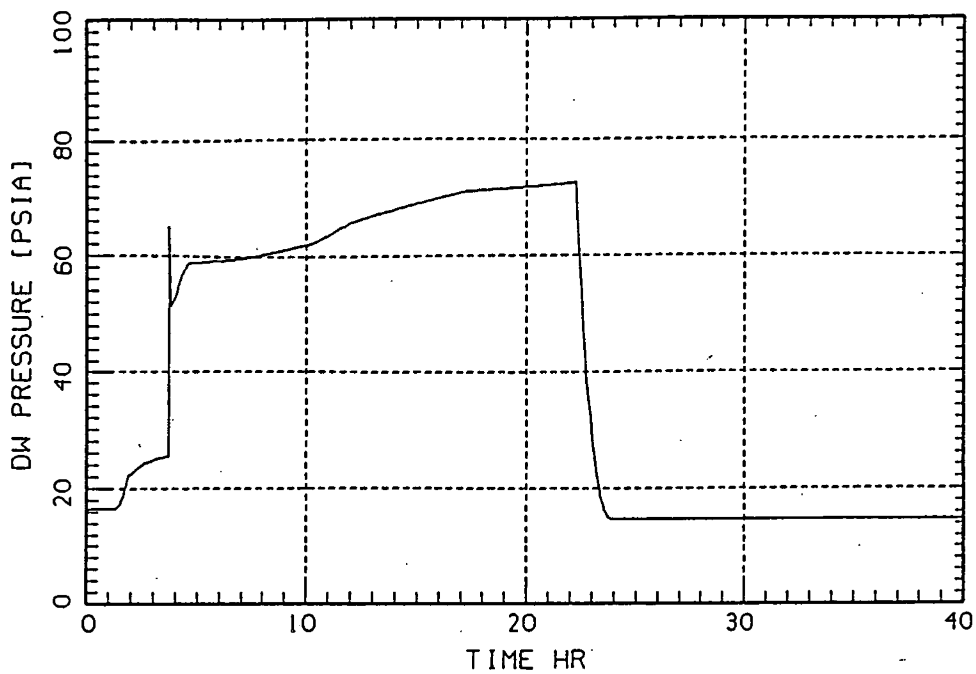


Figure 7.2-22

Station Blaokout with No Injeotion and No Depressurization



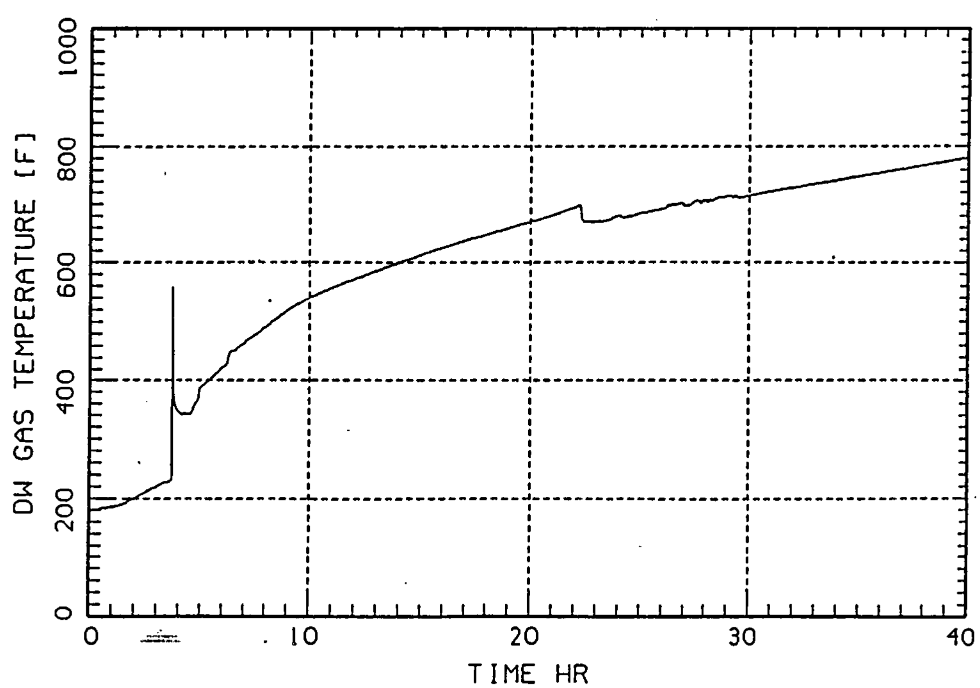
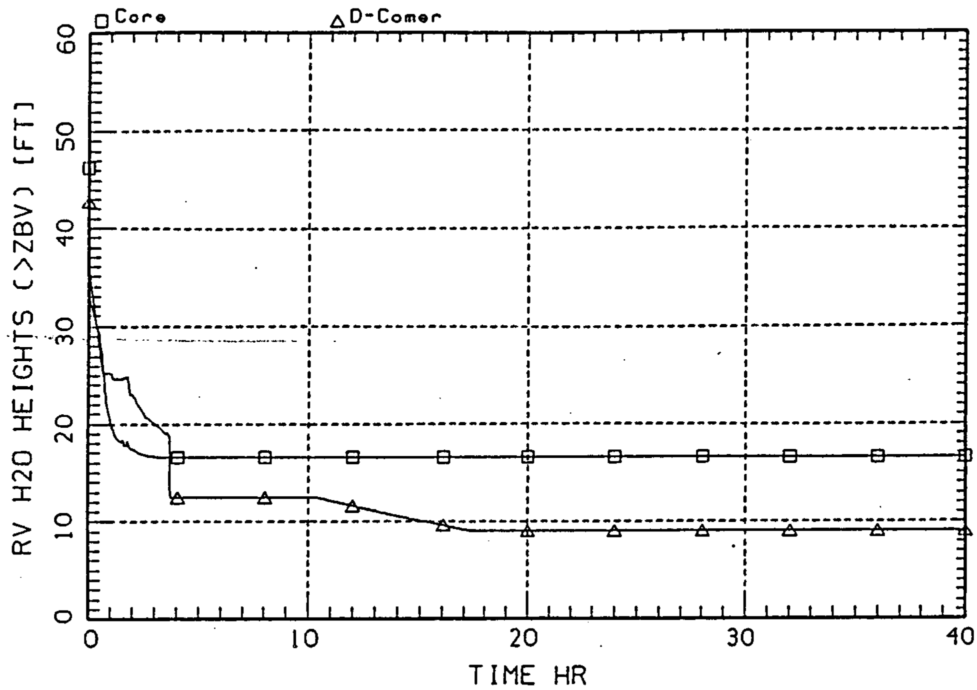


Figure 7.2-23

Station Blackout with No Injeotion and No Depressurization



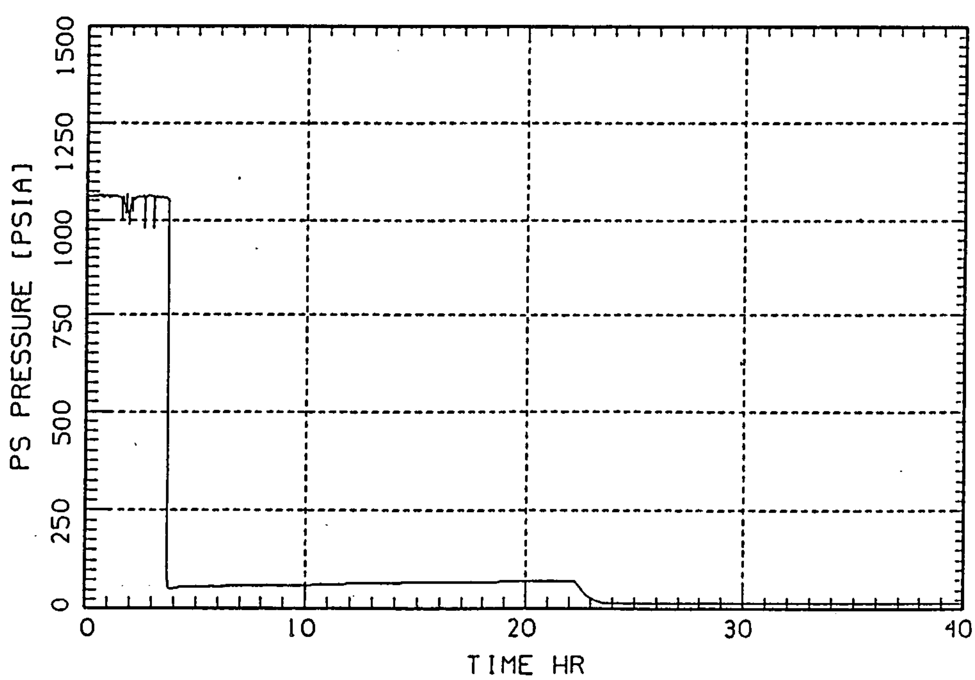


Figure 7.2-24

Station Blackout with No Injection and No Depressurization

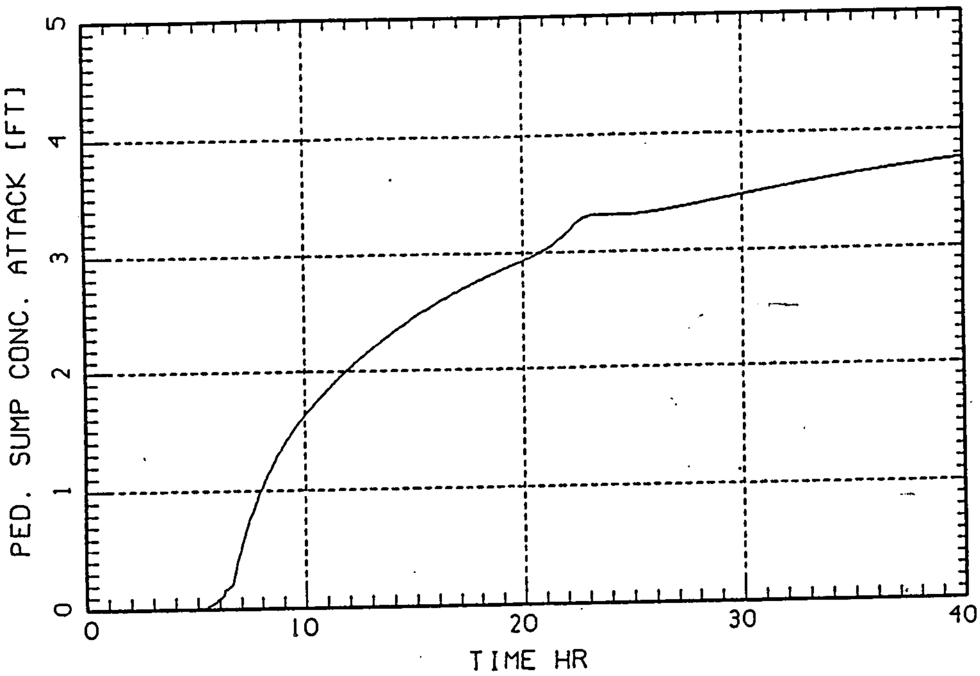
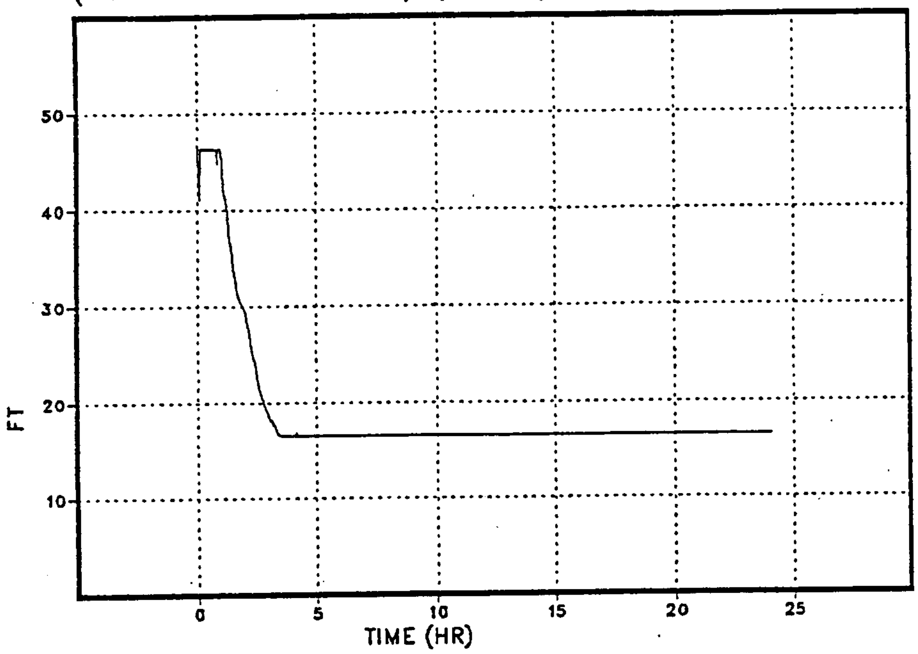


Figure 7.2-25

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

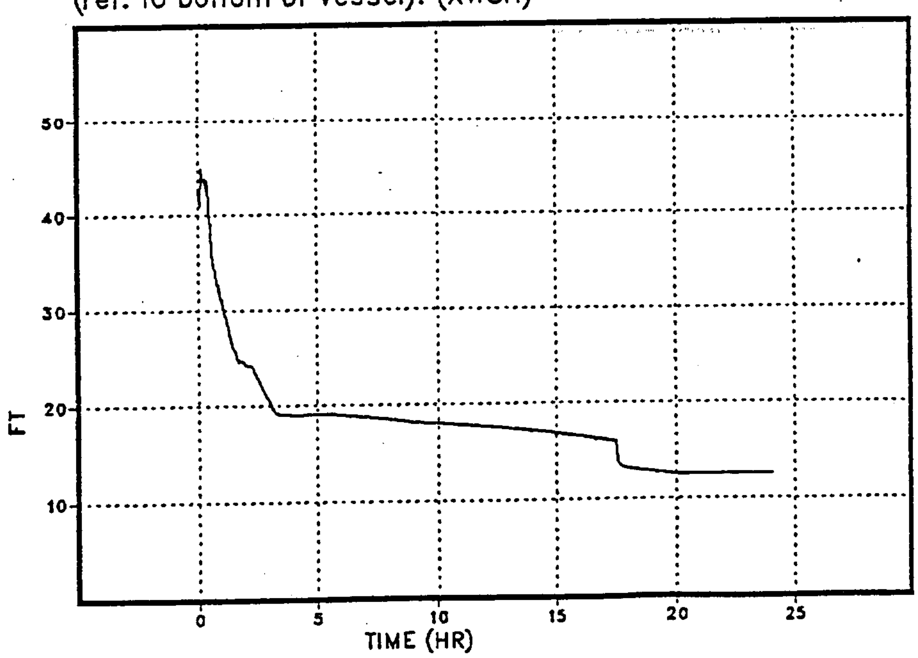
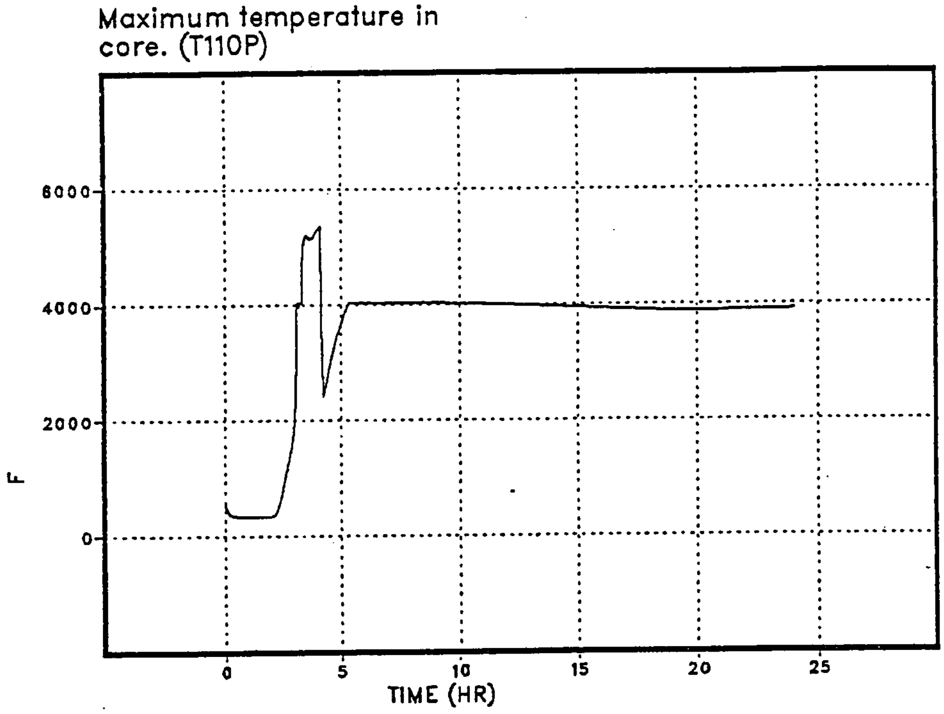


Figure 7.2-26

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS



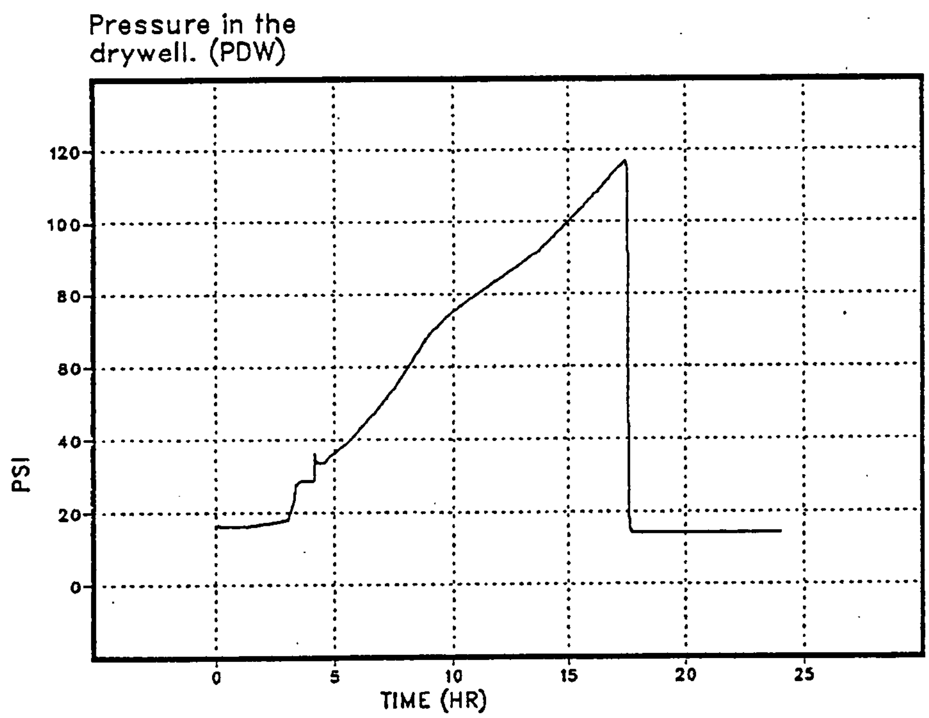
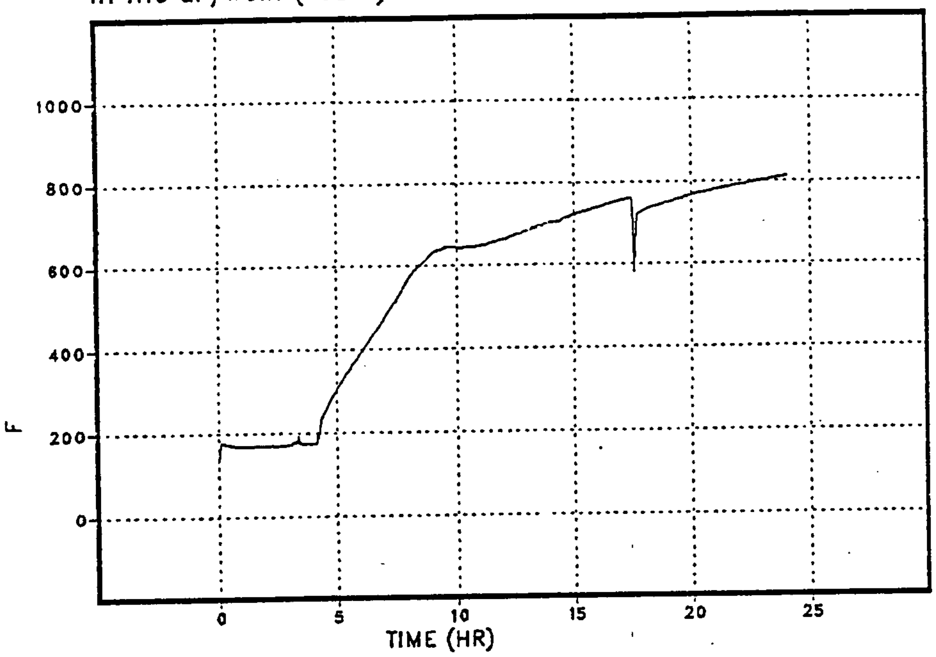


Figure 7.2-27

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS

Temperature of gas
in the drywell. (TGDW)



Torus water temerature.
(TWSP)

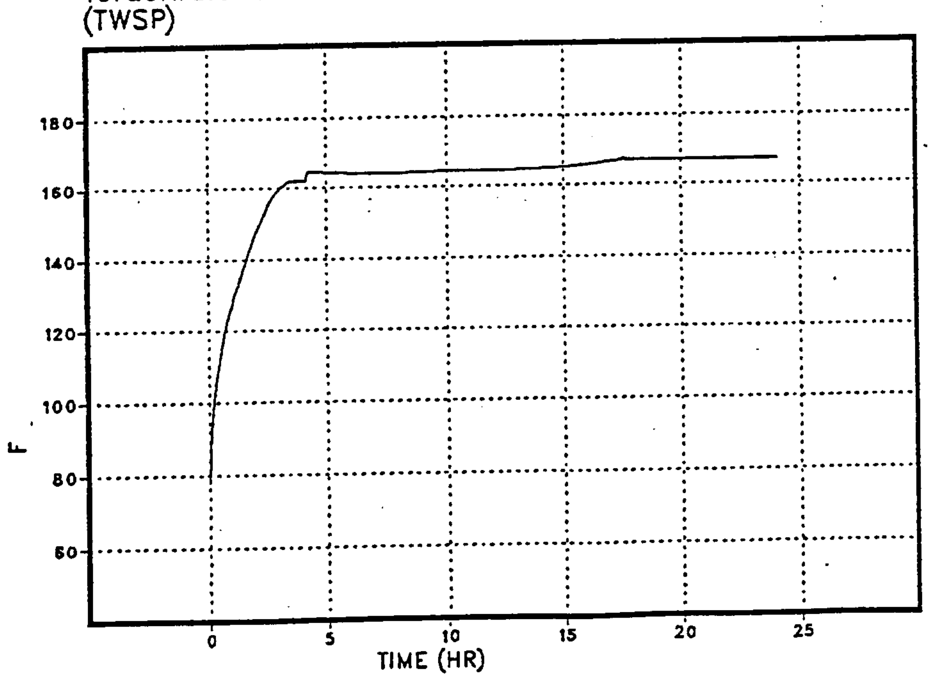
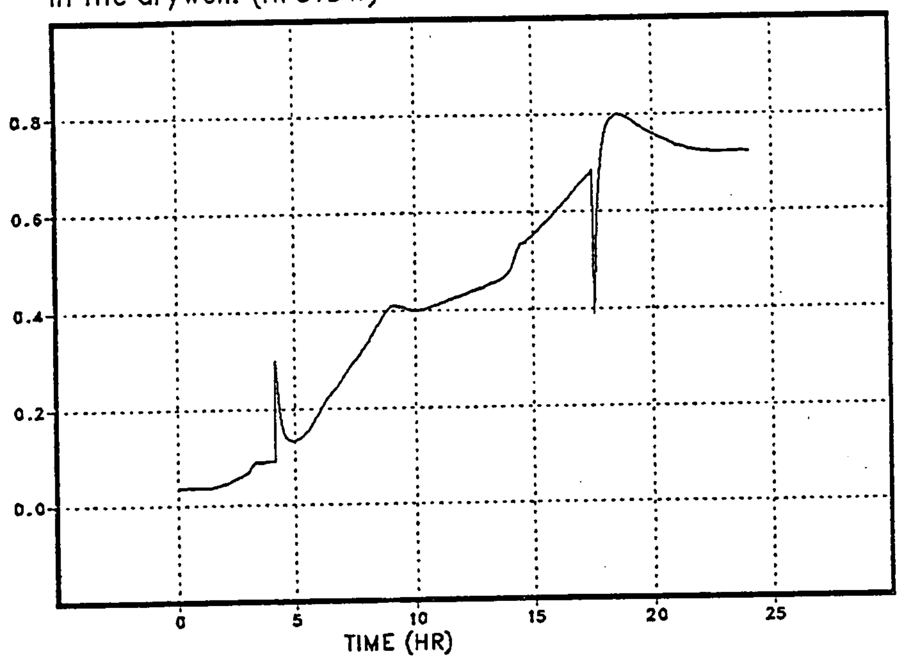


Figure 7.2-28

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

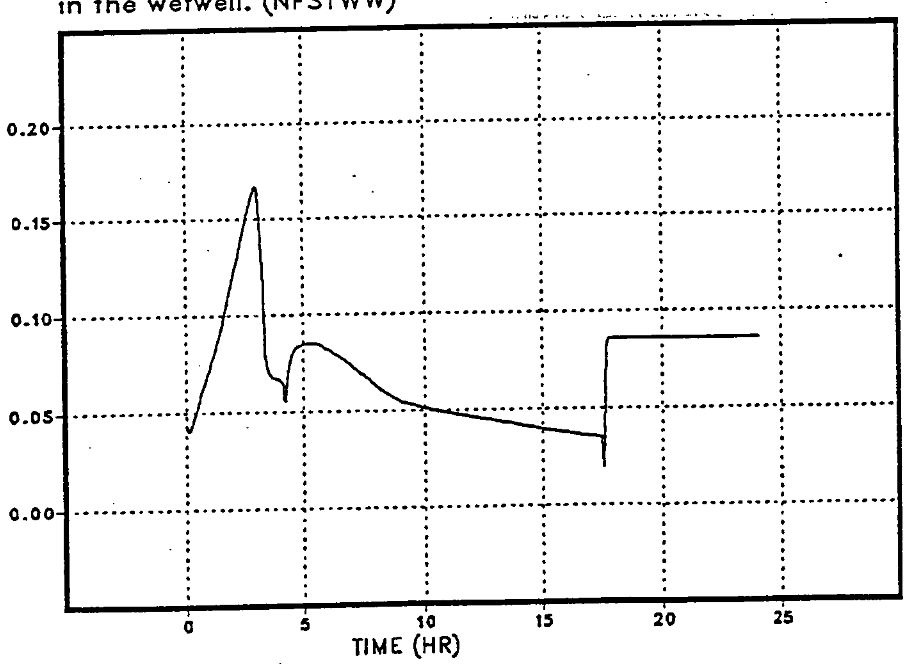
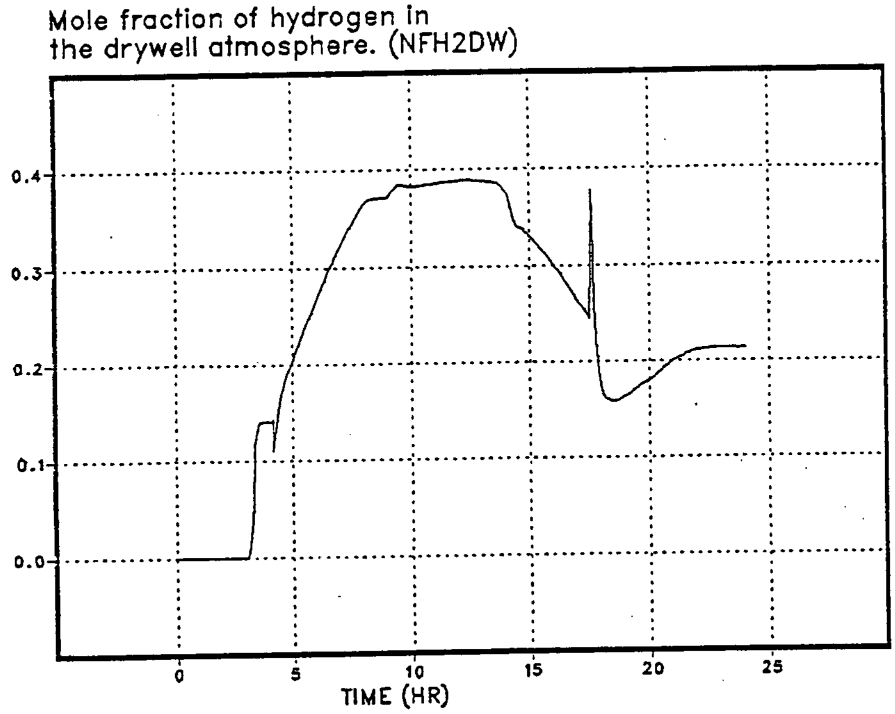


Figure 7.2-29

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS



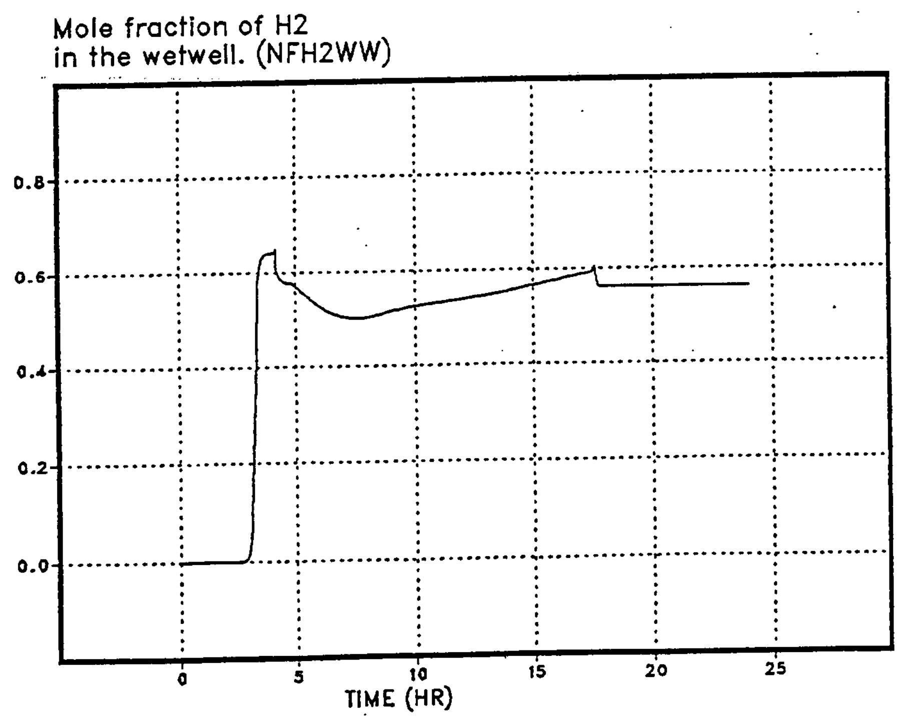
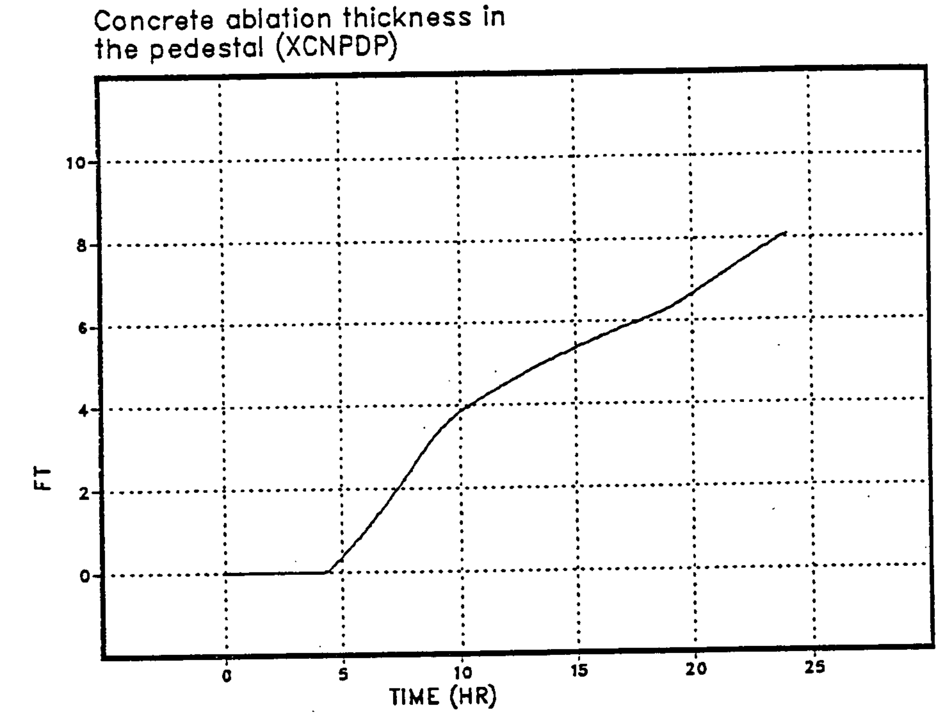


Figure 7.2-30

MAAP MNT SBO W/ 1 SORV
HPCI AVAILABLE ONLY; NO ADS



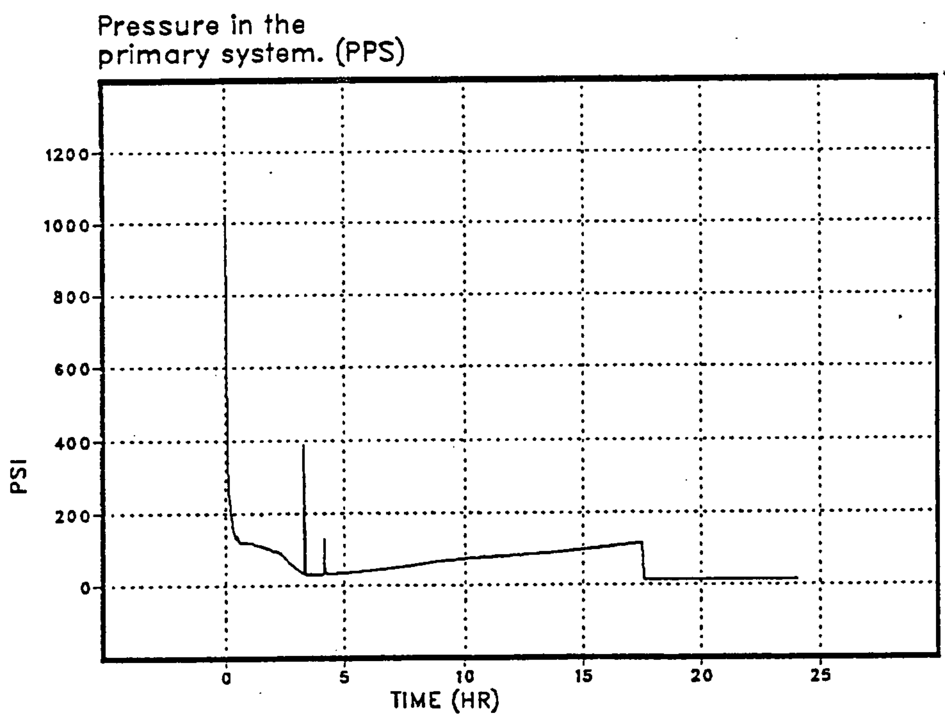
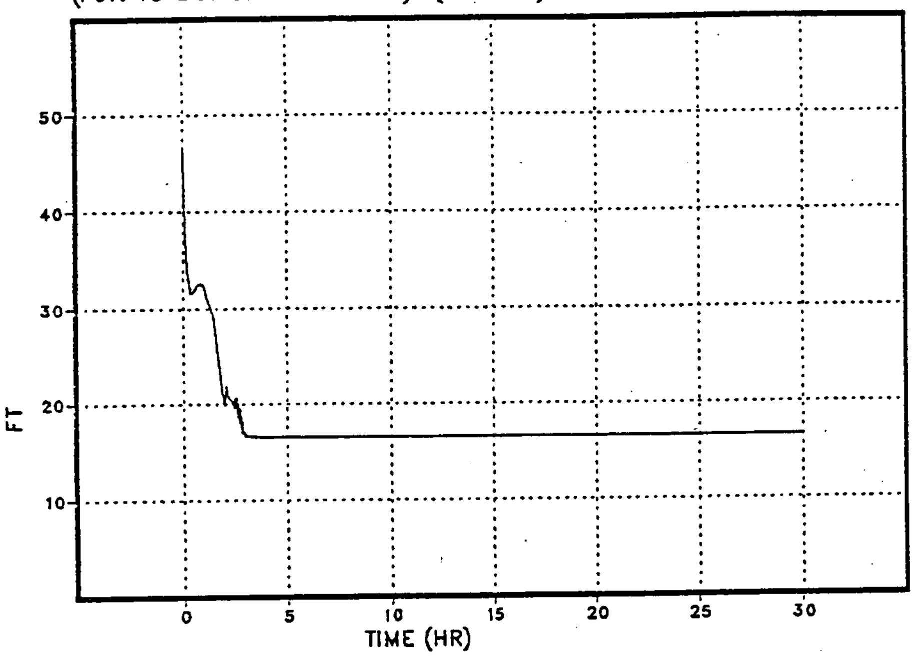


Figure 7.2-31

MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

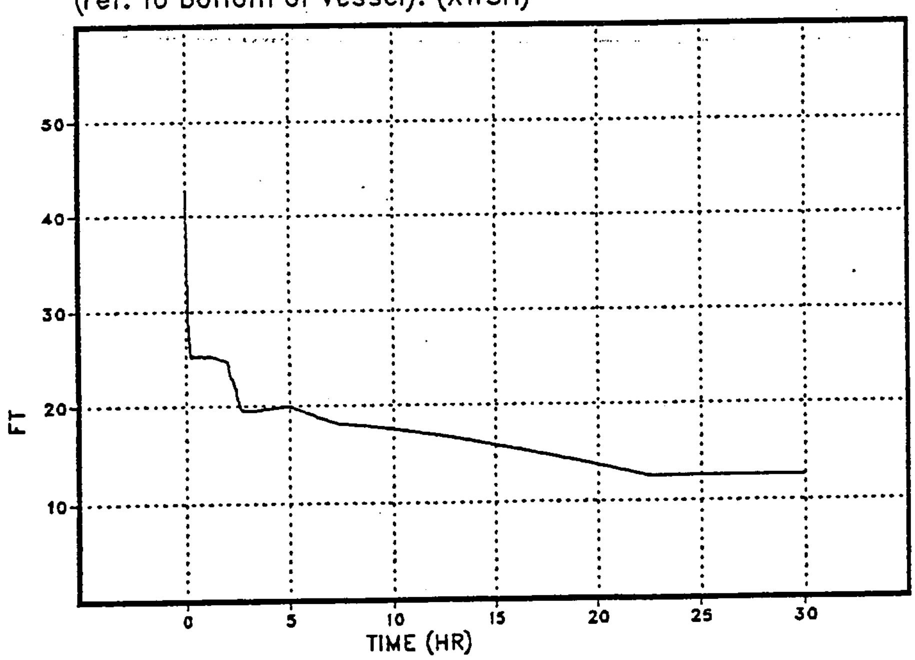
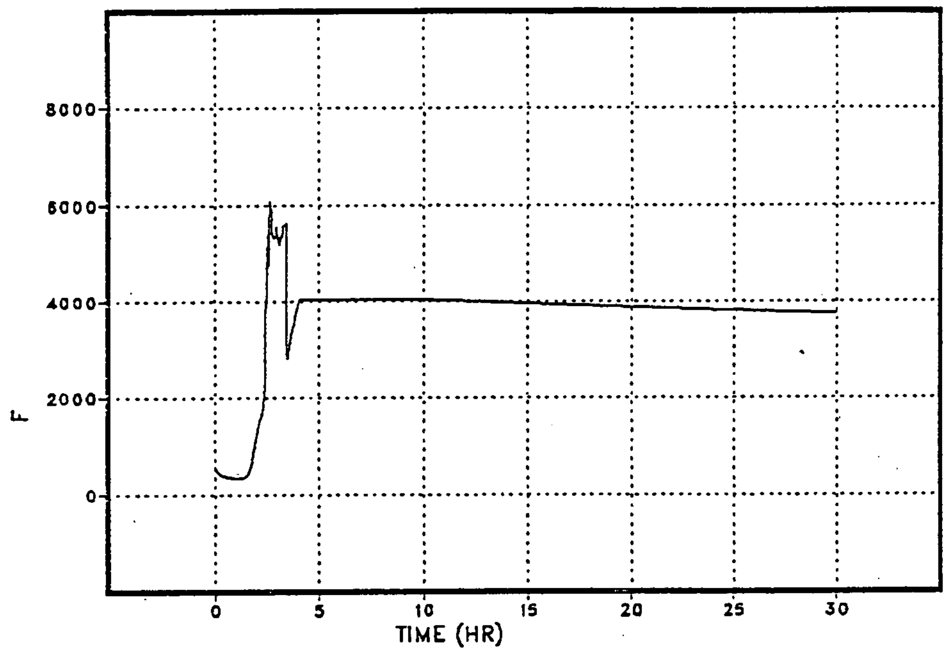


Figure 7.2-32

MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS

Maximum temperature in
core. (T110P)



Pressure in the
drywell. (PDW)

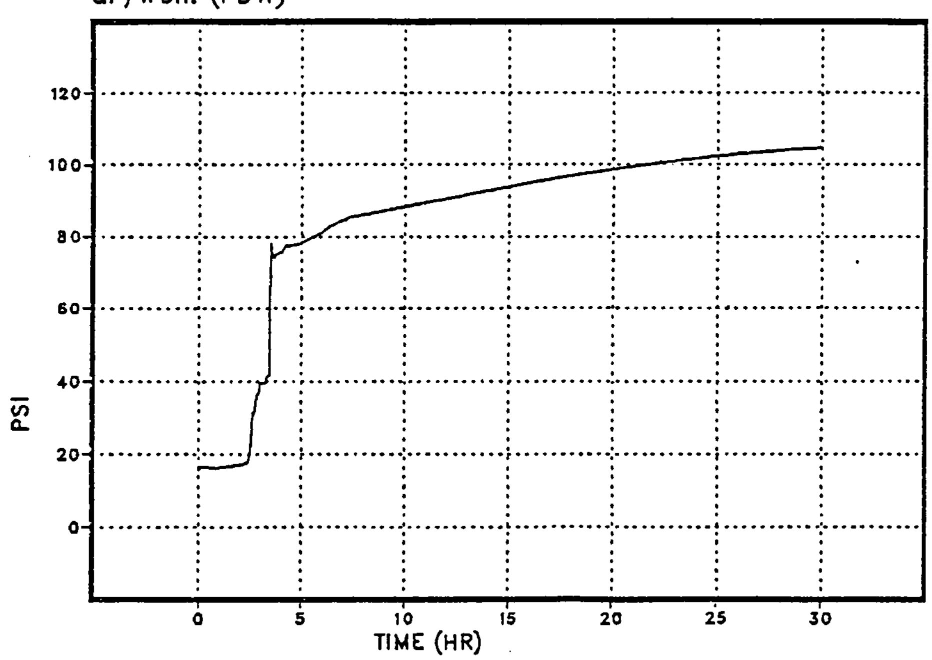
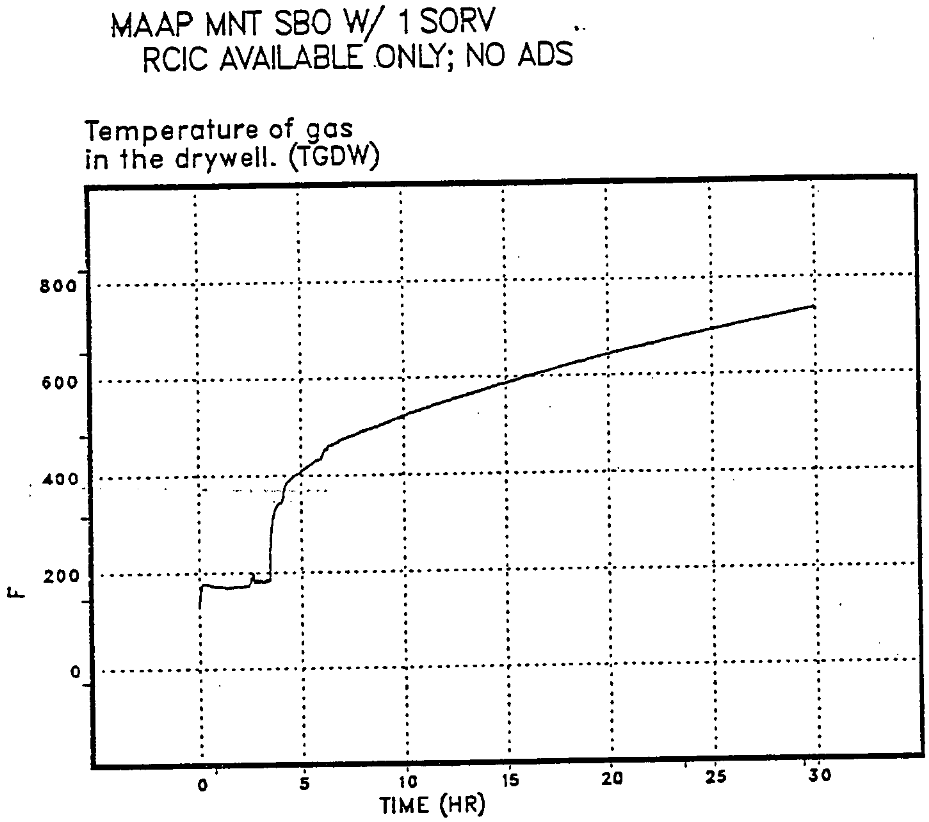


Figure 7.2-33


MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS
Temperature of gas
in the drywell. (TGDW)
800
600
400
200
0
F
0
5
10
15
20
25
30
TIME (HR)

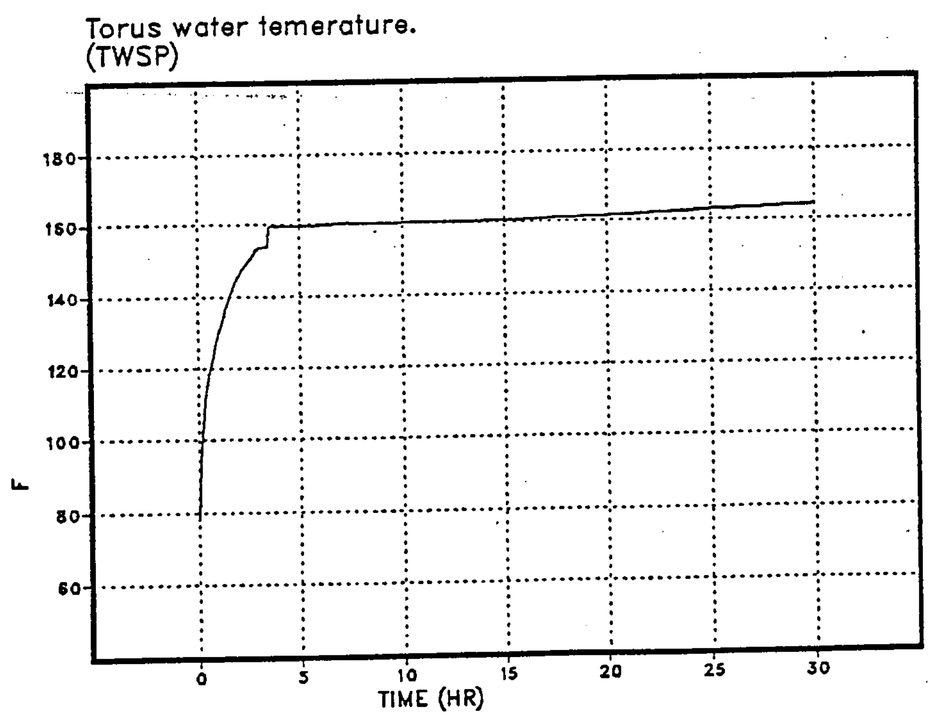
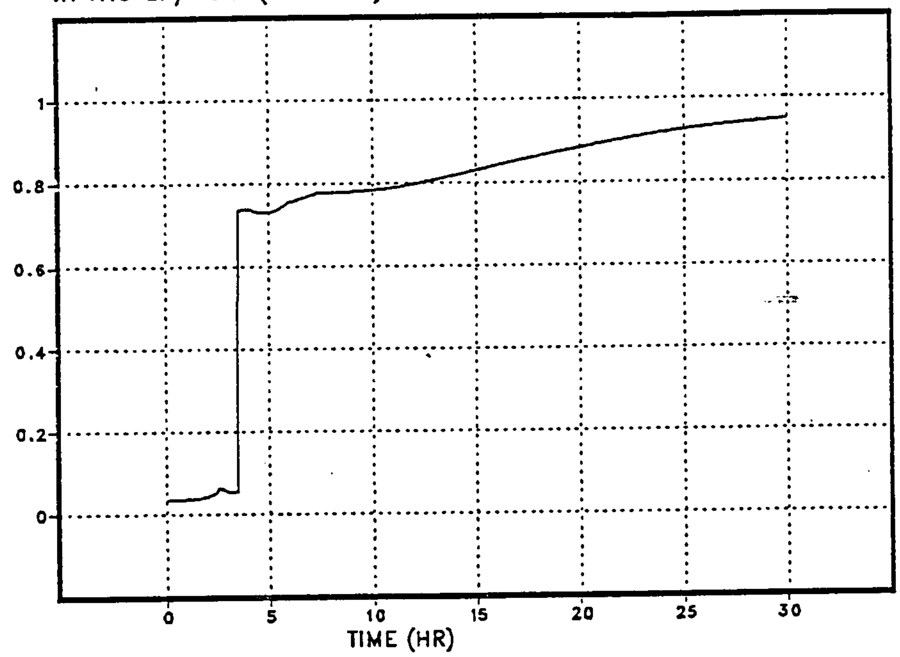

Torus water temerature.
(TWSP)
180
160
140
120
100
80
60
F
0
5
10
15
20
25
30
TIME (HR)

Figure 7.2-34

MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

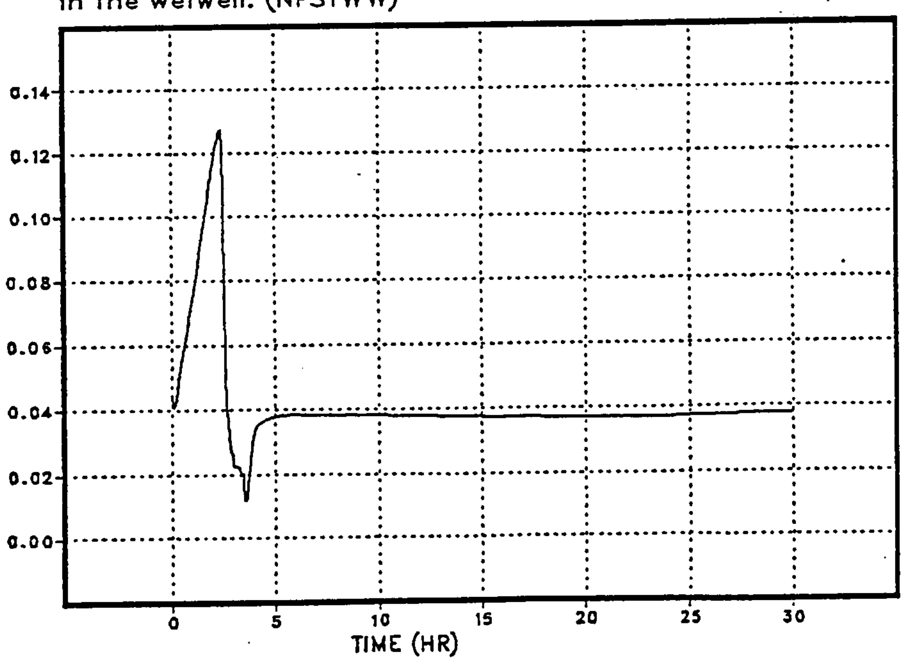
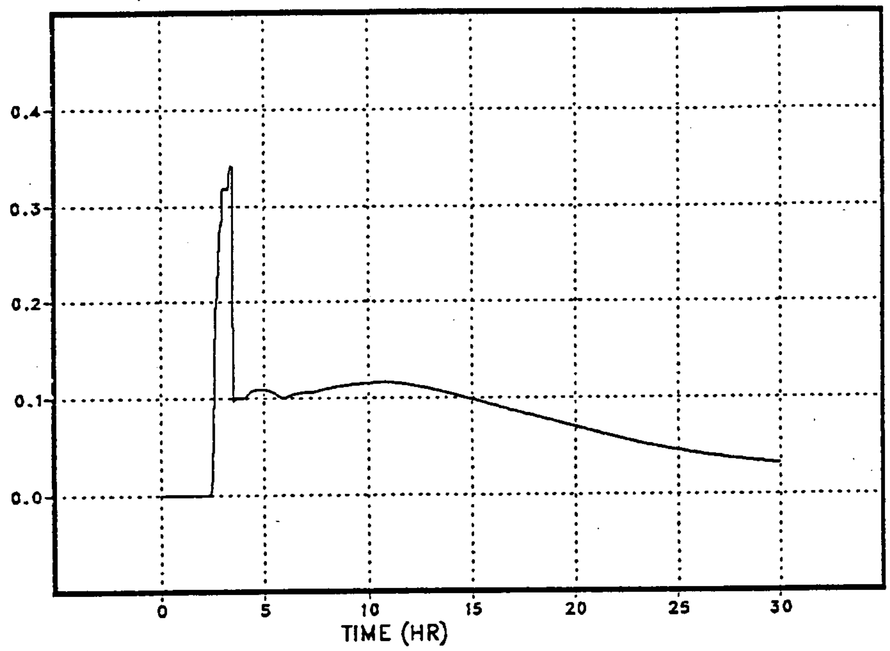


Figure 7.2-35

MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS

Mole fraction of hydrogen in
the drywell atmosphere. (NFH2DW)



Mole fraction of H2
in the wetwell. (NFH2WW)

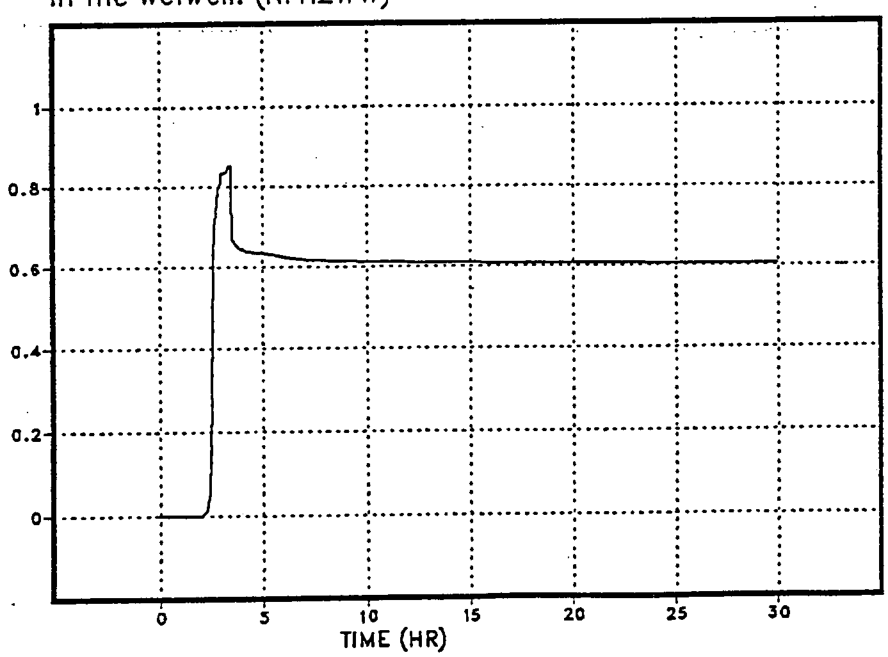
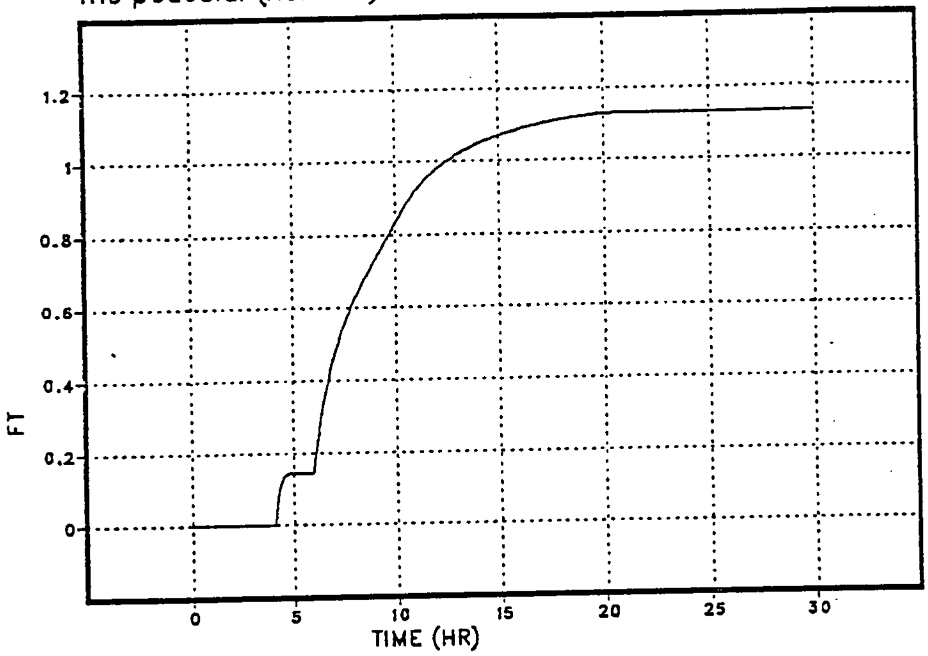


Figure 7.2-36

MAAP MNT SBO W/ 1 SORV
RCIC AVAILABLE ONLY; NO ADS

Concrete ablation thickness in
the pedestal (XCNPDP)



Pressure in the
primary system. (PPS)

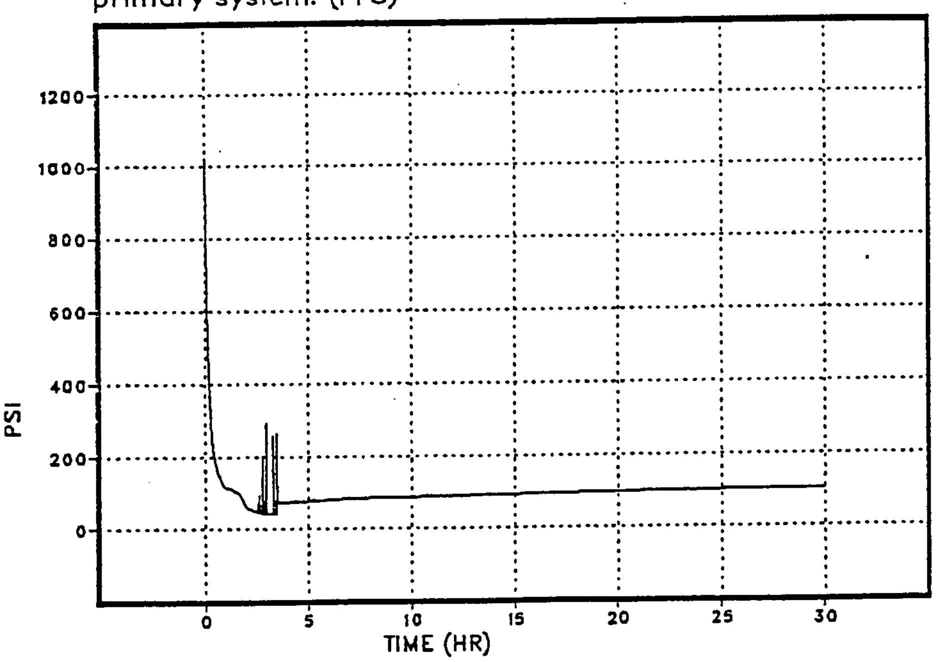
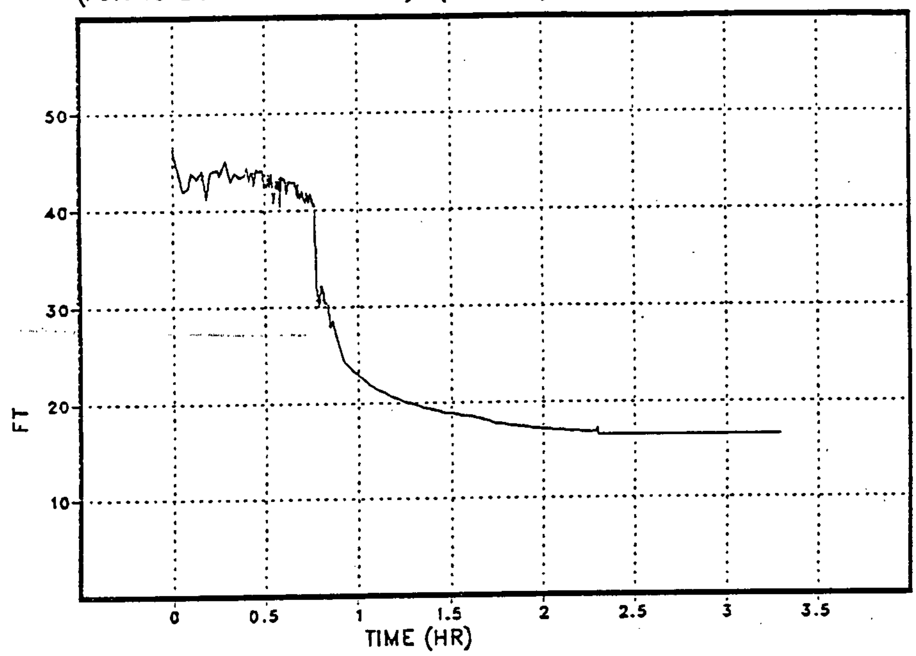


Figure 7.2-37

MNT ATWS: MSIV CLOSE, NO ADS ....
HPCI/RCIC/1 CRD AVAIL 'TIL CONT FAIL

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

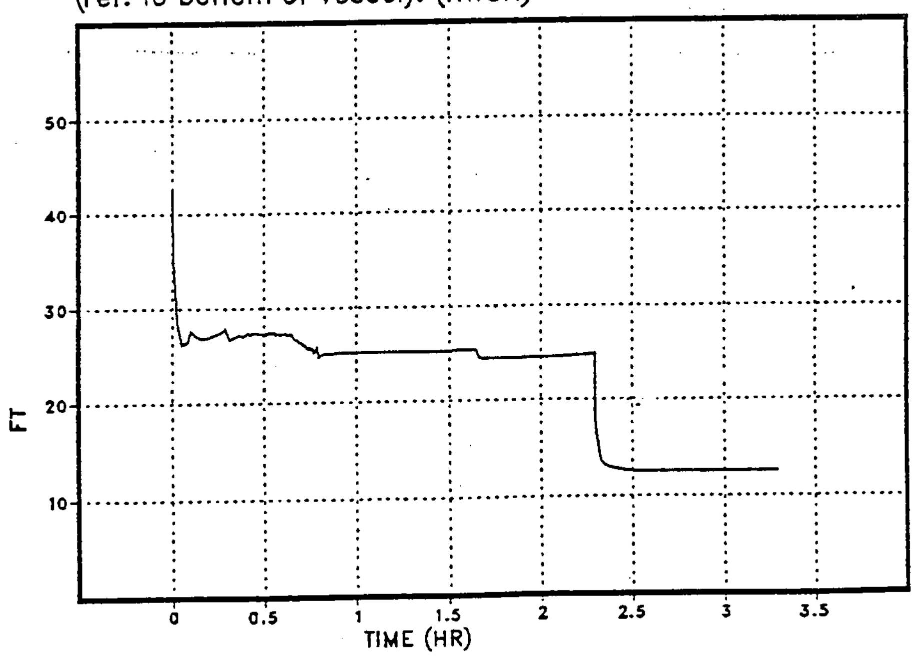
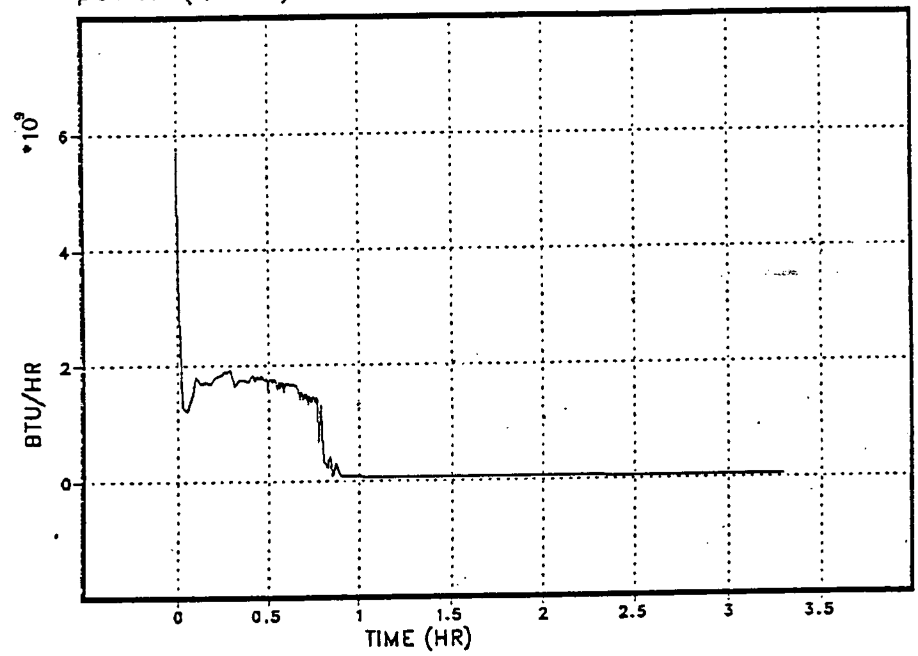


Figure 7.2-38

MNT ATWS: MSIV CLOSE, NO ADS
HPCI/RCIC/1 CRD AVAIL 'TIL CONT FAIL

Total core
power. (QCORE)



Steam flow rate out
of the core. (WSCOR)

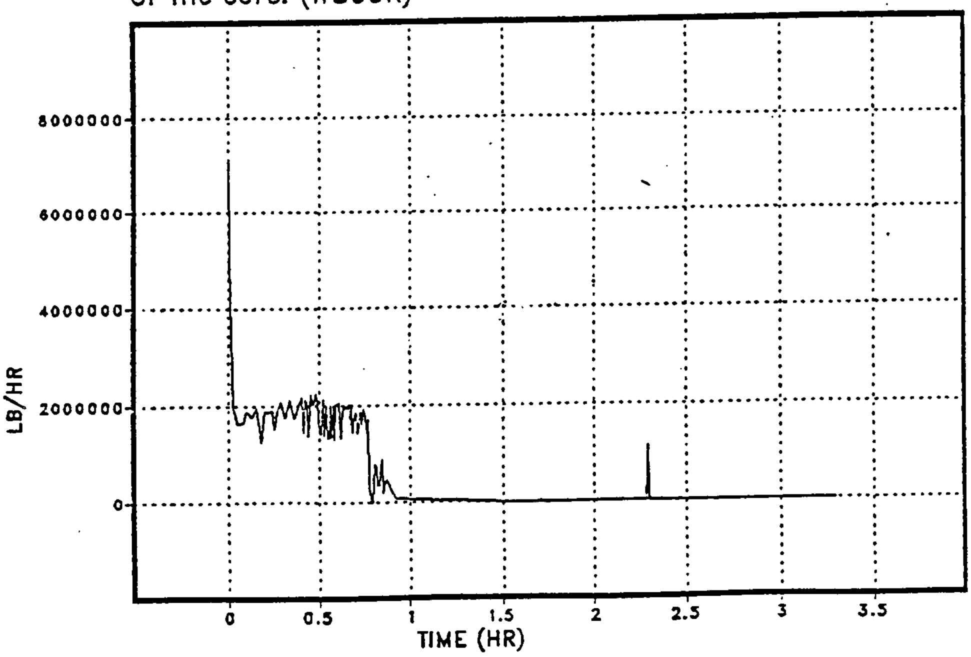
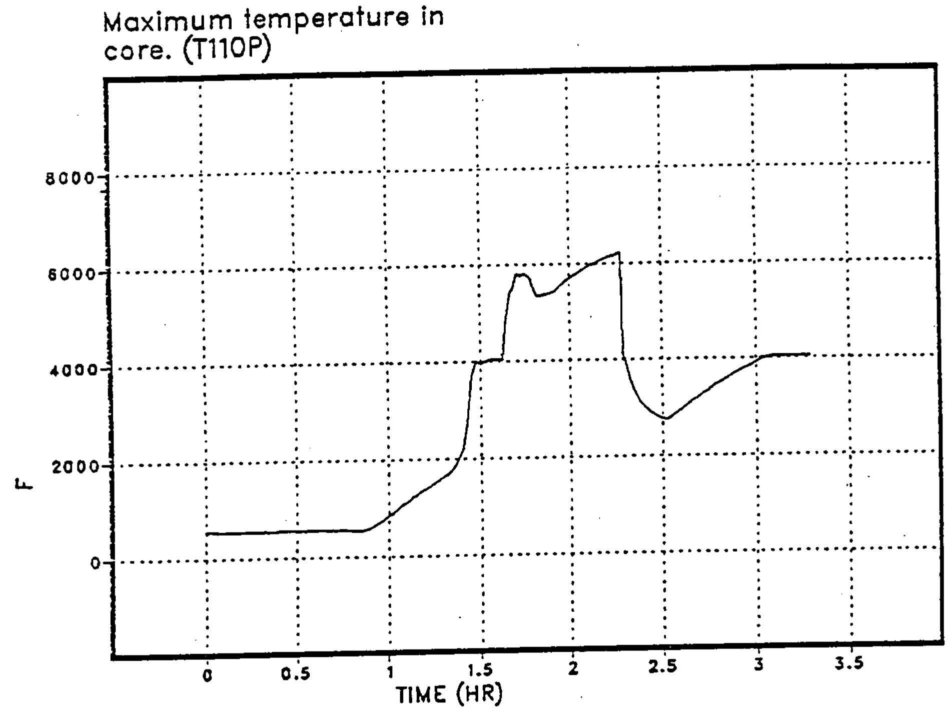


Figure 7.2-39

MNT ATWS: MSIV CLOSE, NO ADS
HPCI/RCIC/1 CRD AVAIL 'TIL CONT FAIL



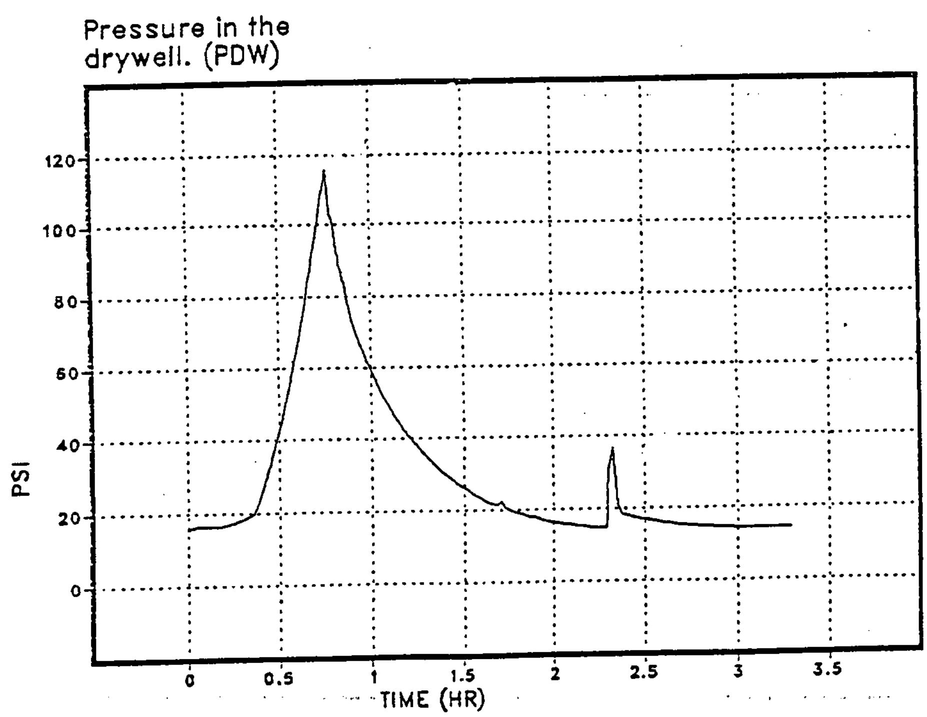


Figure 7.2-40

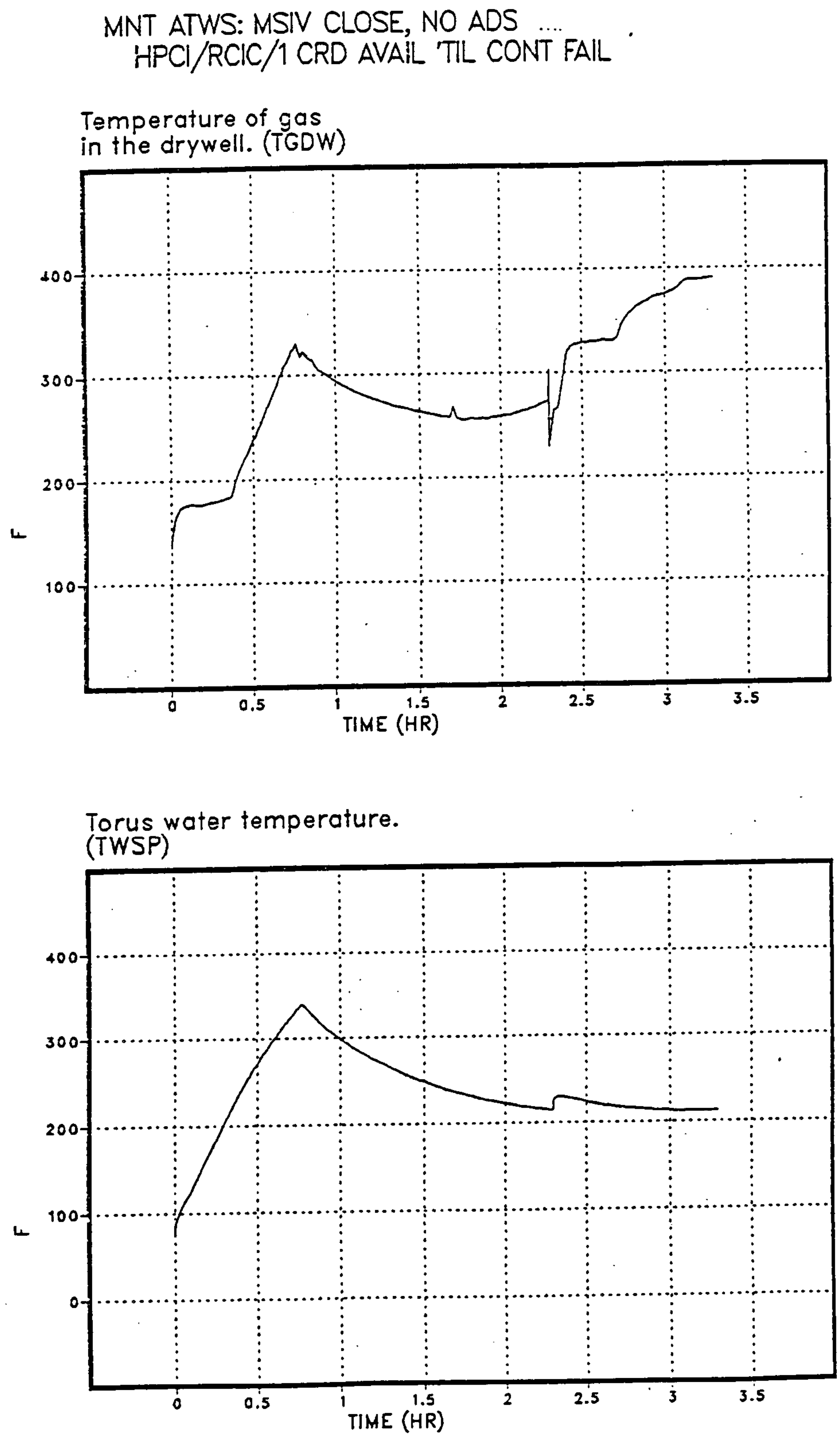
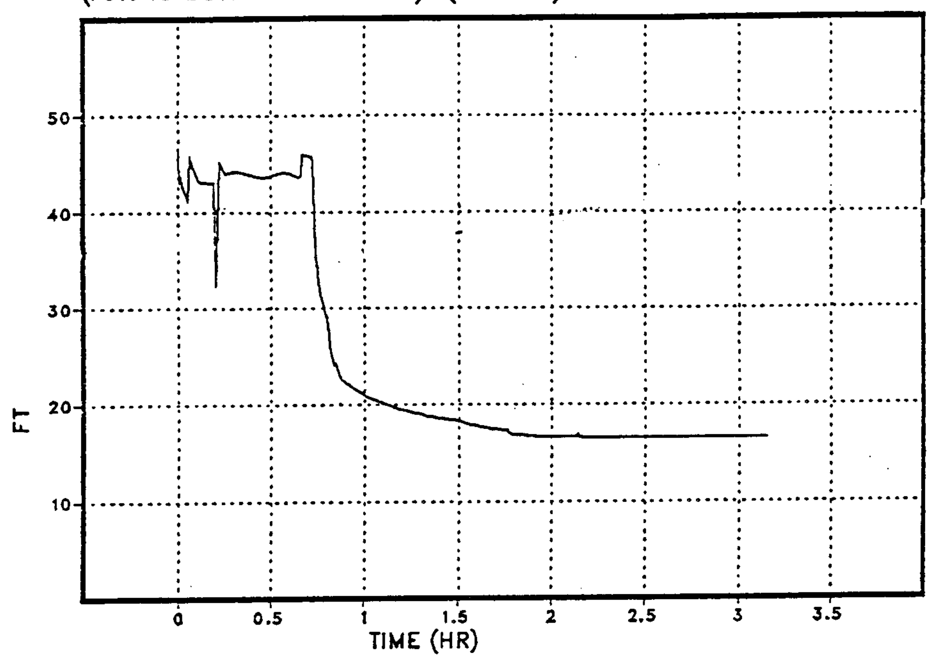

MNT ATWS: MSIV CLOSE, NO ADS ....
HPCI/RCIC/1 CRD AVAIL 'TIL CONT FAIL
Temperature of gas
in the drywell. (TGDW)
400
300
200
100
F
0
0.5
1
1.5
2
2.5
3
3.5
TIME (HR)
Torus water temperature.
(TWSP)
400
300
200
100
0
F
0
0.5
1
1.5
2
2.5
3
3.5
TIME (HR)

Figure 7.2-41

MNT ATWS: MSIV CLOSE, NO ADS
11 MIN FW; HPCI/RCIC/1 CRD TO CONT FAIL

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

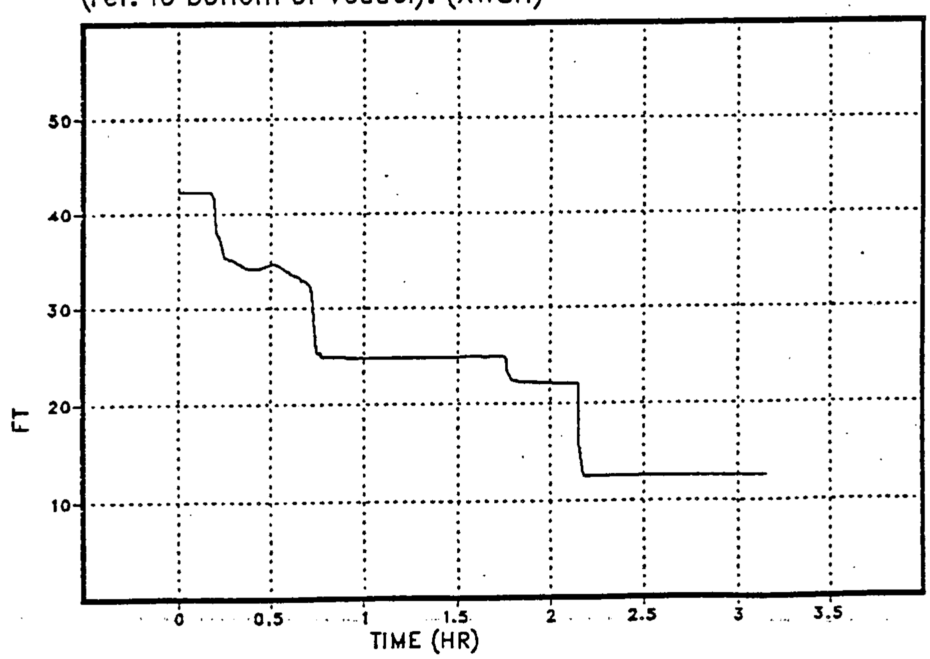
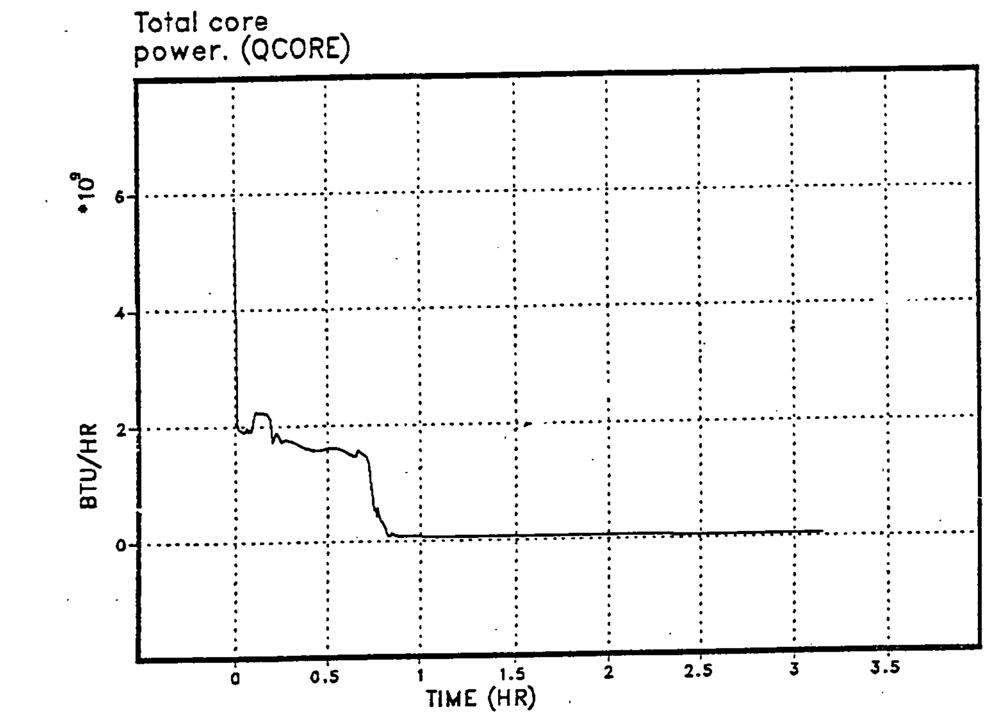


Figure 7.2-42

MNT ATWS: MSIV CLOSE, NO ADS
11 MIN FW; HPCI/RCIC/1 CRD TO CONT FAIL



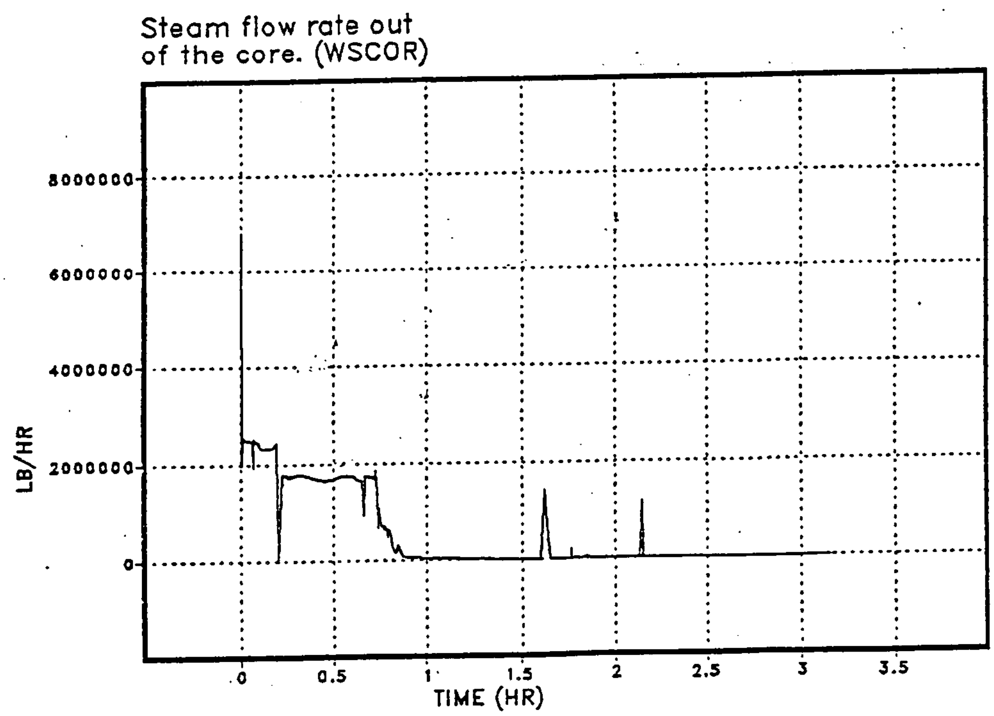
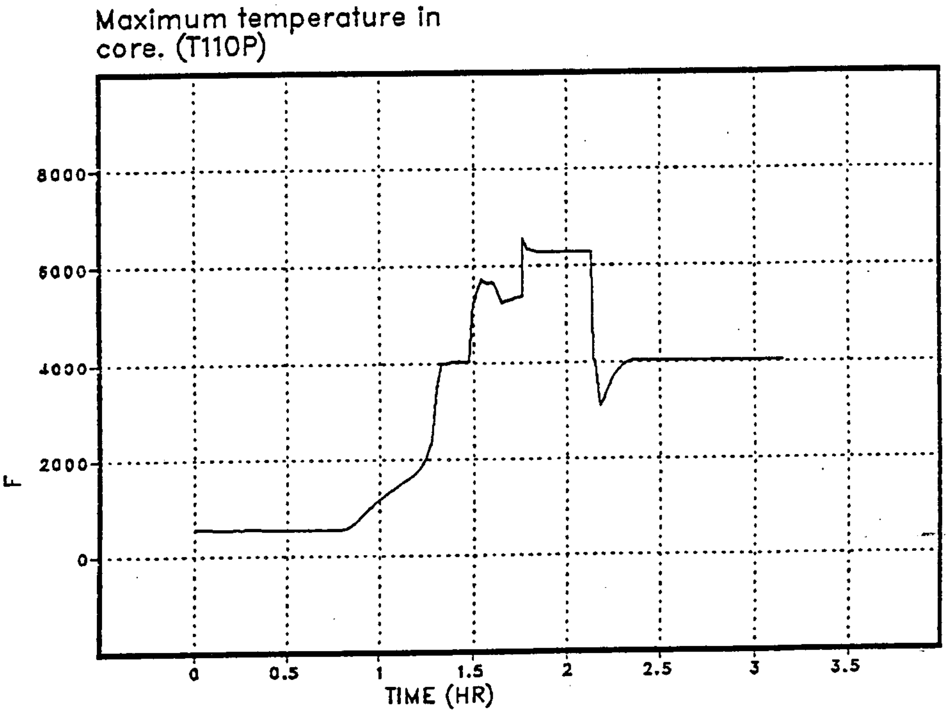


Figure 7.2-43

MNT ATWS: MSIV CLOSE, NO ADS
11 MIN FW; HPCI/RCIC/1 CRD TO CONT FAIL



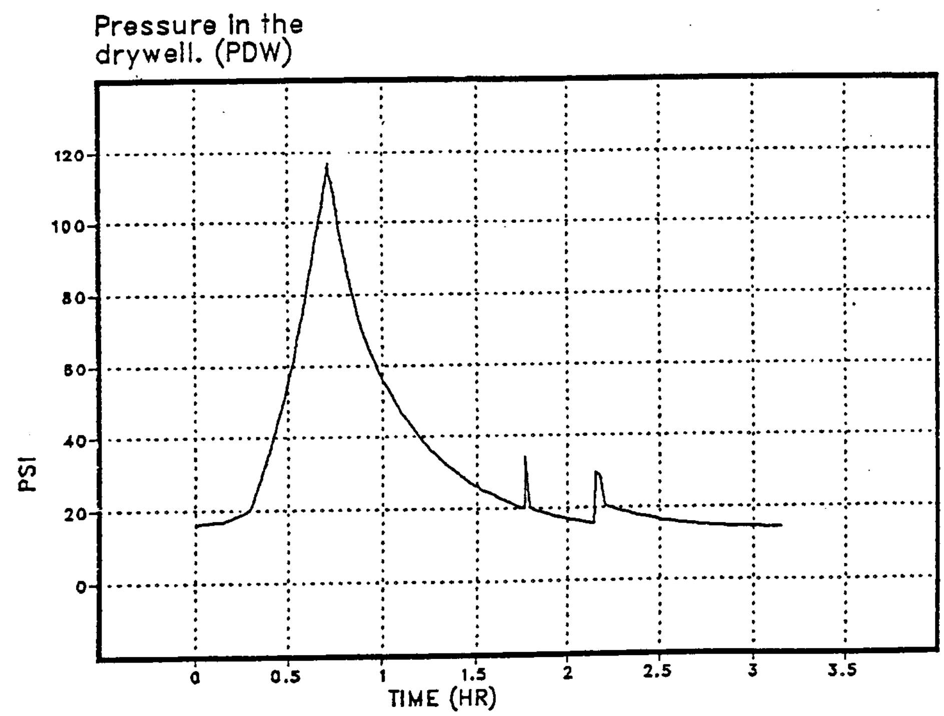
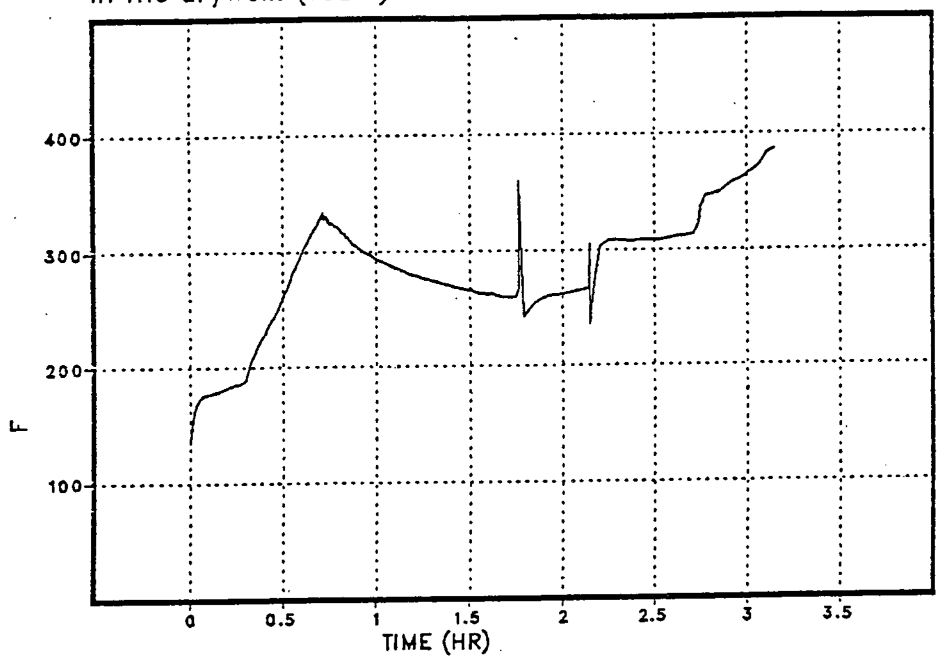


Figure 7.2-44

MNT ATWS: MSIV CLOSE, NO ADS
11 MIN FW; HPCI/RCIC/1 CRD TO CONT FAIL

Temperature of gas
in the drywell. (TGDW)



Torus water temerature.
(TWSP)

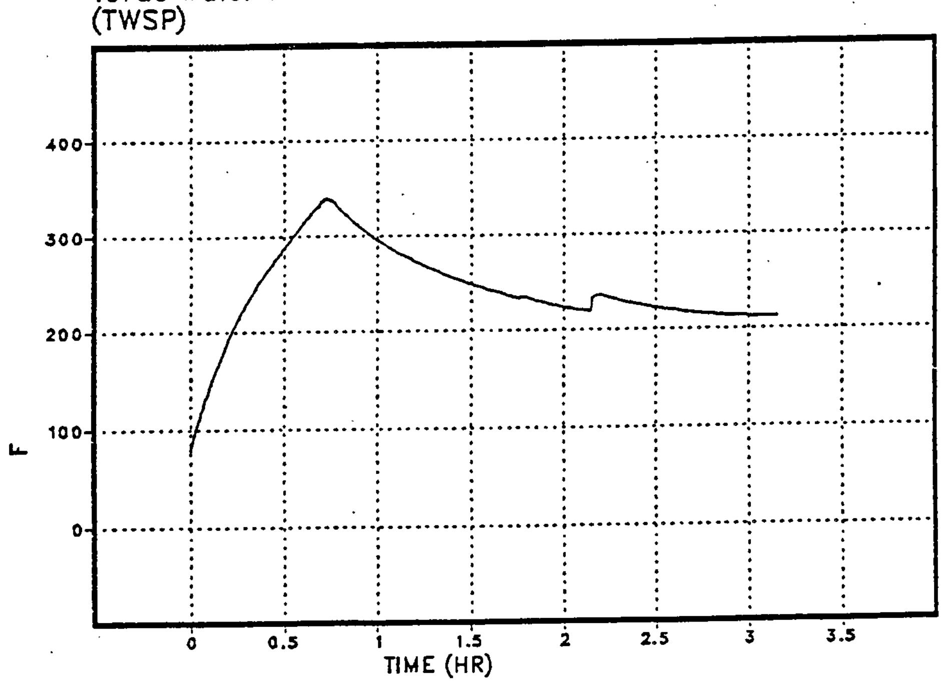
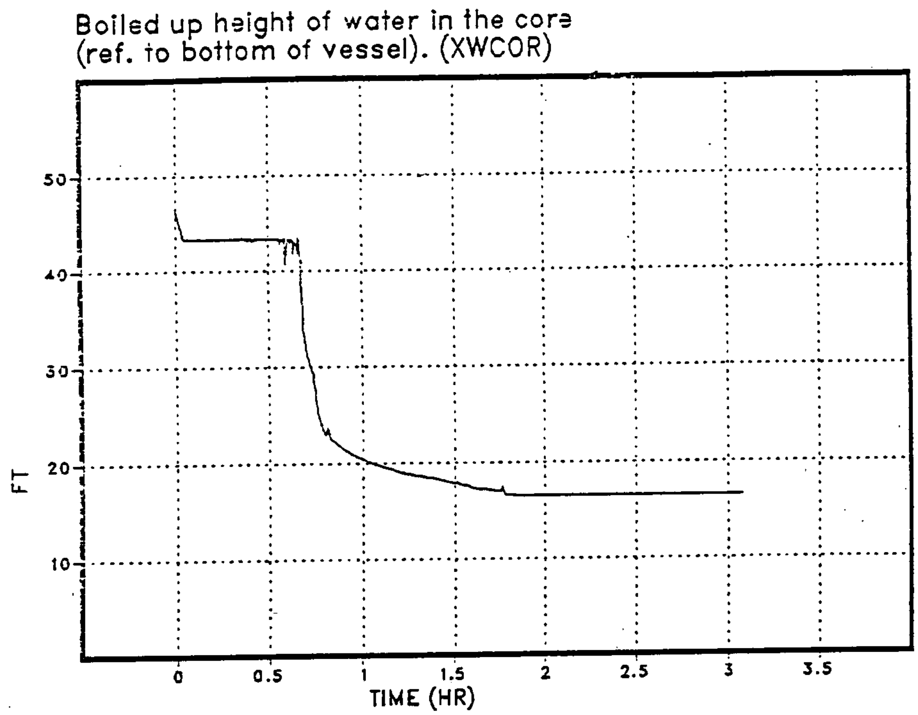


Figure 7.2-45
ATWS: Turbine Trip with Bypass
Maintain Reactor Level at +48"



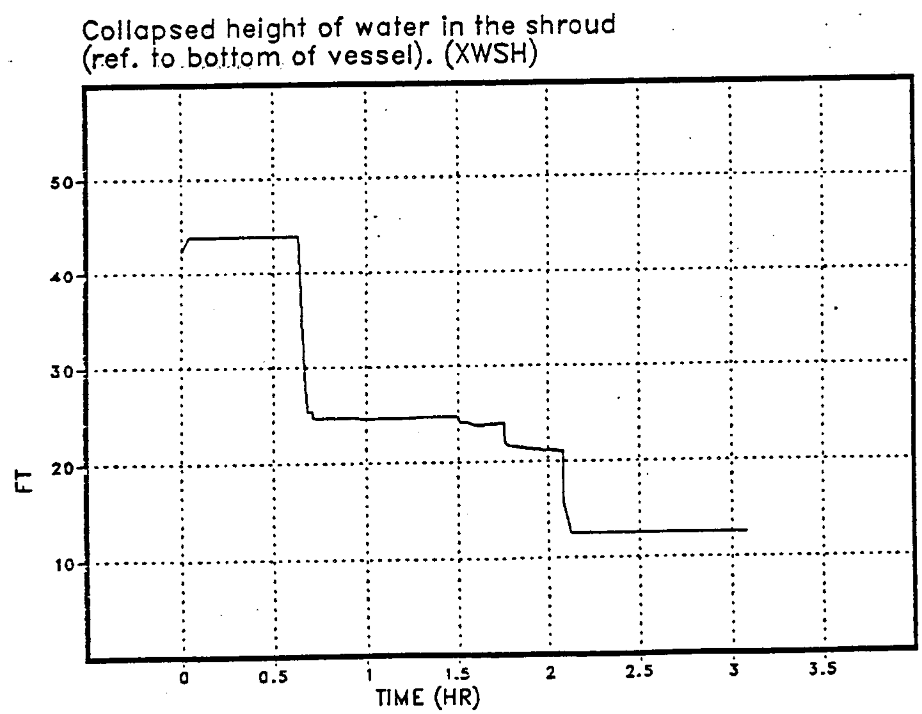
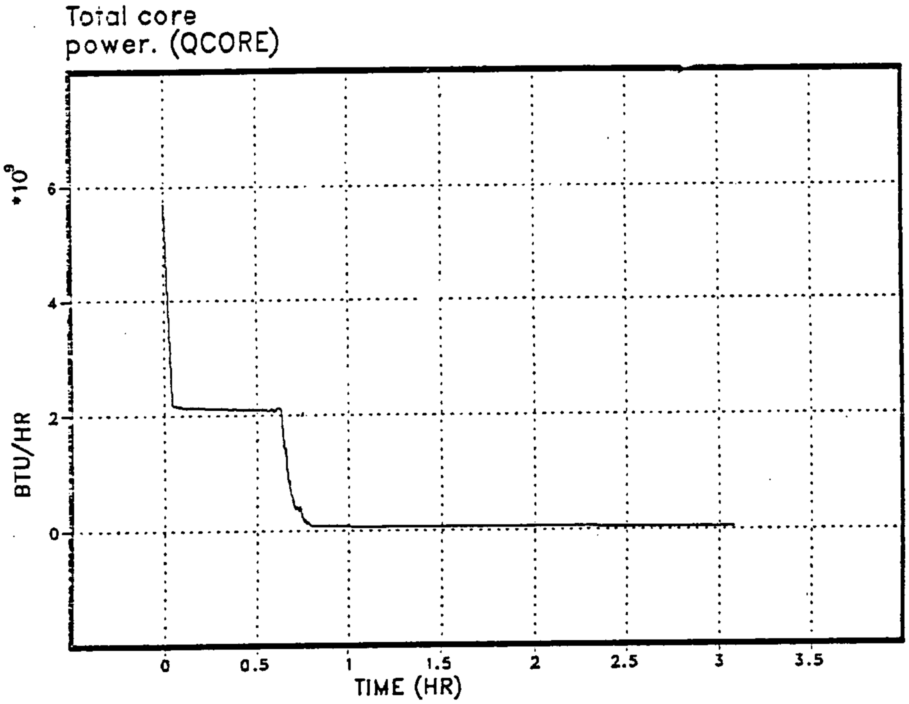


Figure 7.2-46
ATWS: Turbine Trip with Bypass
Maintain Reactor Level at +48"



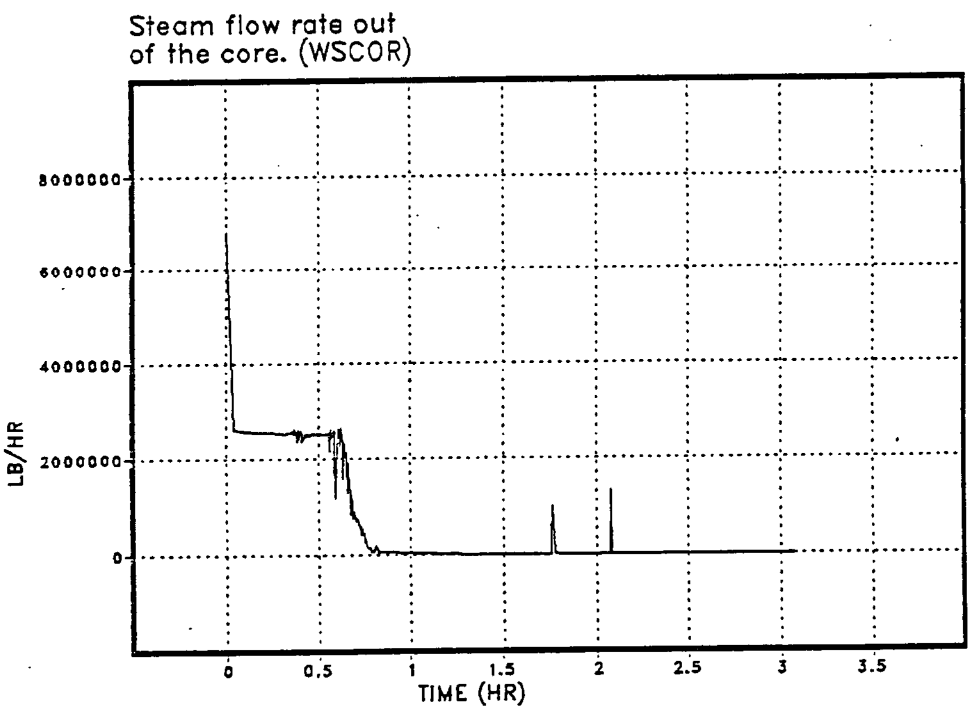
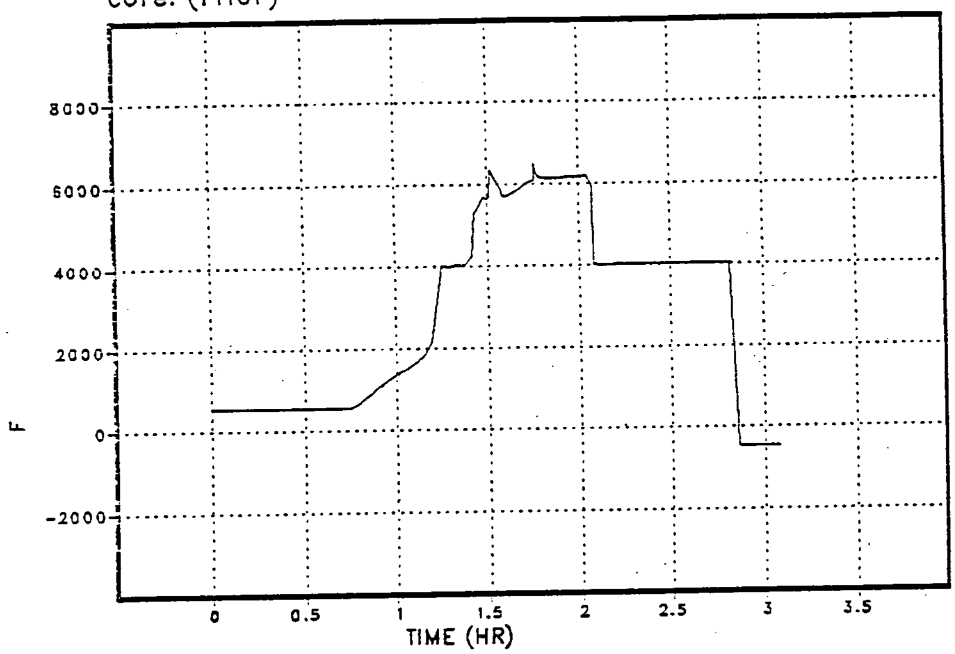


Figure 7.2-47
ATWS: Turbine Trip with Bypass
Maintain Reactor Level at +48"

Maximum temperature in core. (T110P)



Pressure in the drywell. (PDW)

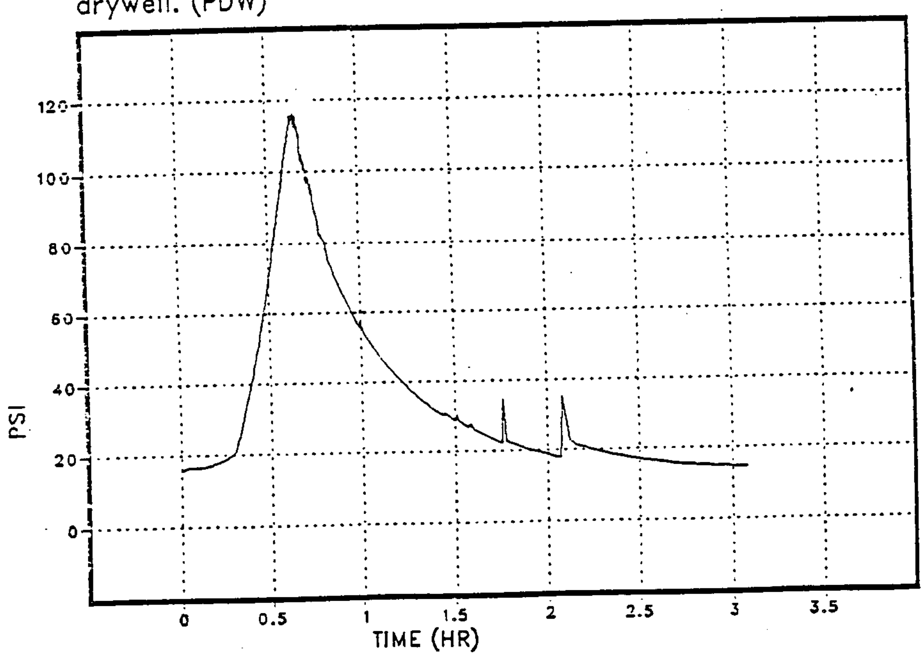
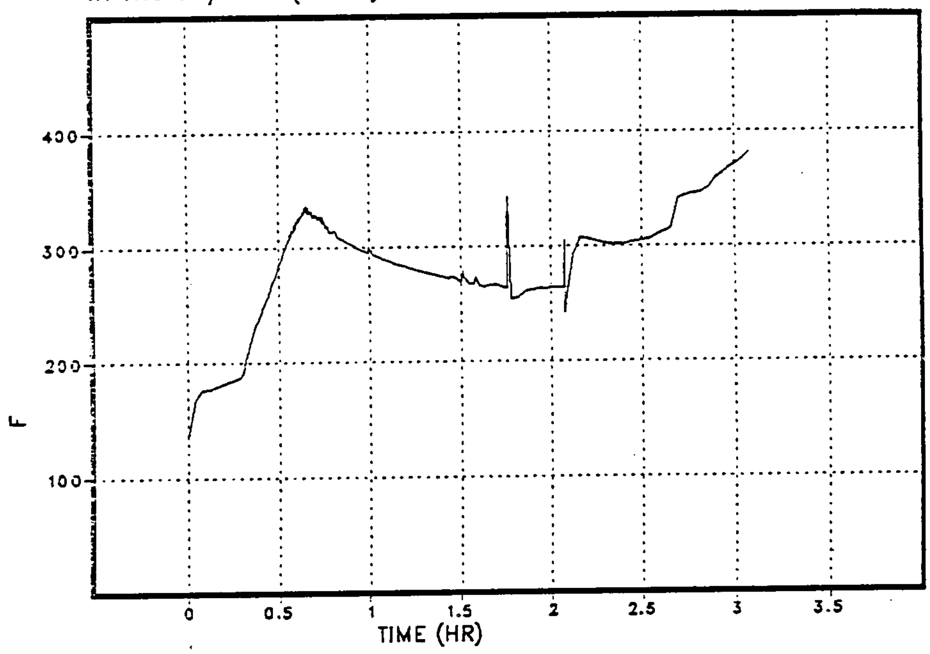


Figure 7.2-48
ATWS: Turbine Trip with Bypass
Maintain Reactor Level at +48"

Temperature of gas in the drywell. (TGDW)



Torus water temperature. (TWSP)

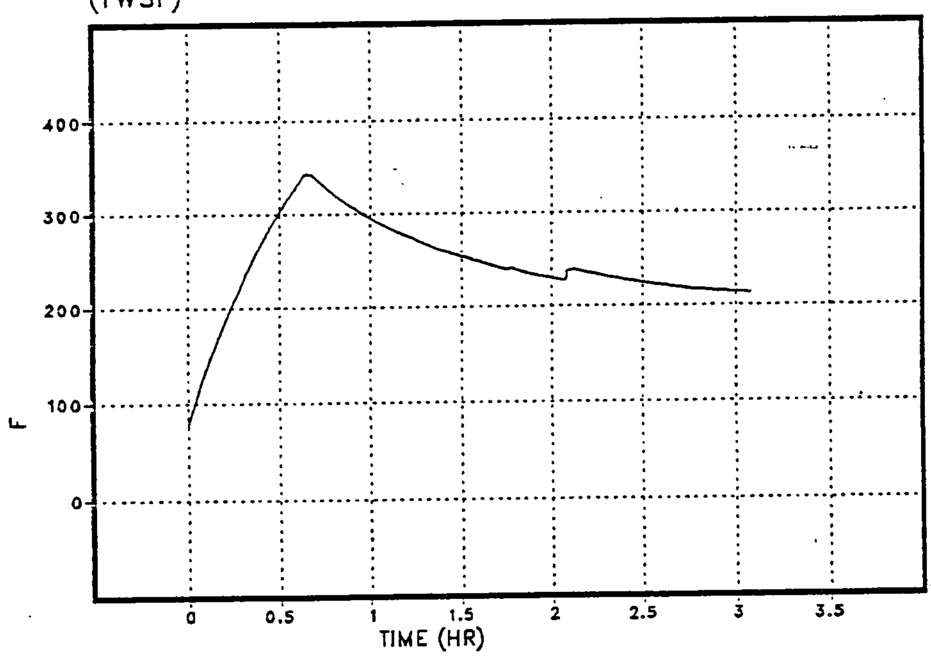
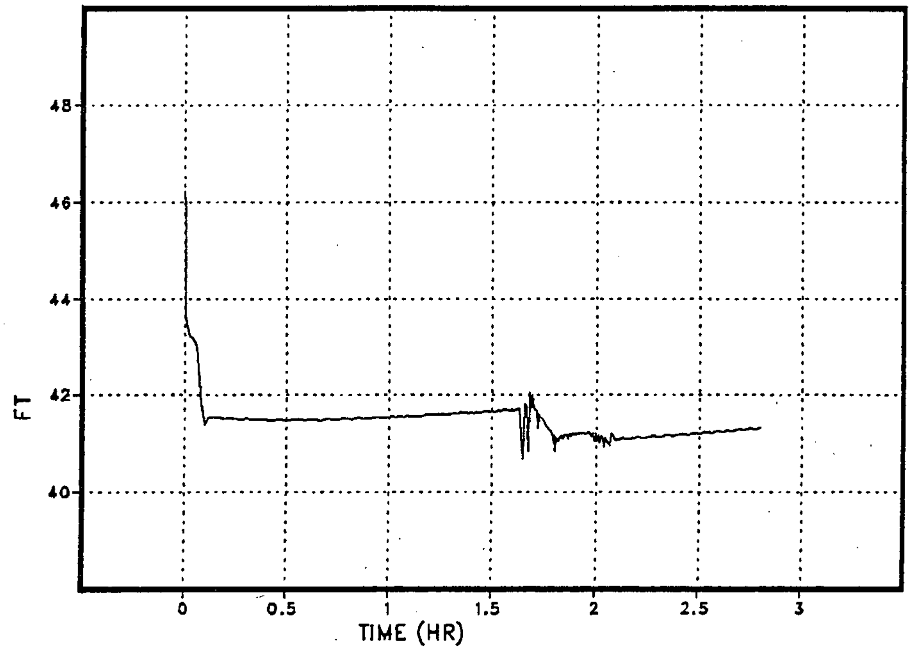


Figure 7.2-49

MAAP MNT ATWS: TT W/ BP; FW
AVAIL ONLY; THROTTLE FW TO TAF (RCIC)

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

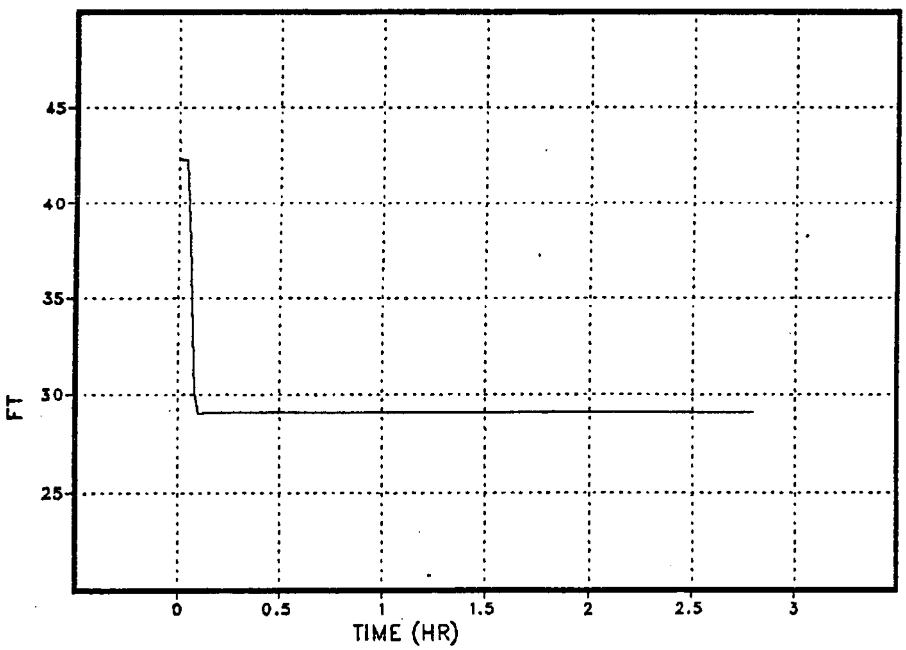
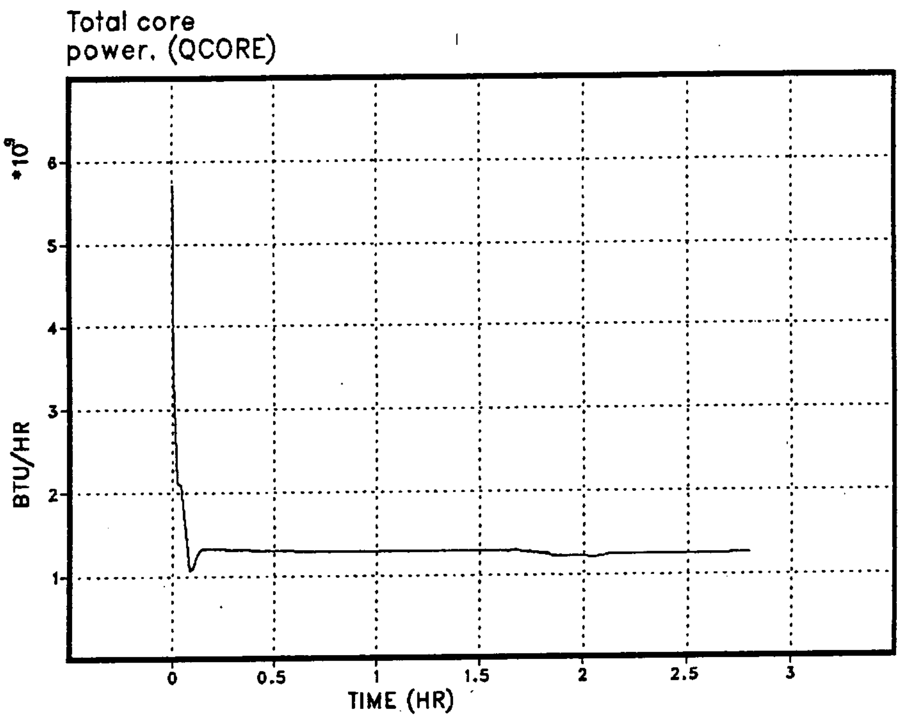


Figure 7.2-50

MAAP MNT ATWS: TT W/ BP; FW
AVAIL ONLY; THROTTLE FW TO TAF (RCIC)



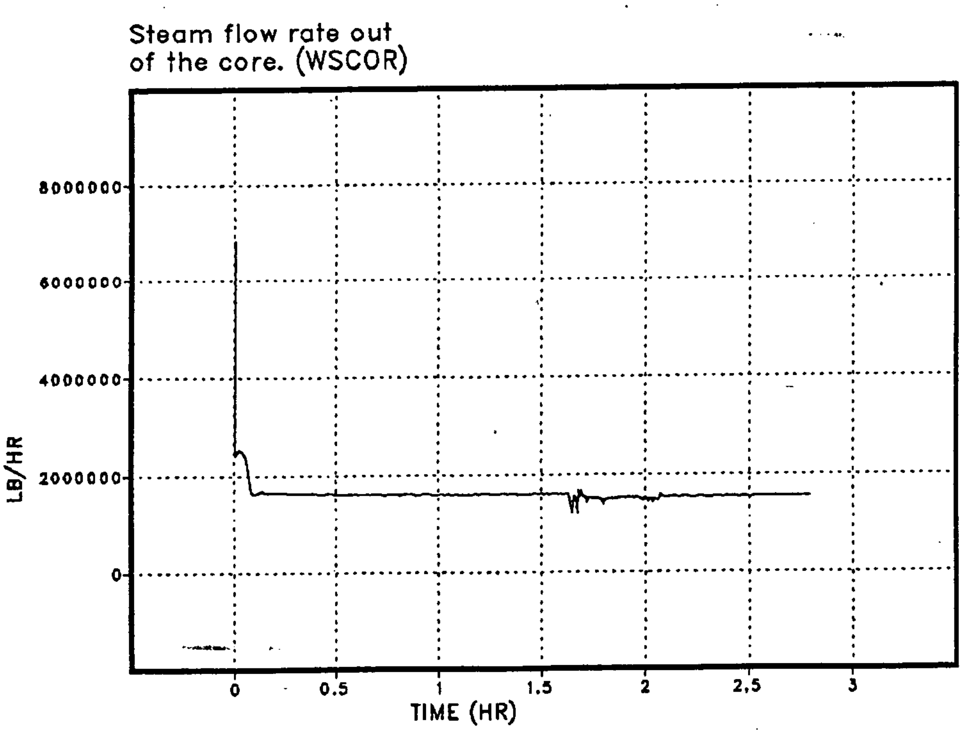
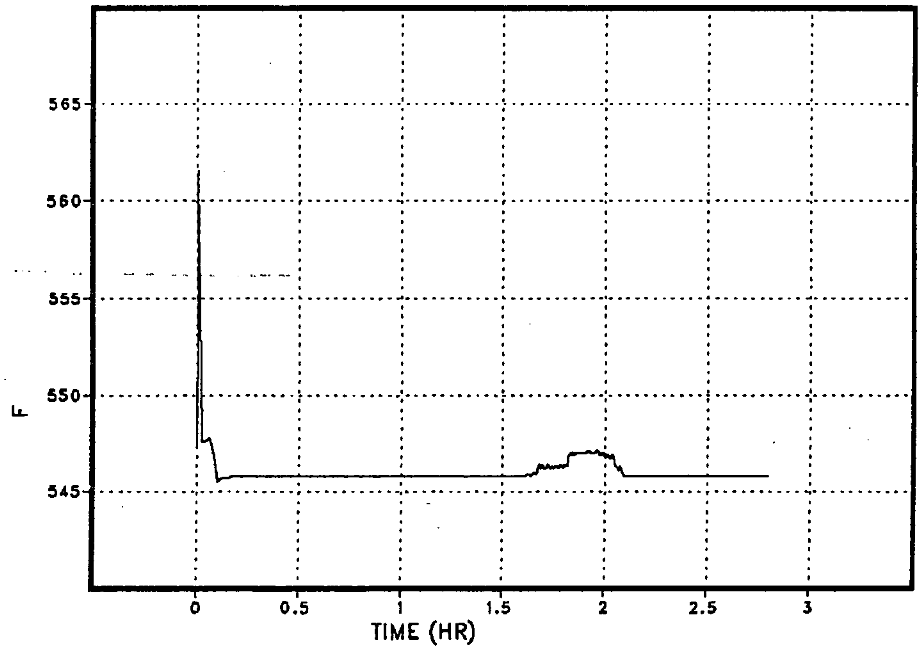


Figure 7.2-51

MAAP MNT ATWS: TT W/ BP; FW
AVAIL ONLY; THROTTLE FW TO TAF (RCIC)

Maximum temperature in
core. (T110P)



Pressure in the
drywell. (PDW)

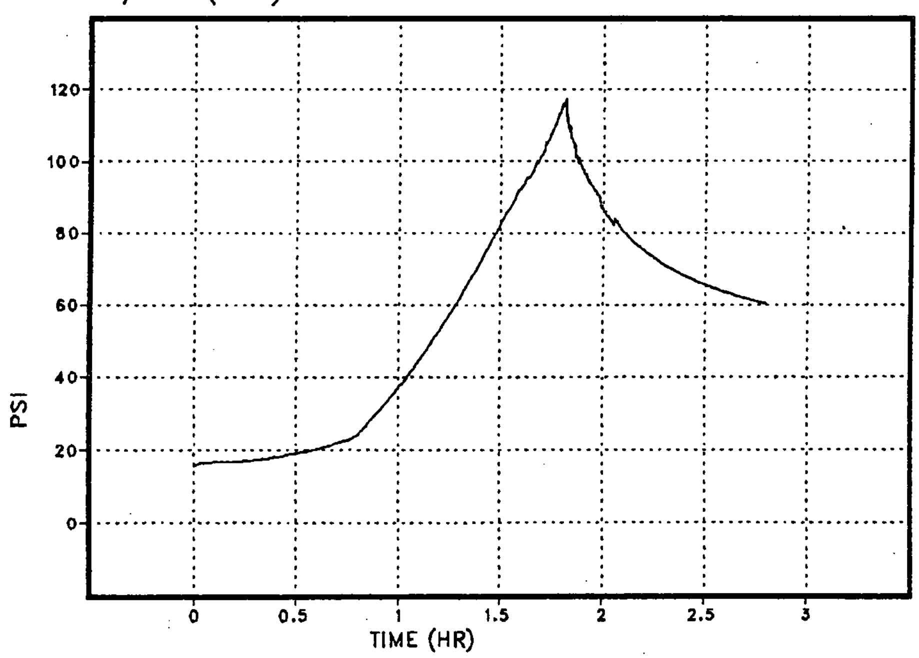
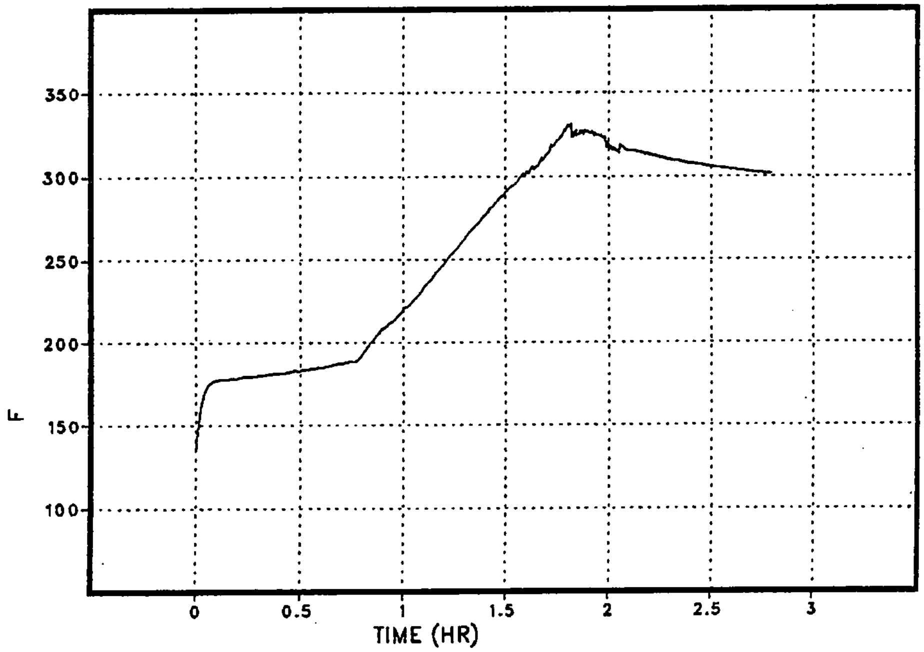


Figure 7.2-52

MAAP MNT ATWS: TT W/ BP; FW
AVAIL ONLY; THROTTLE FW TO TAF (RCIC)

Temperature of gas
in the drywell. (TGDW)



Torus water temperature.
(TWSP)

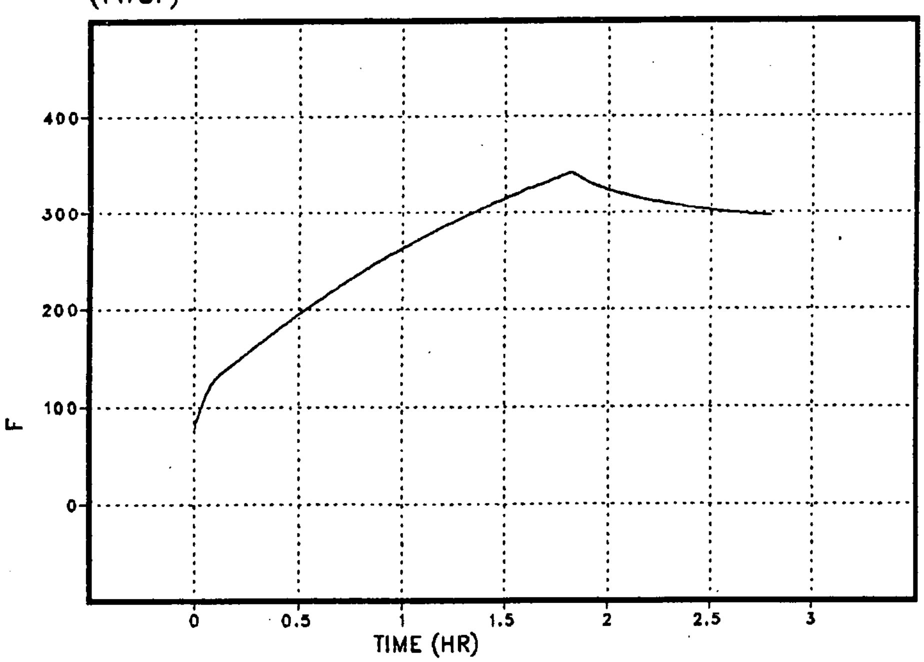
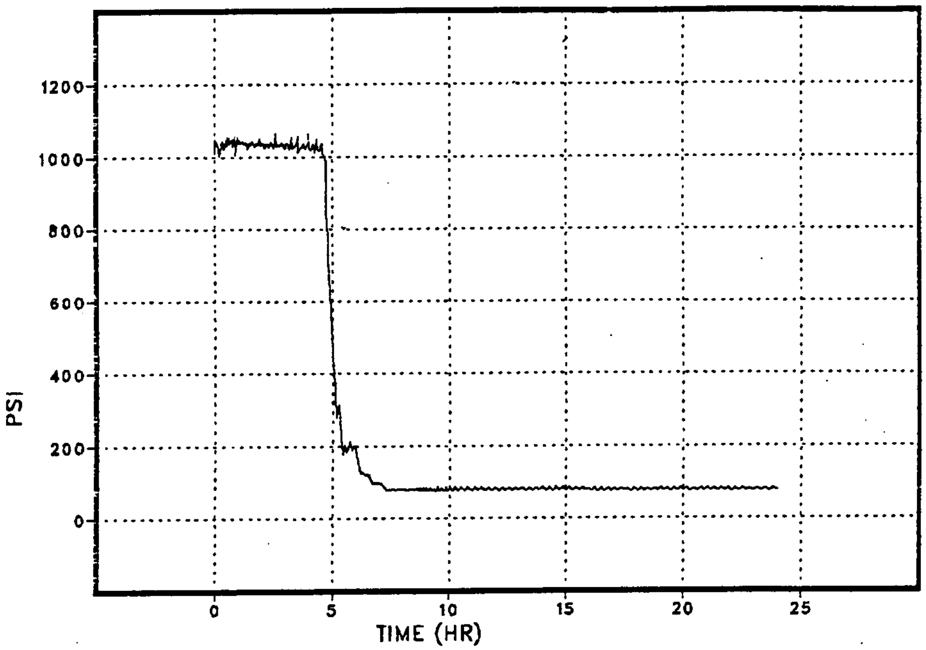


Figure 7.2-53

TW MSIV CLOSE; INJECT FROM ....
TORUS PER DETAILED DESCRIPTION; HCTL

Pressure in the
primary system. (PPS)



Maximum temperature in
core. (T110P)

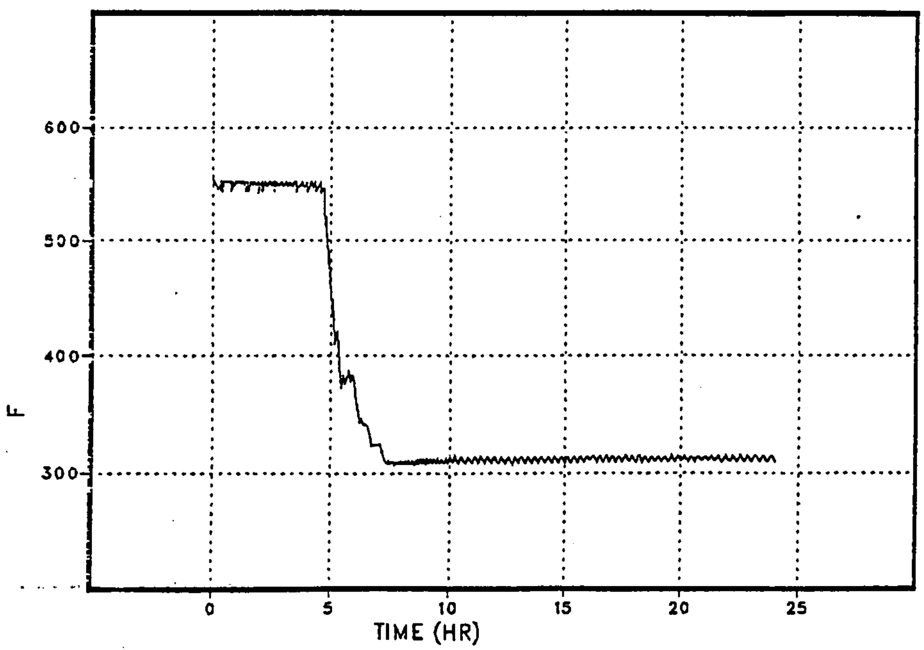
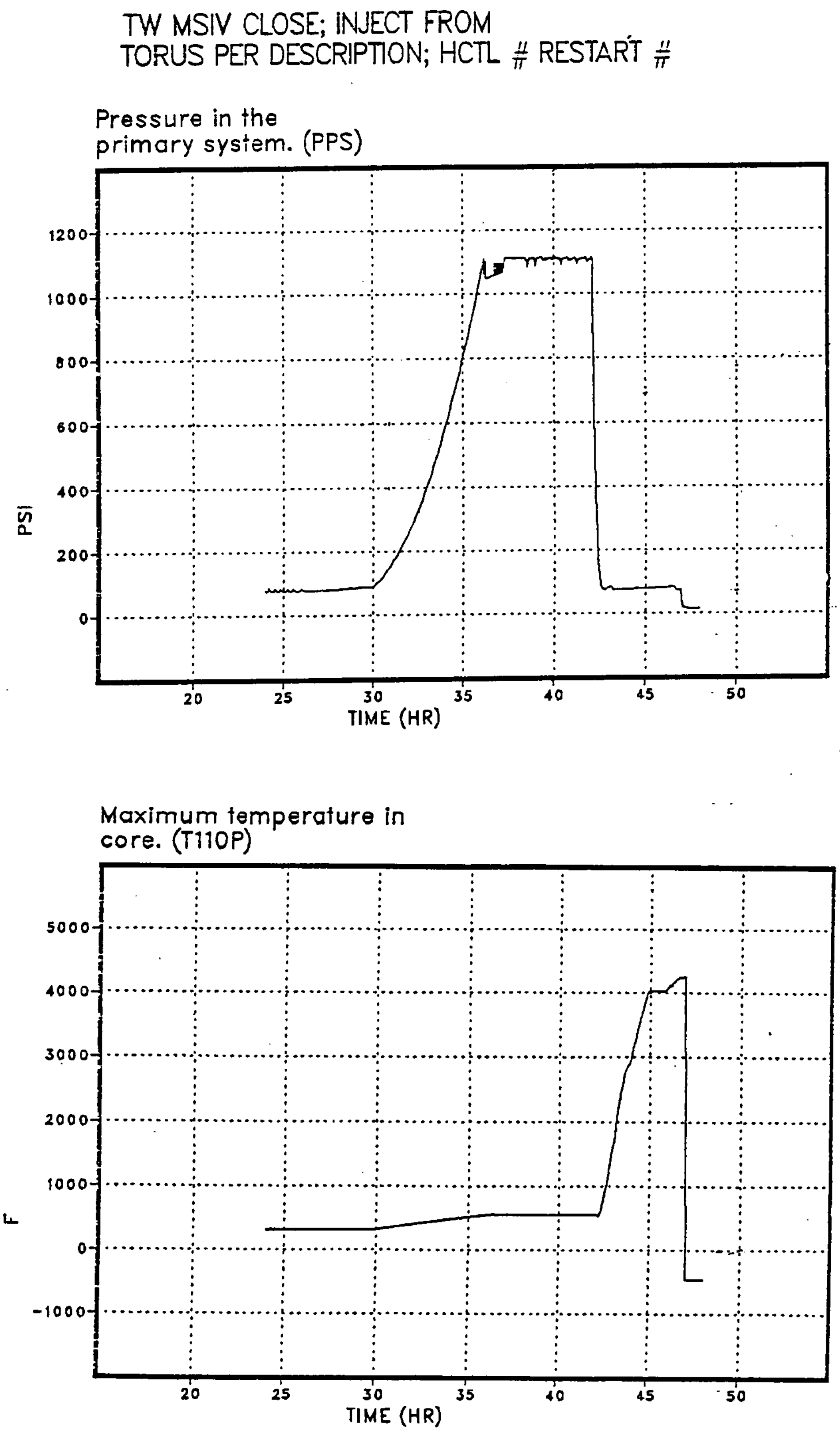


Figure 7.2-54

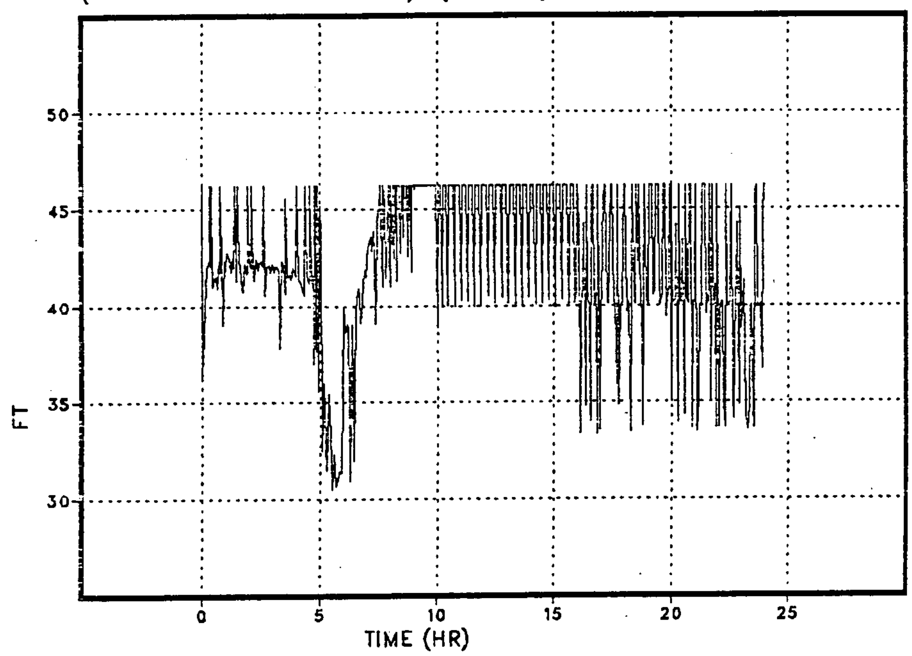


TW MSIV CLOSE; INJECT FROM
TORUS PER DESCRIPTION; HCTL # RESTART #
Pressure in the
primary system. (PPS)
PSI
1200
1000
800
600
400
200
0
20
25
30
35
40
45
50
TIME (HR)
Maximum temperature in
core. (T110P)
F
5000
4000
3000
2000
1000
0
-1000
20
25
30
35
40
45
50
TIME (HR)

Figure 7.2-55

TW MSIV CLOSE; INJECT FROM
TORUS PER DETAILED DESCRIPTION; HCTL

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

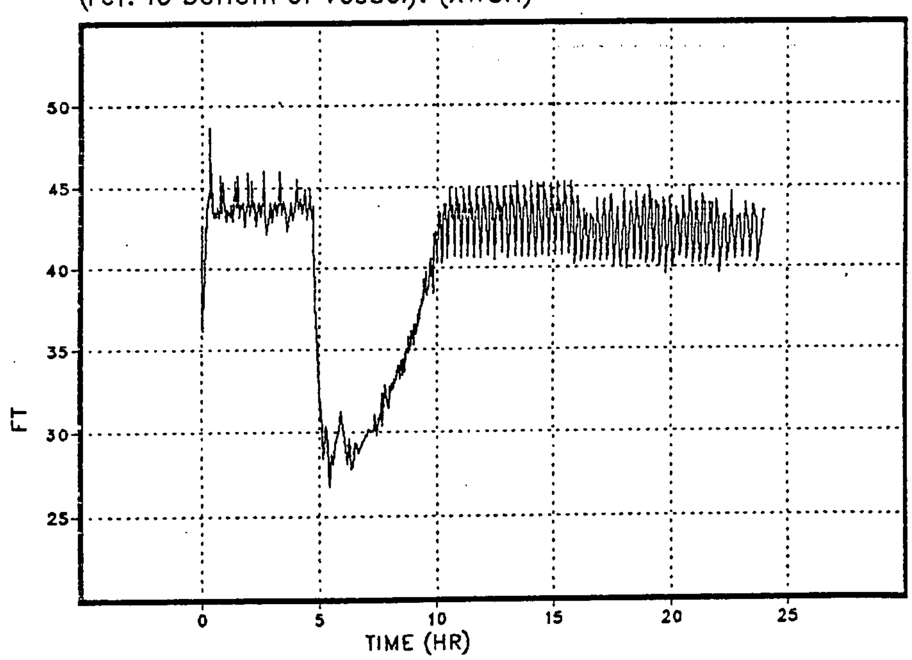
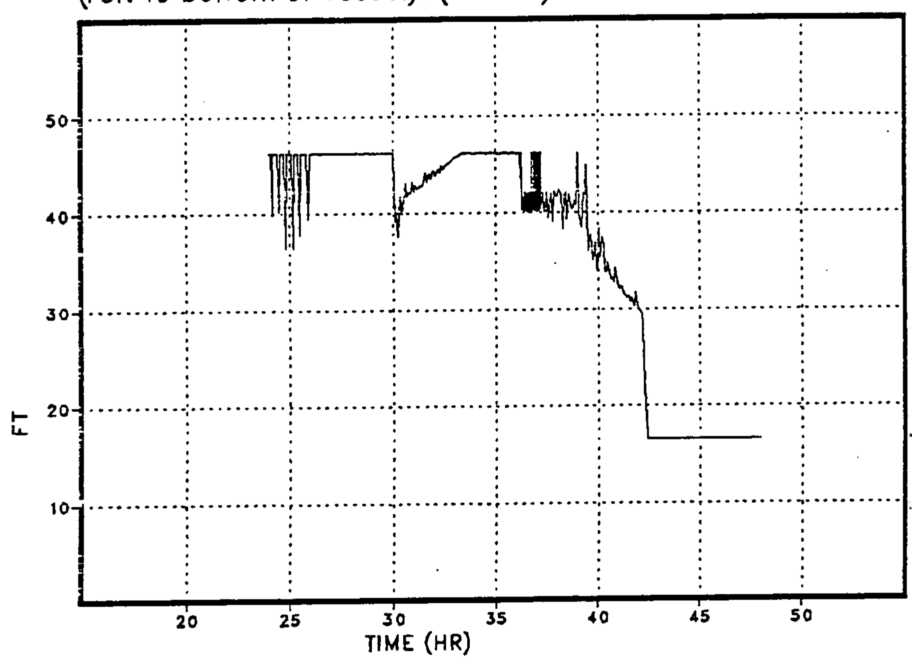


Figure 7.2-56

TW MSIV CLOSE; INJECT FROM
TORUS PER DESCRIPTION; HCTL # RESTART #

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

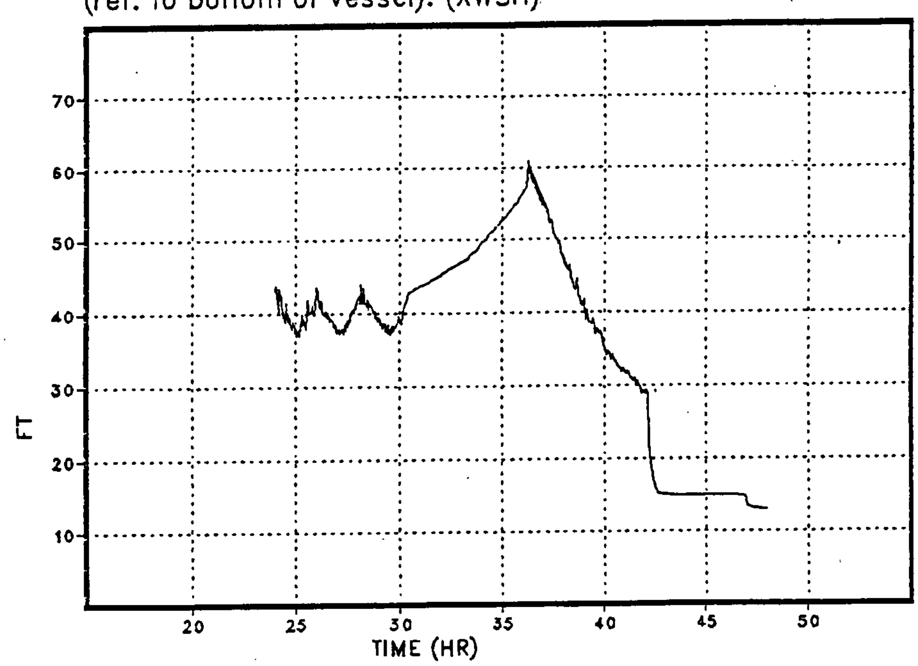
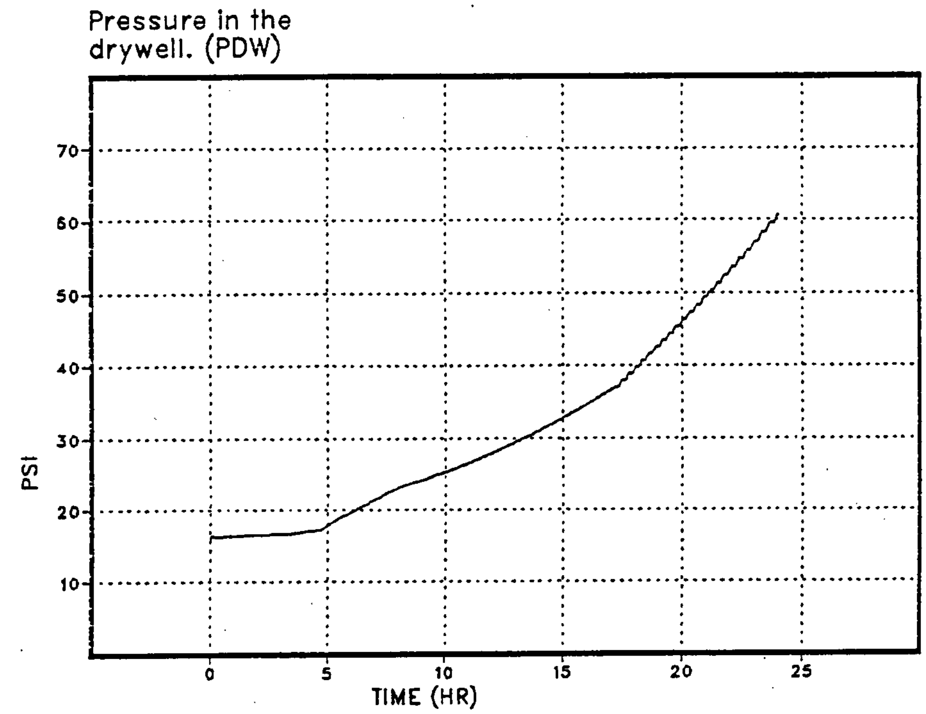


Figure 7.2-57

TW MSIV CLOSE; INJECT FROM
TORUS PER DETAILED DESCRIPTION; HCTL



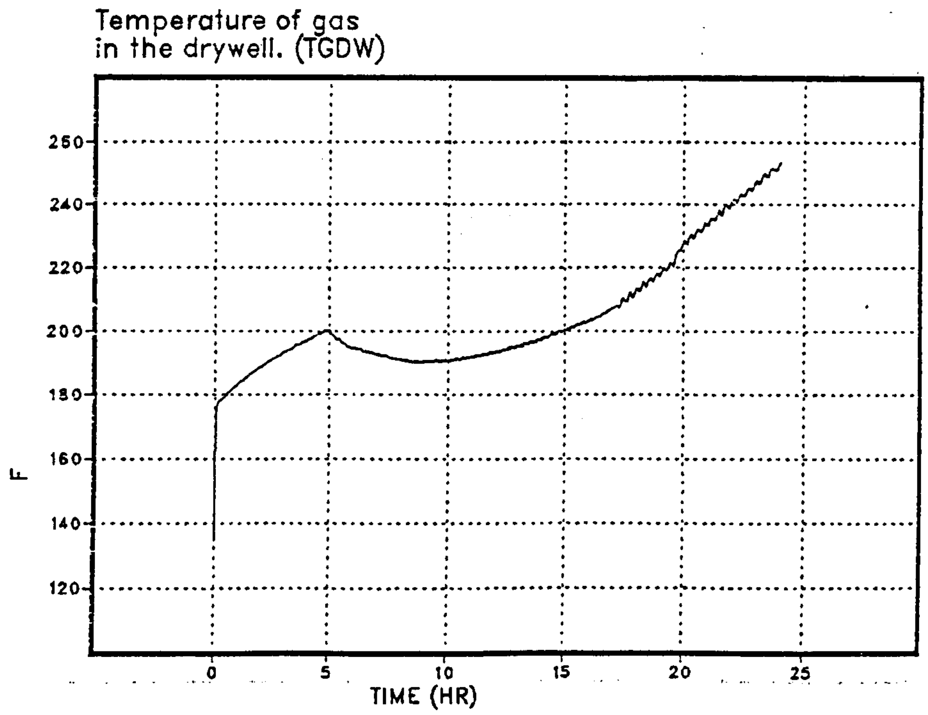
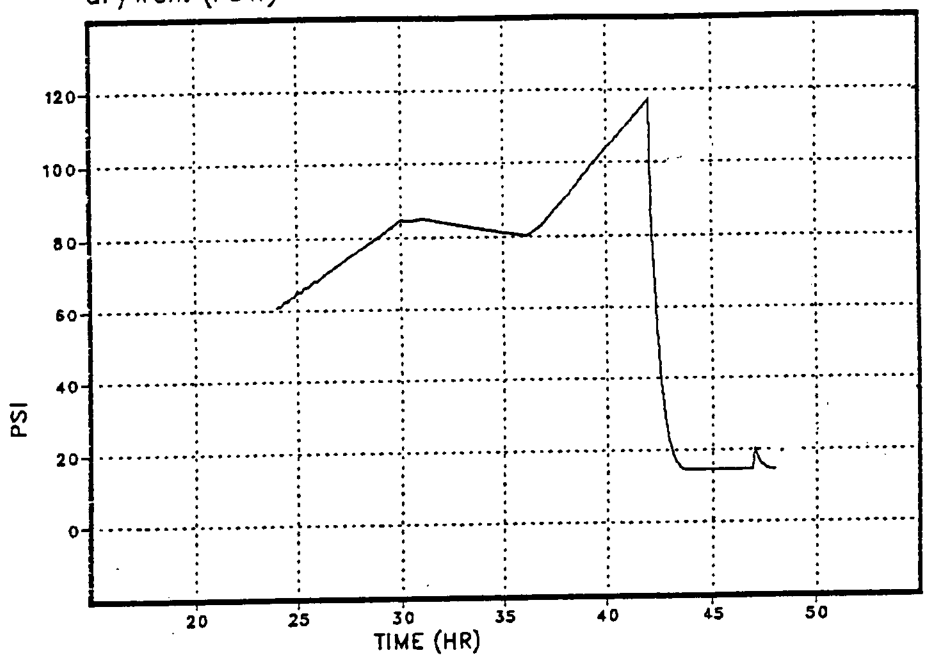


Figure 7.2-58

TW MSIV CLOSE; INJECT FROM
TORUS PER DESCRIPTION; HCTL # RESTART #

Pressure in the
drywell. (PDW)



Temperature of gas
in the drywell. (TGDW)

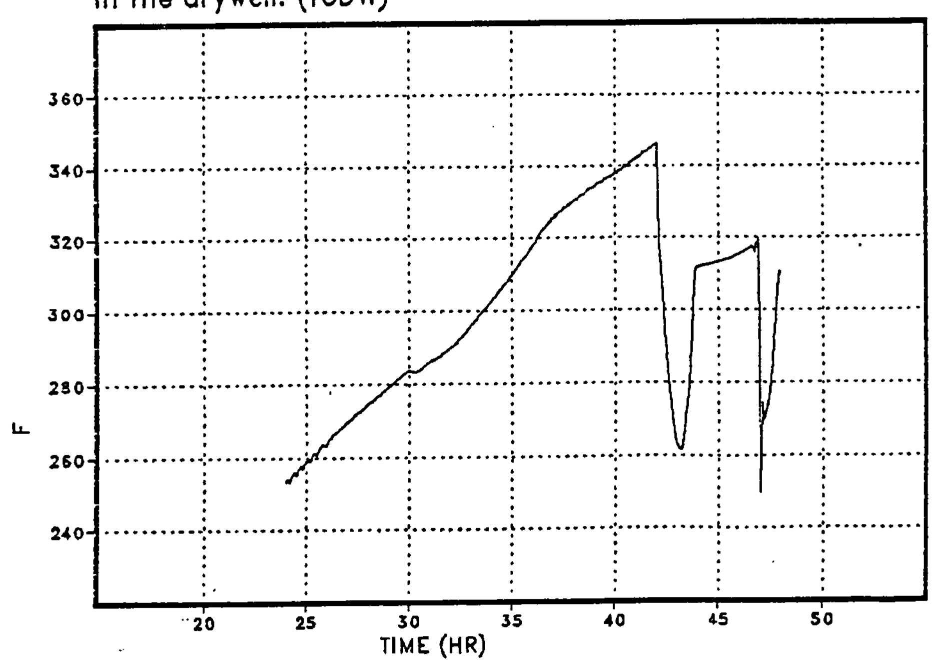
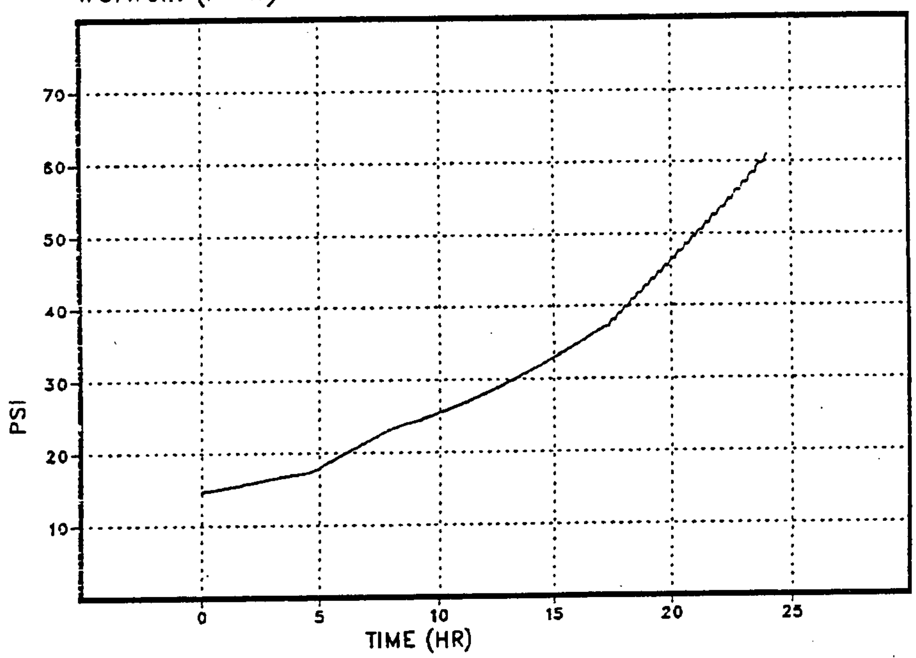


Figure 7.2-59

TW MSIV CLOSE; INJECT FROM
TORUS PER DETAILED DESCRIPTION; HCTL

Pressure in the
wetwell. (PWW)



Torus water temperature.
(TWSP)

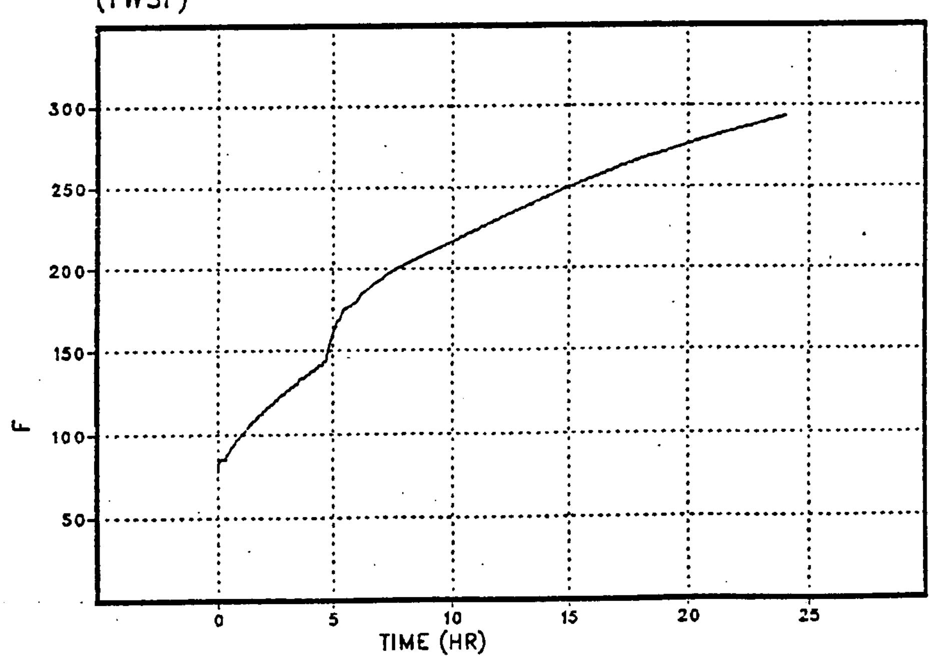
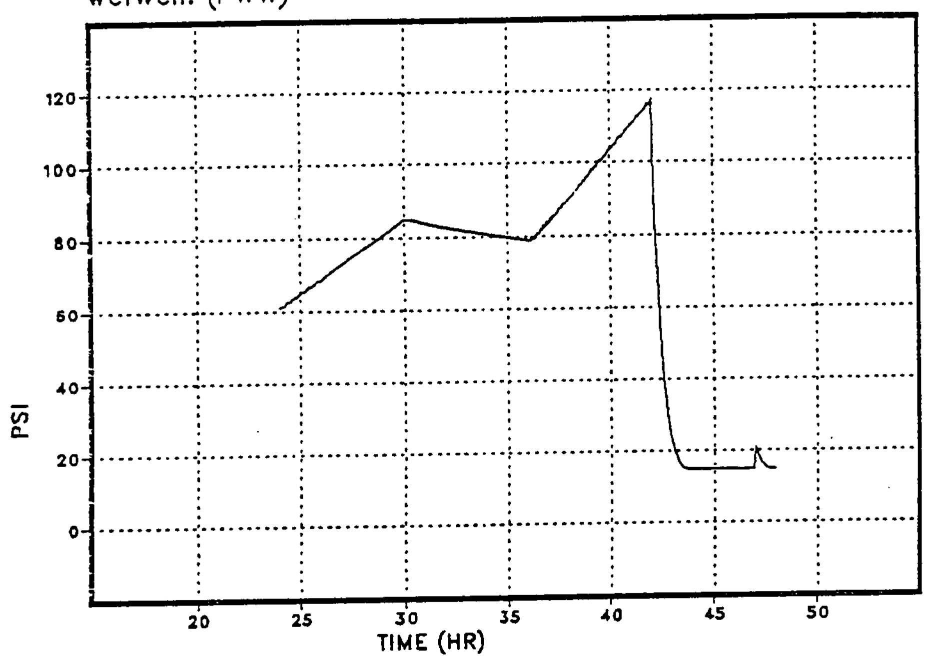


Figure 7.2-60

TW MSIV CLOSE; INJECT FROM
TORUS PER DESCRIPTION; HCTL # RESTART #

Pressure in the
wetwell. (PWW)



Torus water temperature.
(TWSP)

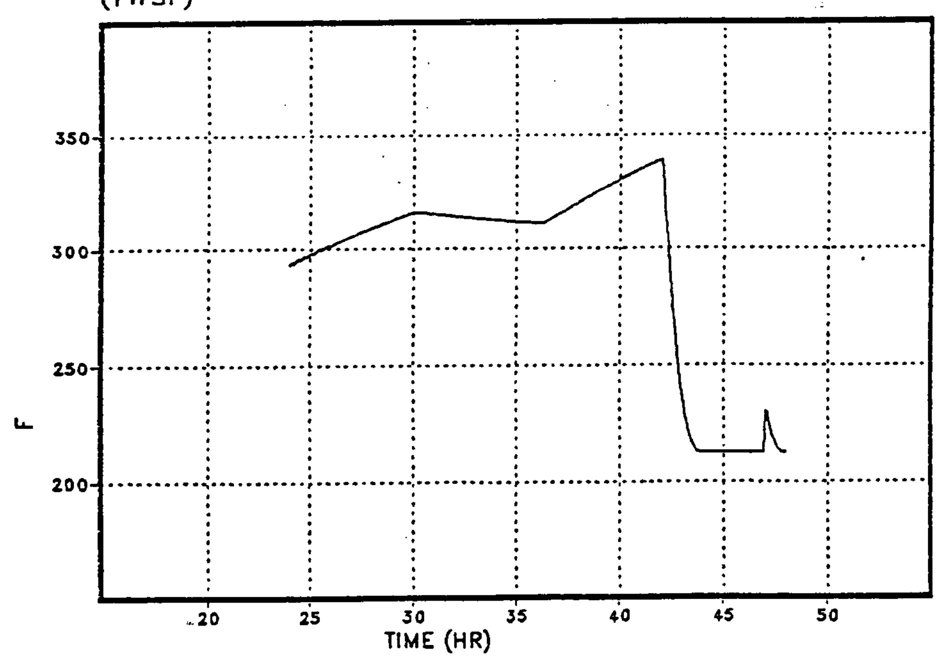
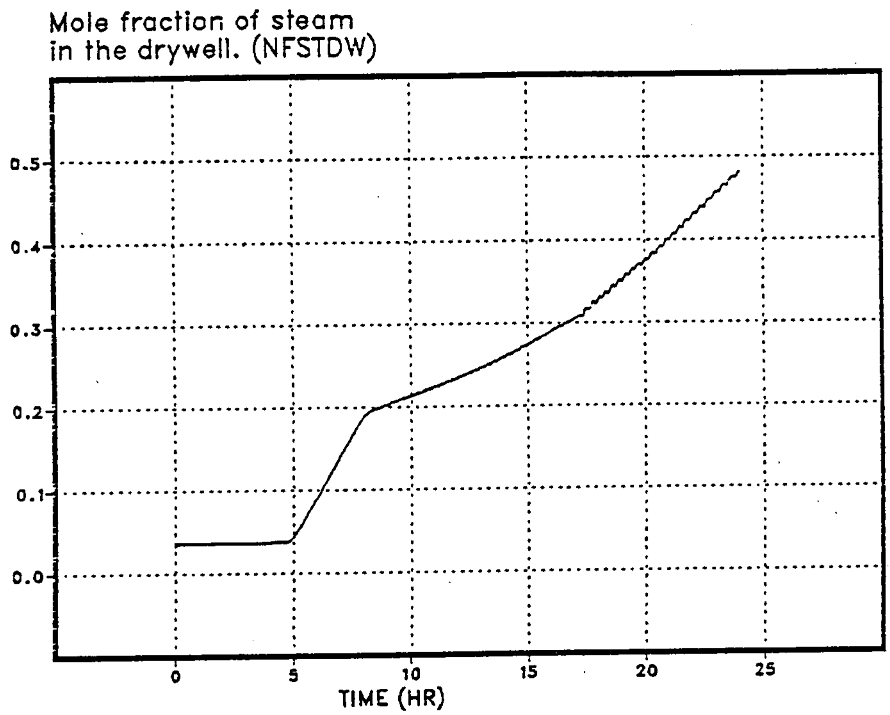


Figure 7.2-61

TW MSIV CLOSE; INJECT FROM
TORUS PER DETAILED DESCRIPTION; HCTL



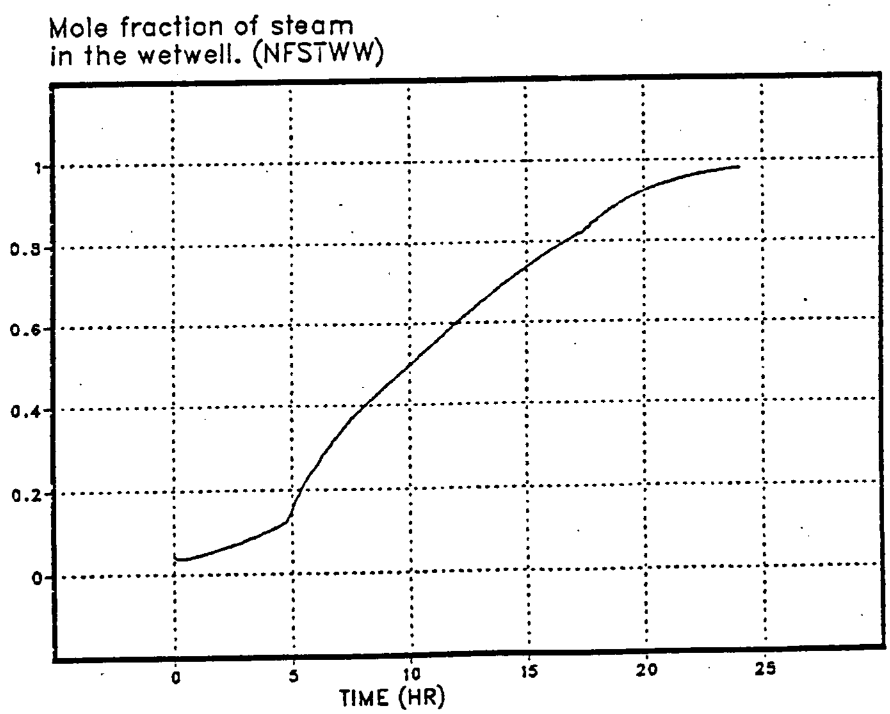
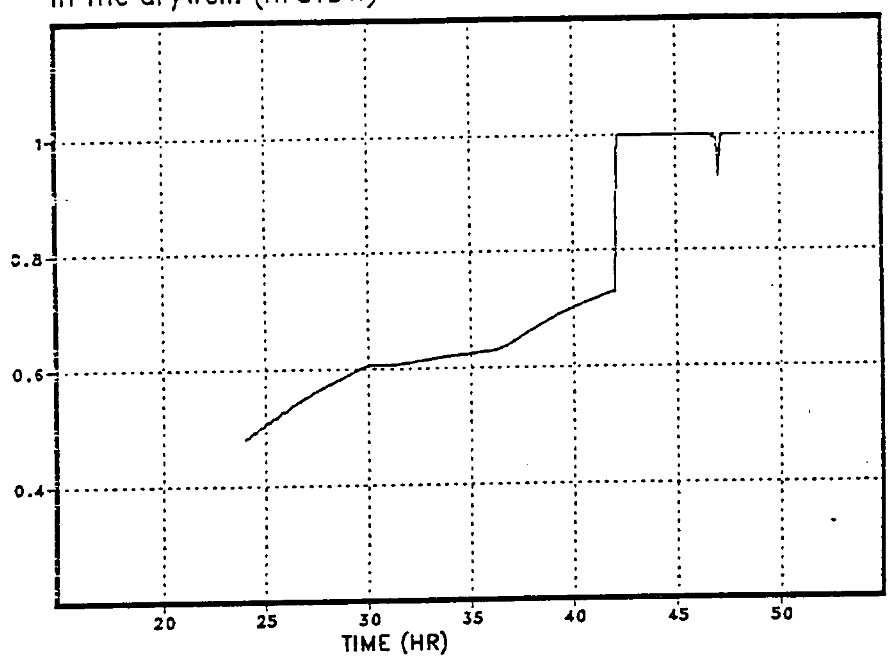


Figure 7.2-62

TW MSIV CLOSE; INJECT FROM
TORUS PER DESCRIPTION; HCTL # RESTART #

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

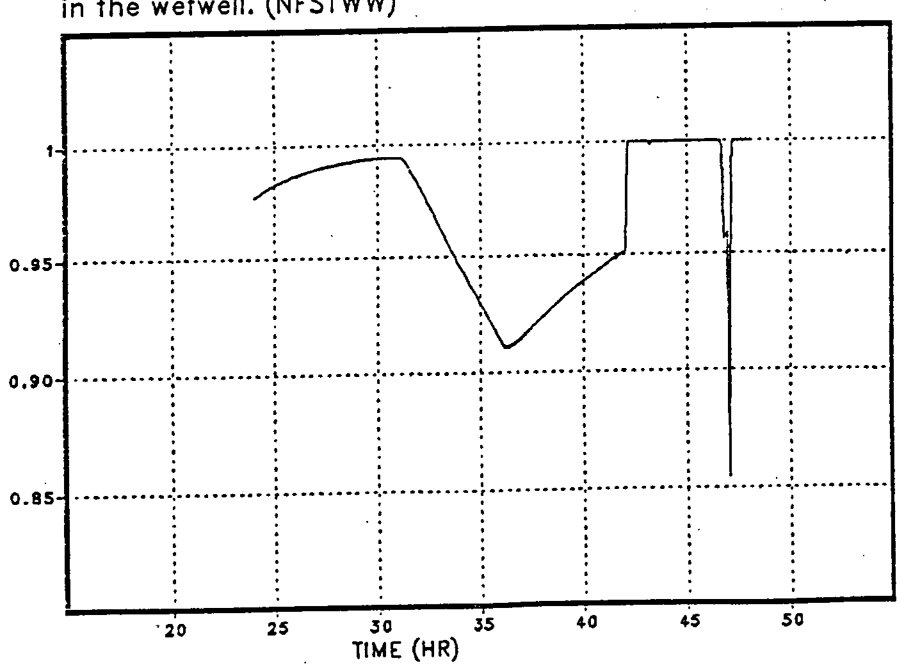
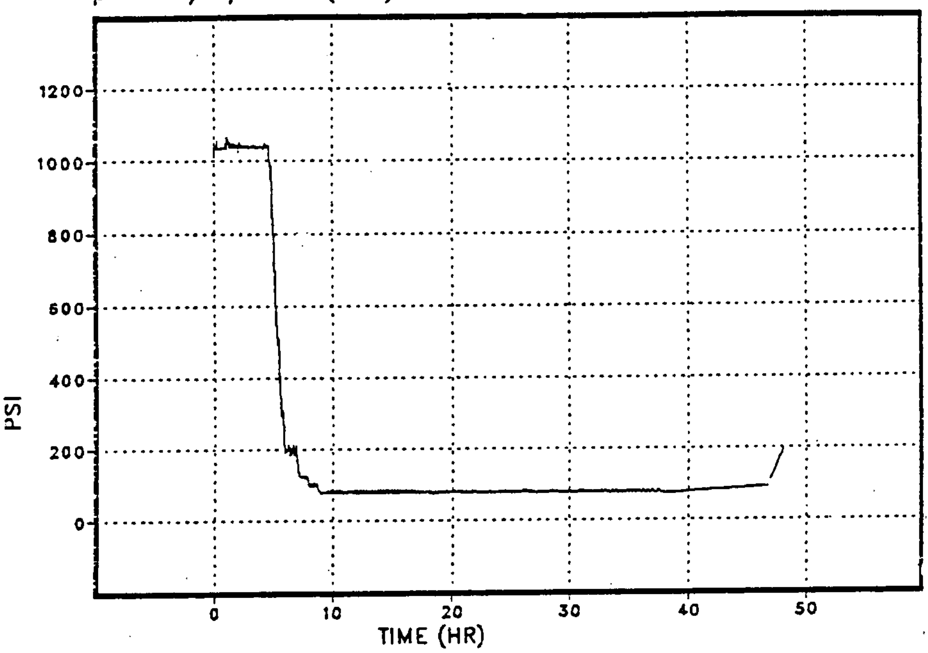


Figure 7.2-63

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; DEPRESS PER HCTL

Pressure in the
primary system. (PPS)



Maximum temperature in
core. (T110P)

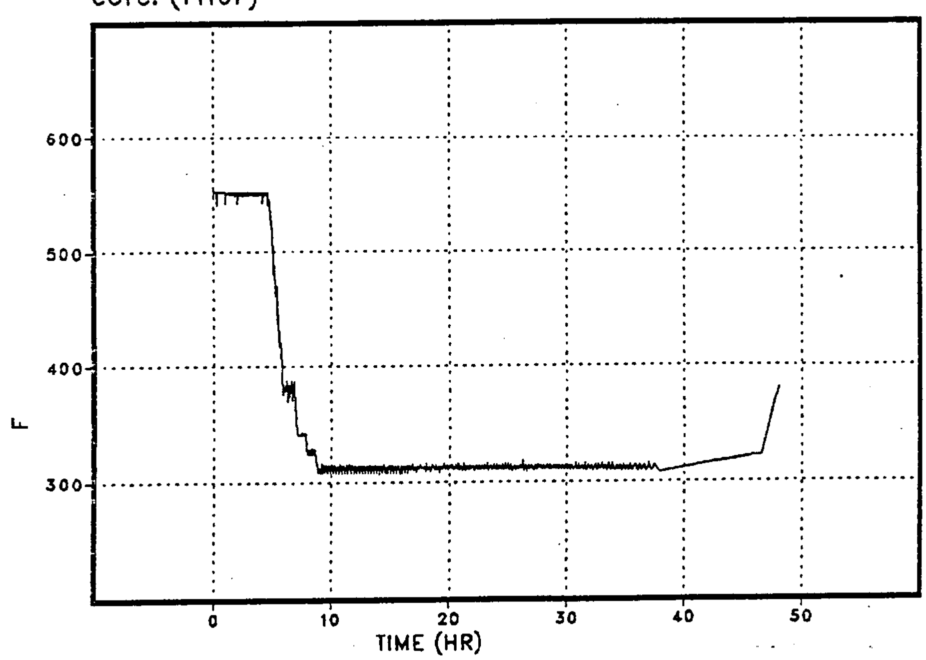
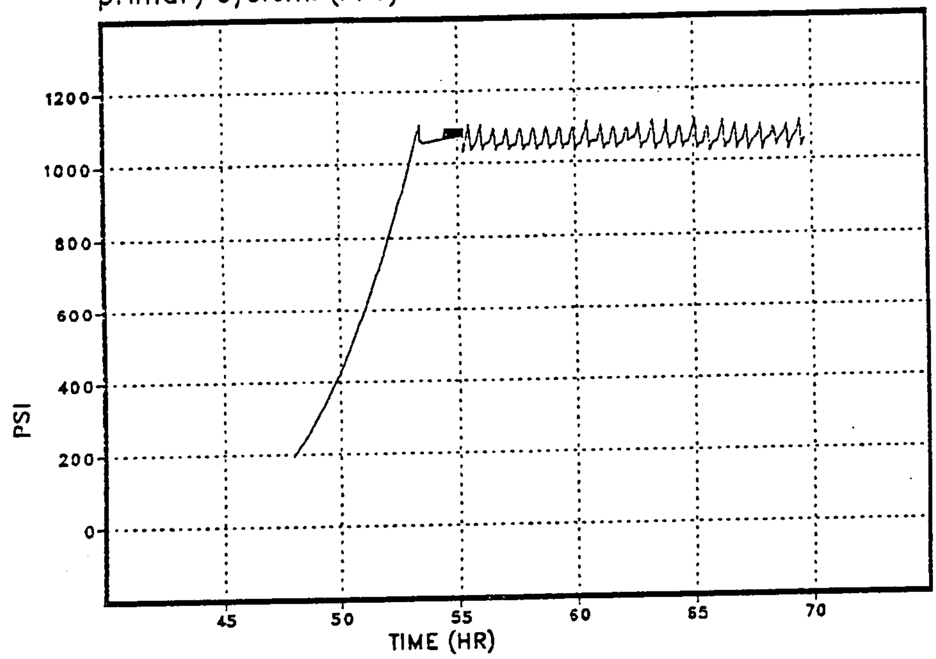


Figure 7.2-64

TW MSIV CLOSE; INJECTION FROM ....
CST'S ONLY; HCTL %% RESTART 1 %%

Pressure in the
primary system. (PPS)



Maximum temperature in
core. (T110P)

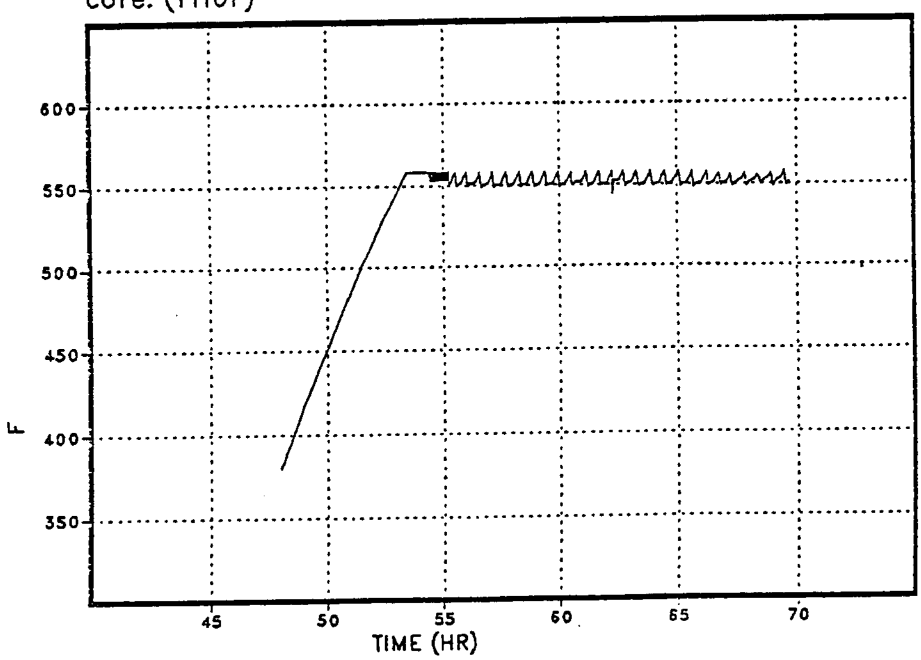
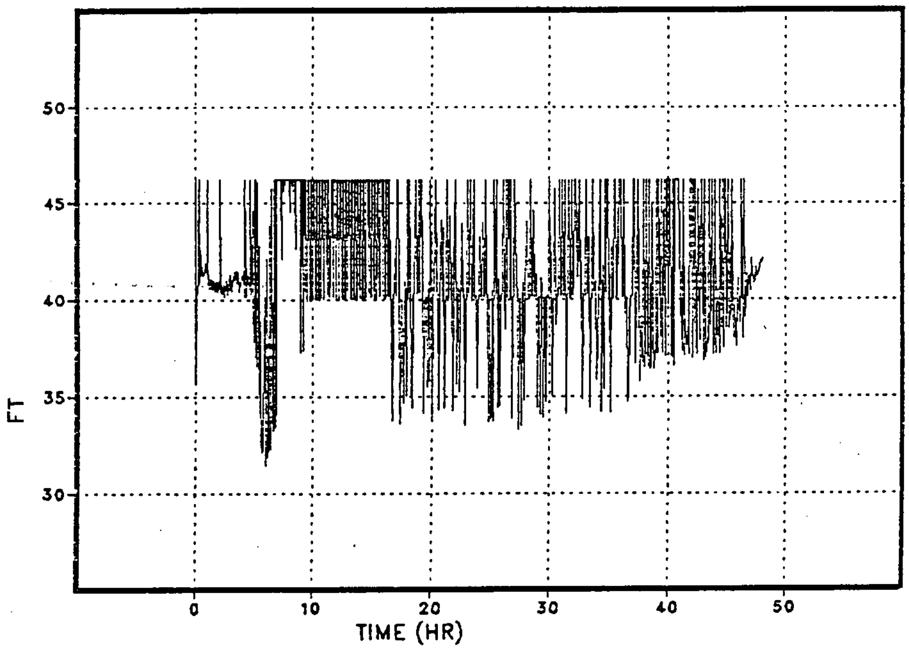


Figure 7.2-65

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; DEPRESS PER HCTL

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

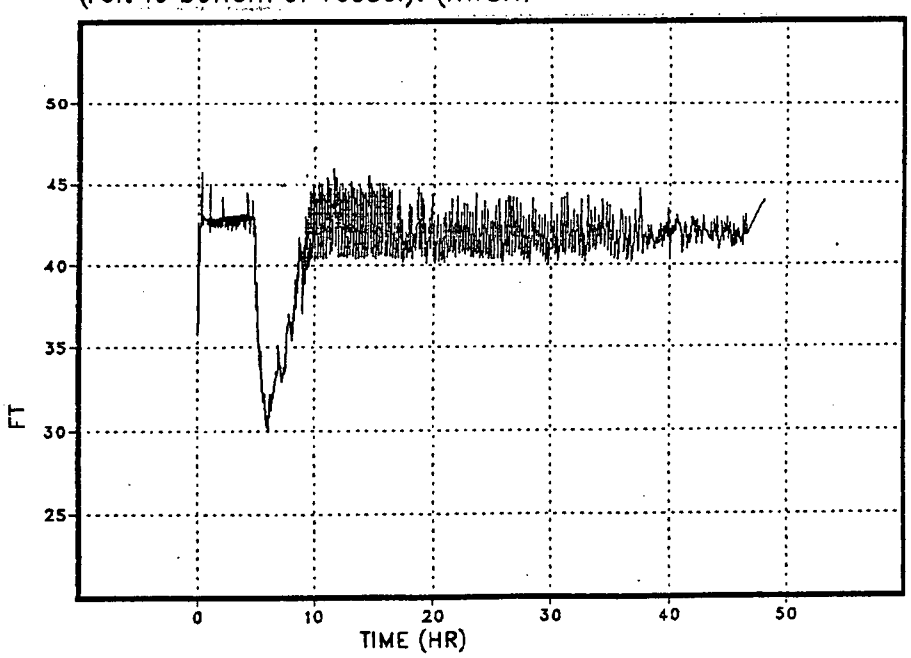
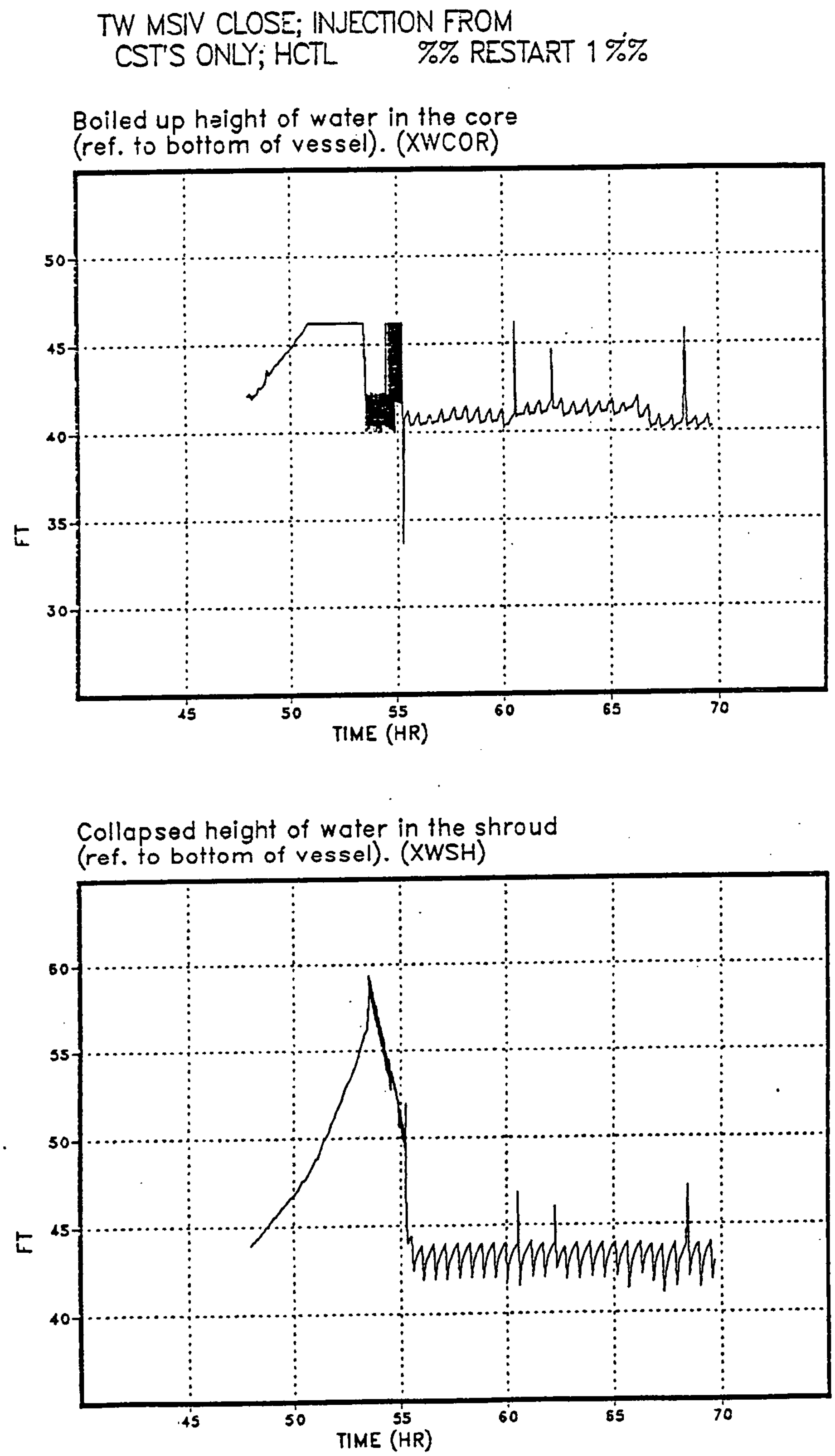


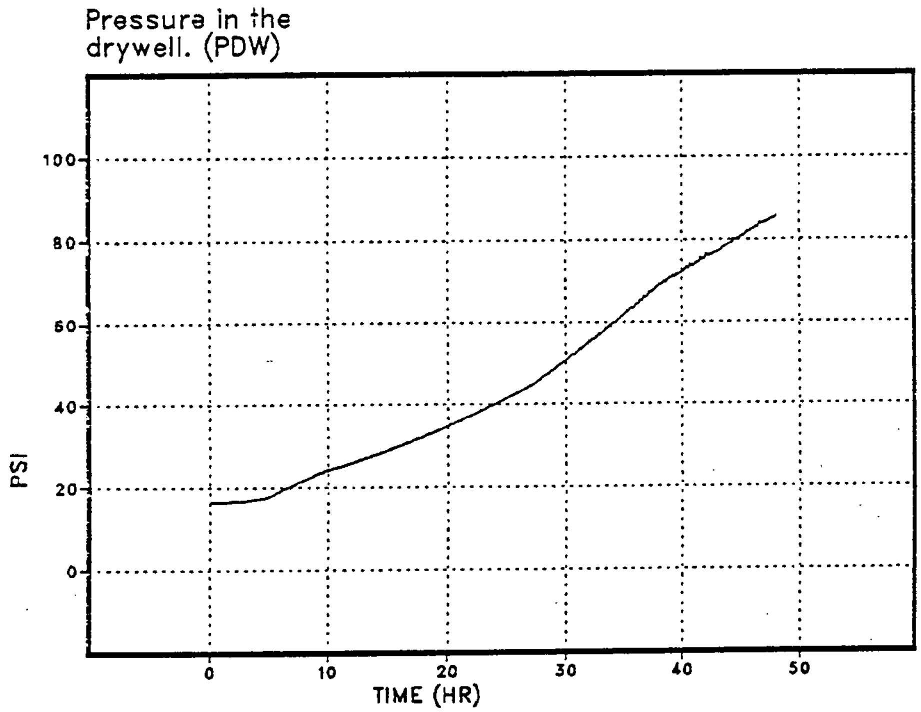
Figure 7.2-66



TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; HCTL %% RESTART 1 %%
Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)
50
45
40
35
30
FT
45
50
55
60
65
70
TIME (HR)
Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)
60
55
50
45
40
FT
45
50
55
60
65
70
TIME (HR)

Figure 7.2-67

TW MSIV CLOSE; INJECTION FROM CST'S ONLY; DEPRESS PER HCTL



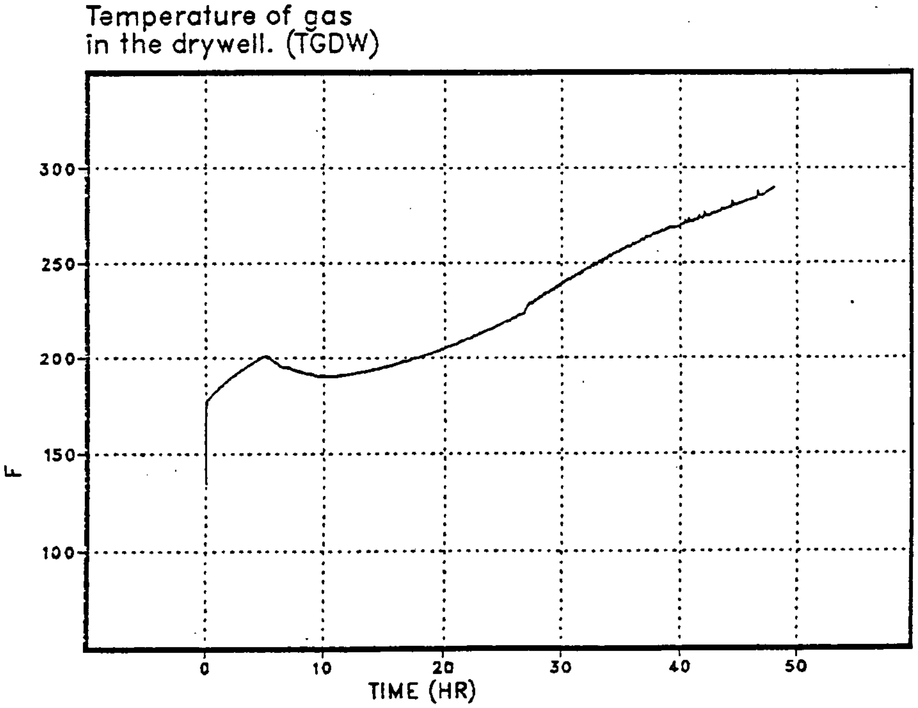
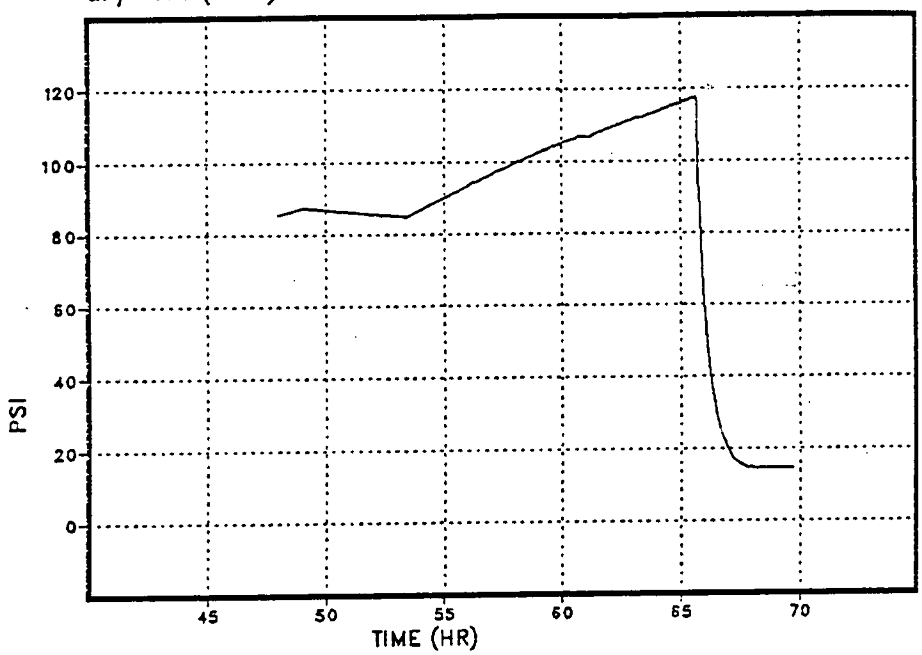


Figure 7.2-68

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; HCTL %% RESTART 1%%



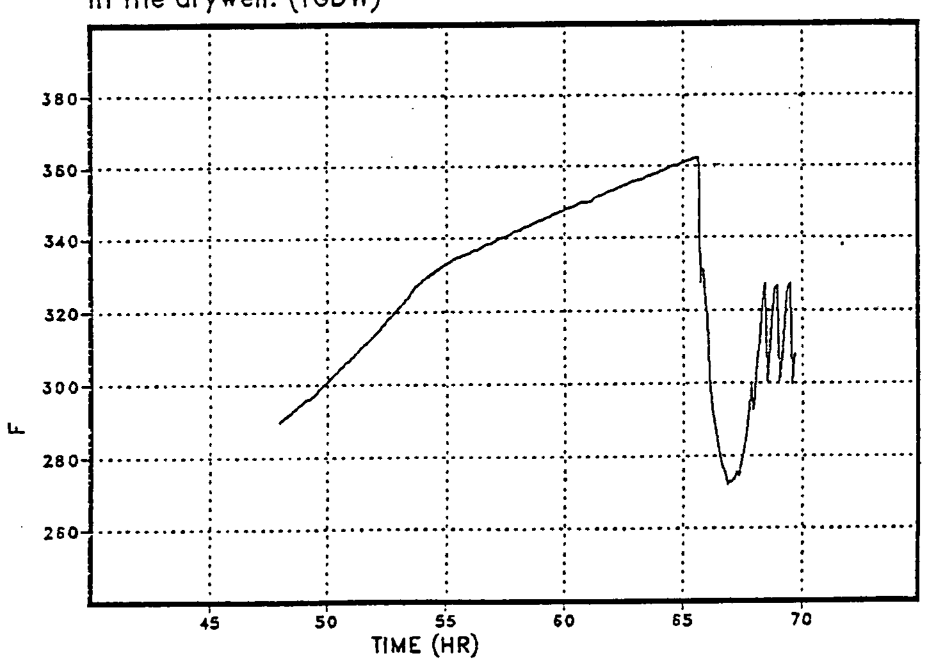
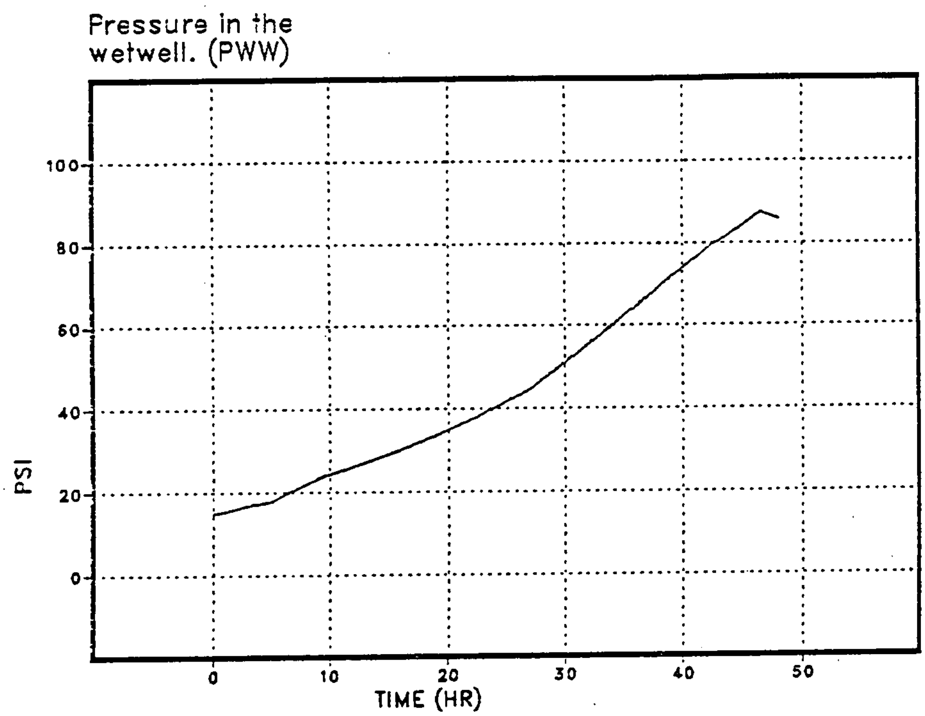


Figure 7.2-69

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; DEPRESS PER HCTL



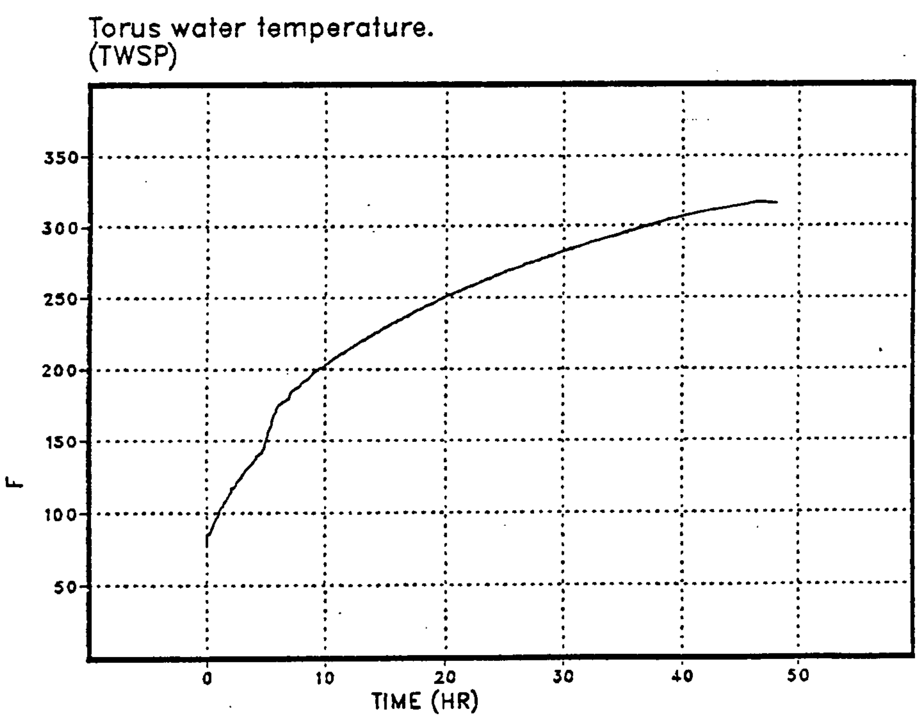
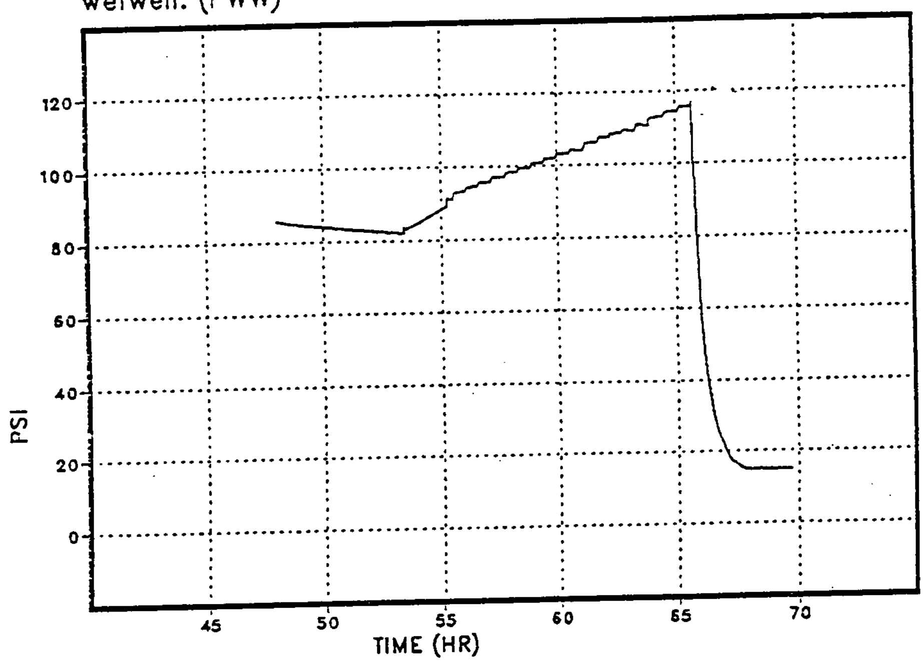


Figure 7.2-70

TW MSIV CLOSE; INJECTION FROM ...
CST'S ONLY; HCTL %% RESTART 1 %%

Pressure in the wetwell. (PWW)



Torus water temperature. (TWSP)

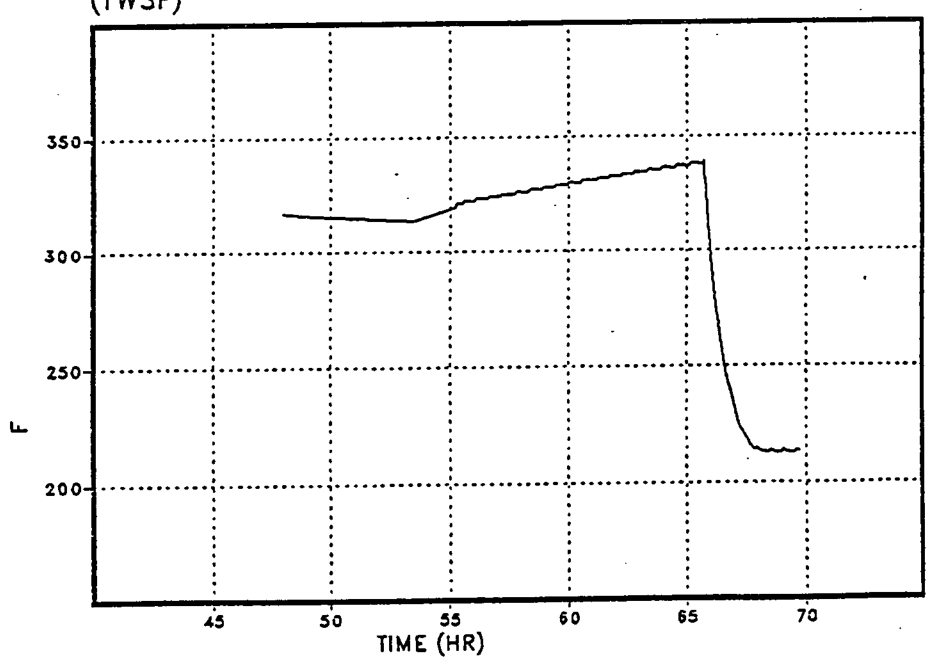
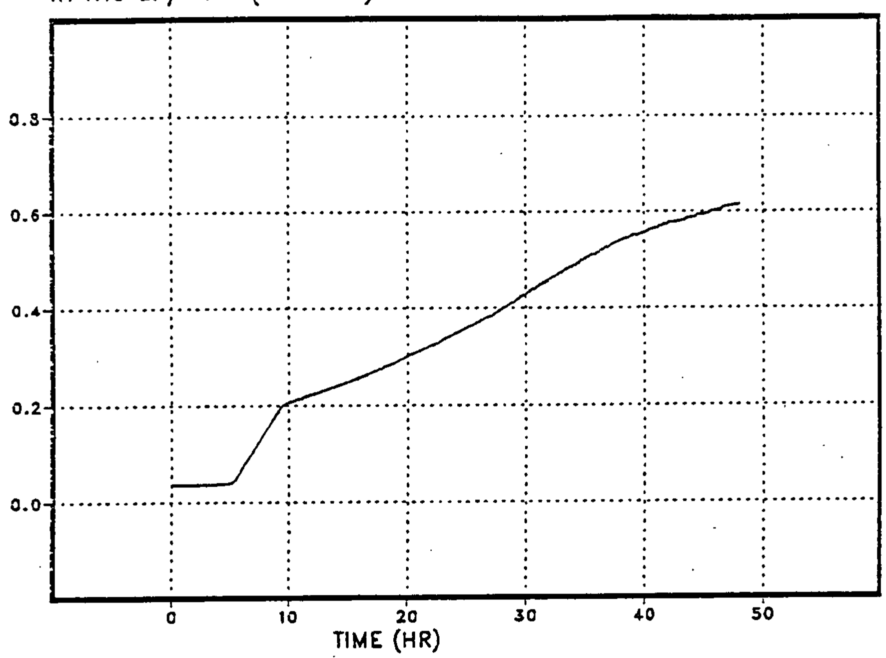


Figure 7.2-71

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; DEPRESS PER HCTL

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

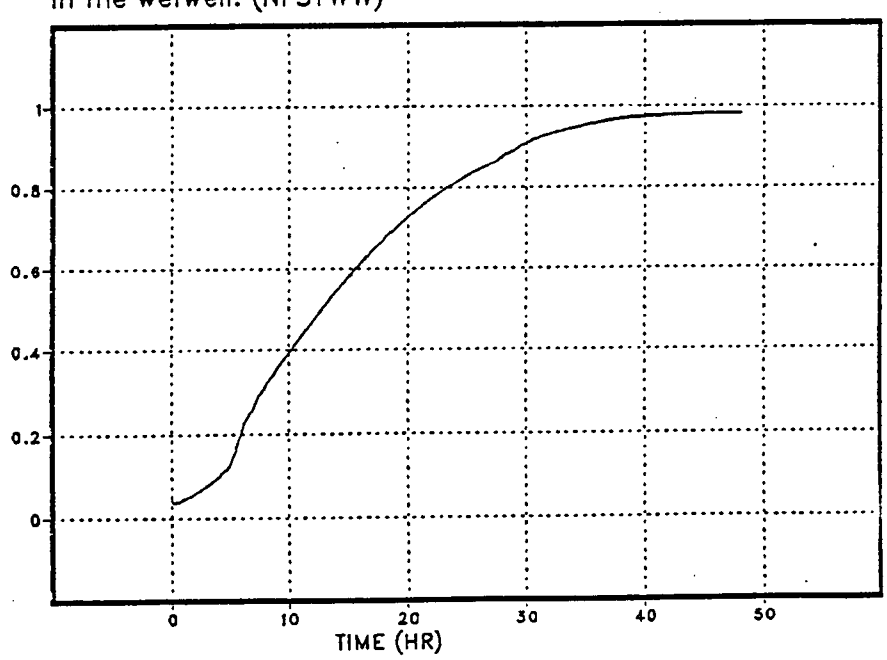
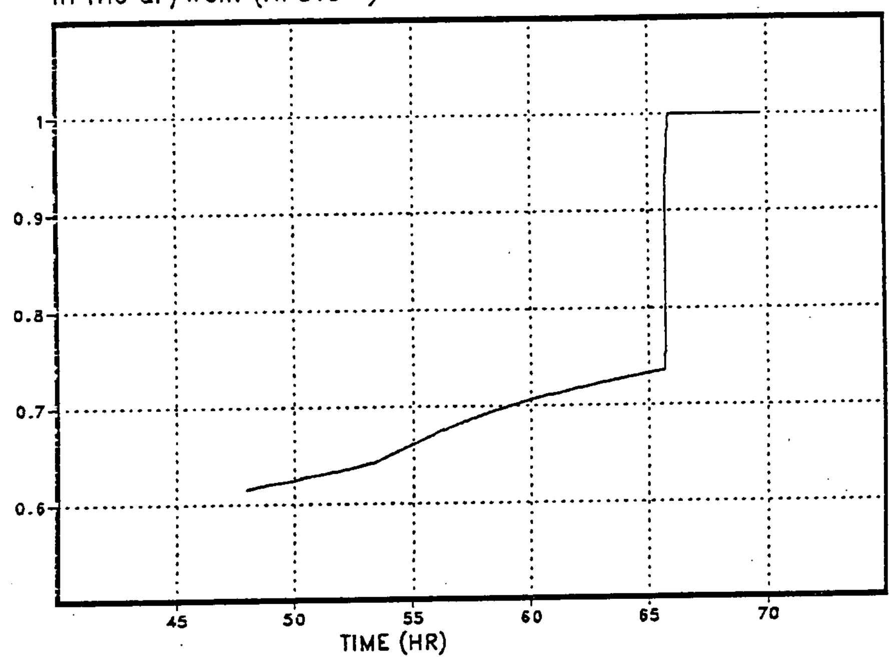


Figure 7.2-72

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; HCTL %% RESTART 1 %%

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

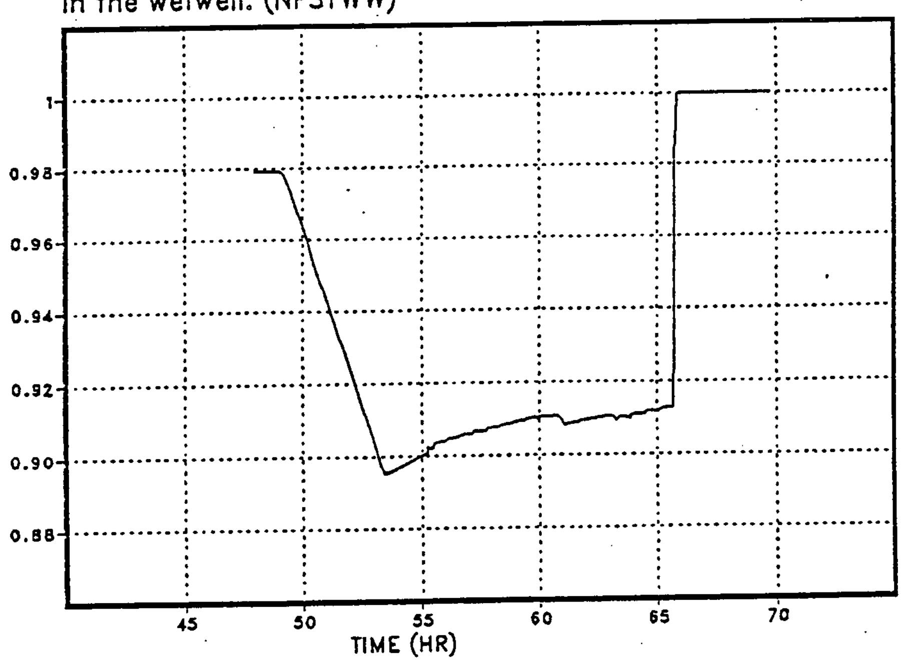
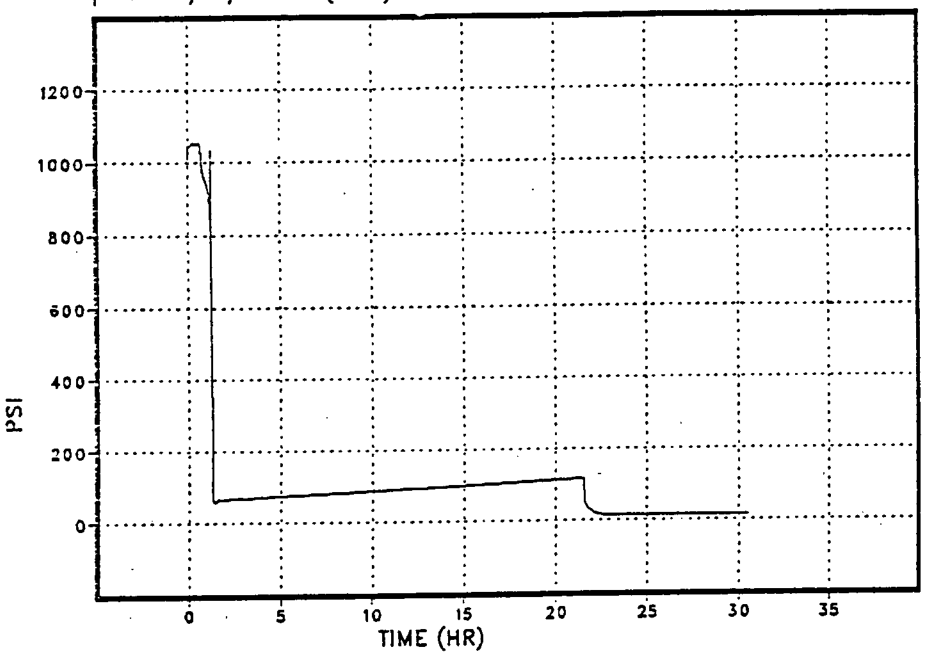


Figure 7.2-73

MAAP MNT SBLOCA W/ NO ADS
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE

Pressure in the primary system. (PPS)



Maximum temperature in core. (T110P)

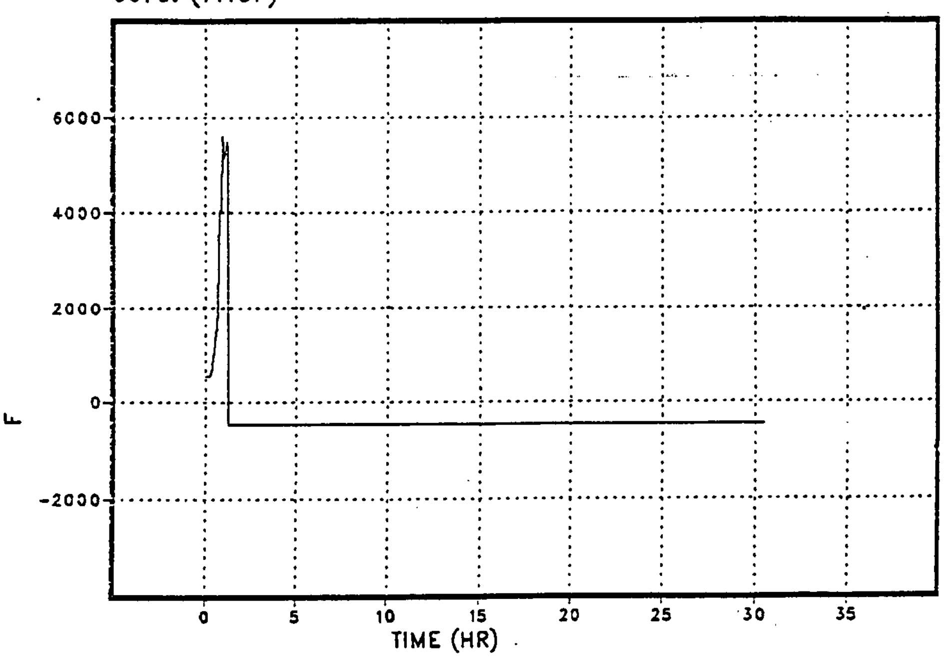
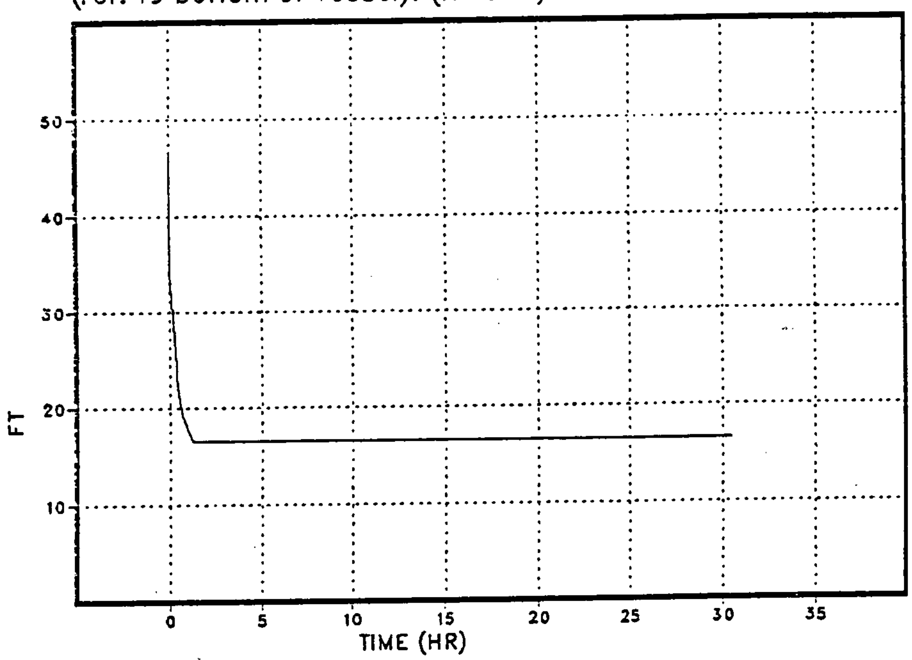


Figure 7.2-74

MAAP MNT SBLOCA W/ NO ADS
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

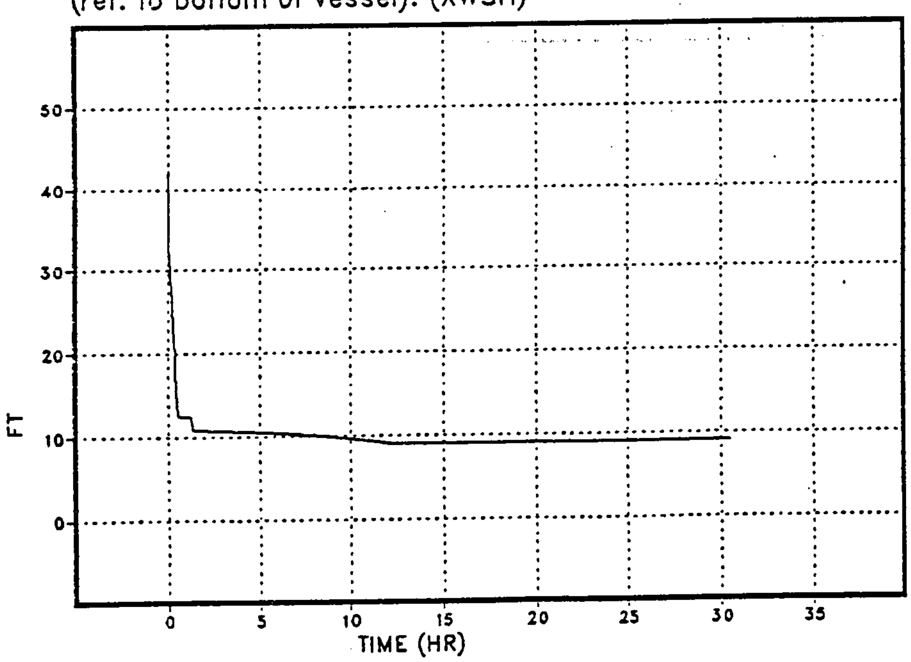
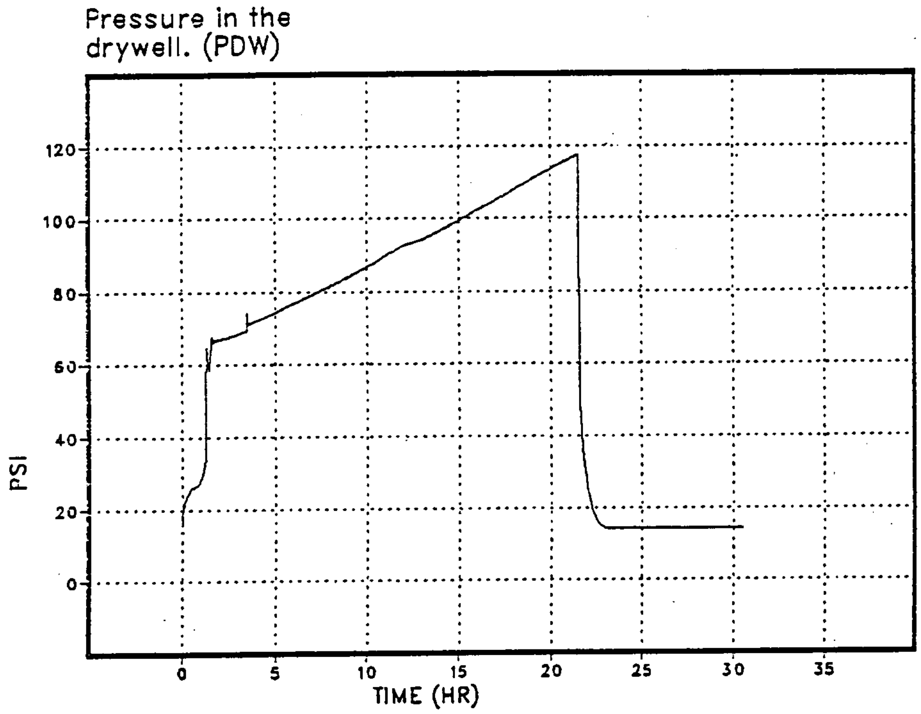


Figure 7.2-75

MAAP MNT SBLOCA W/ NO ADS
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE



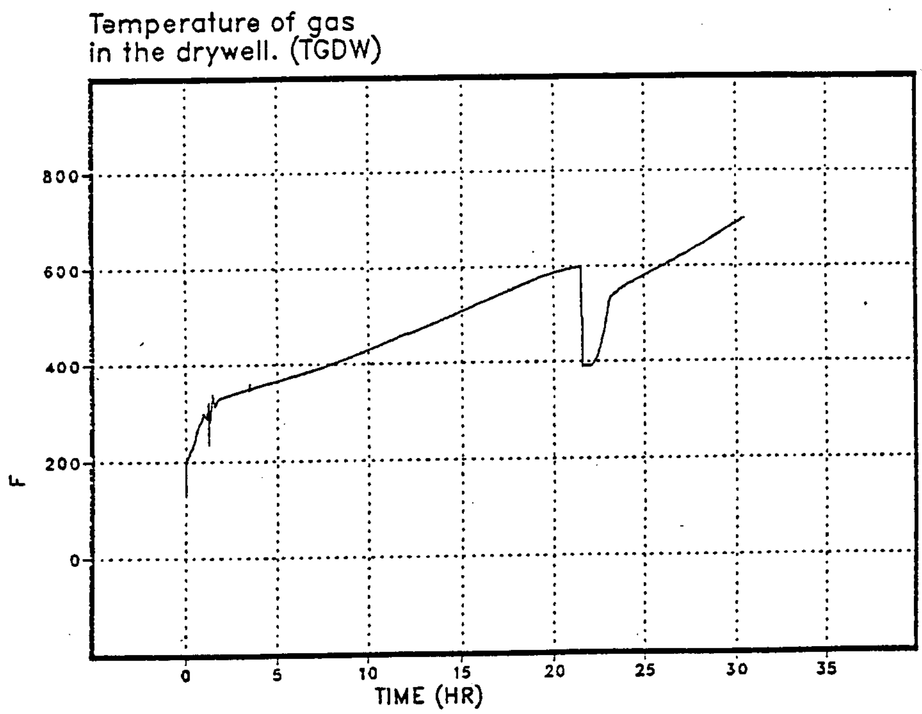
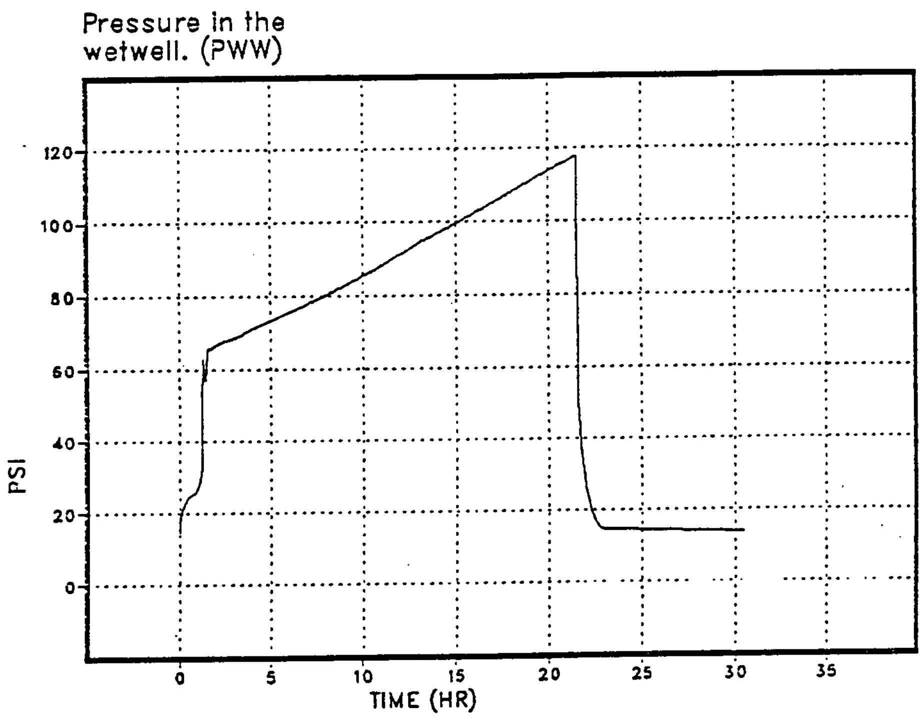


Figure 7.2-76

MAAP MNT SBLOCA W/ NO ADS ....
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE



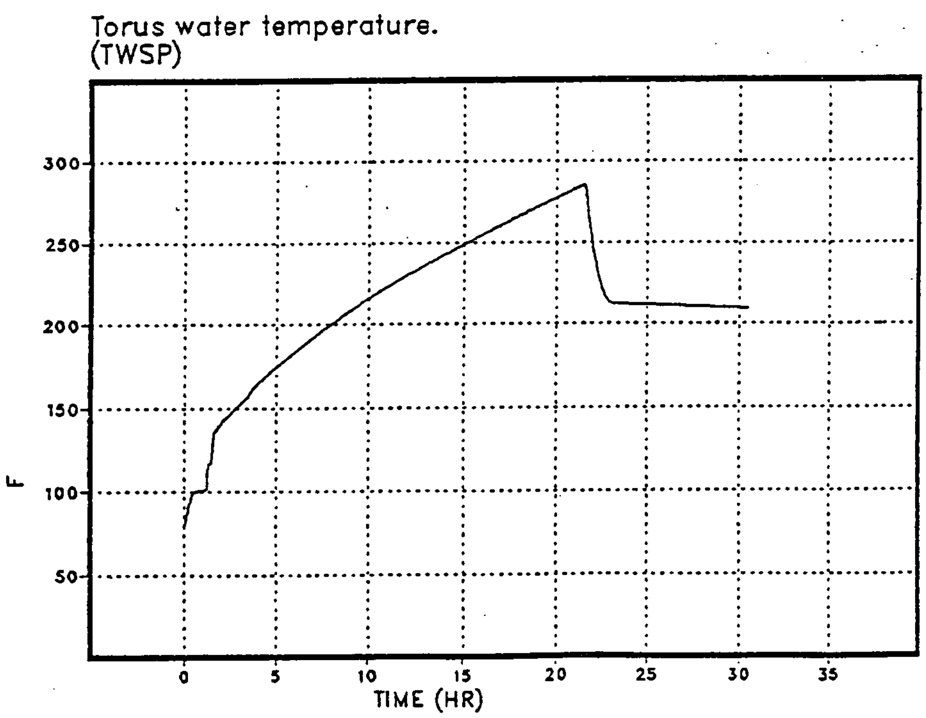
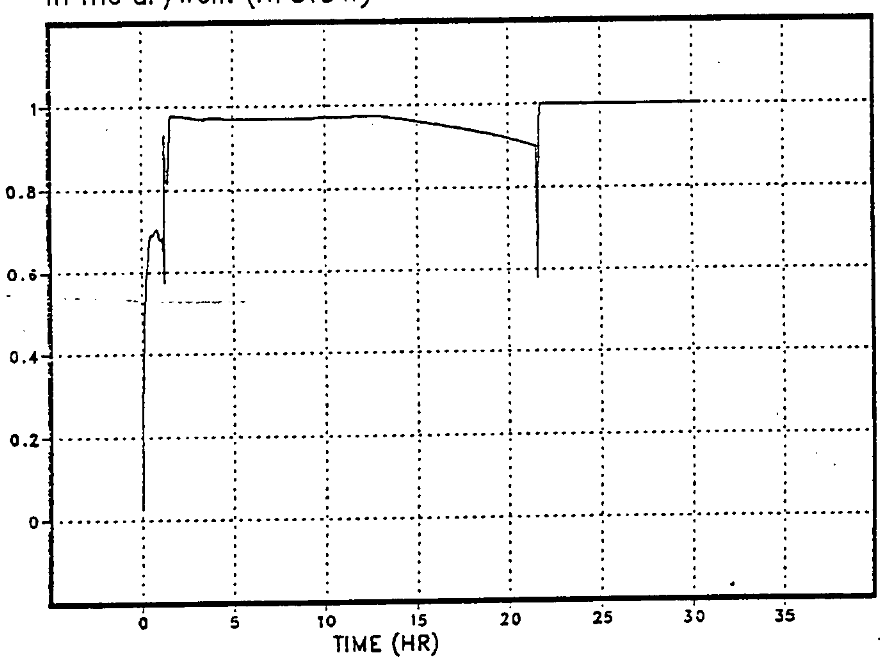


Figure 7.2-77

MAAP MNT SBLOCA W/ NO ADS
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

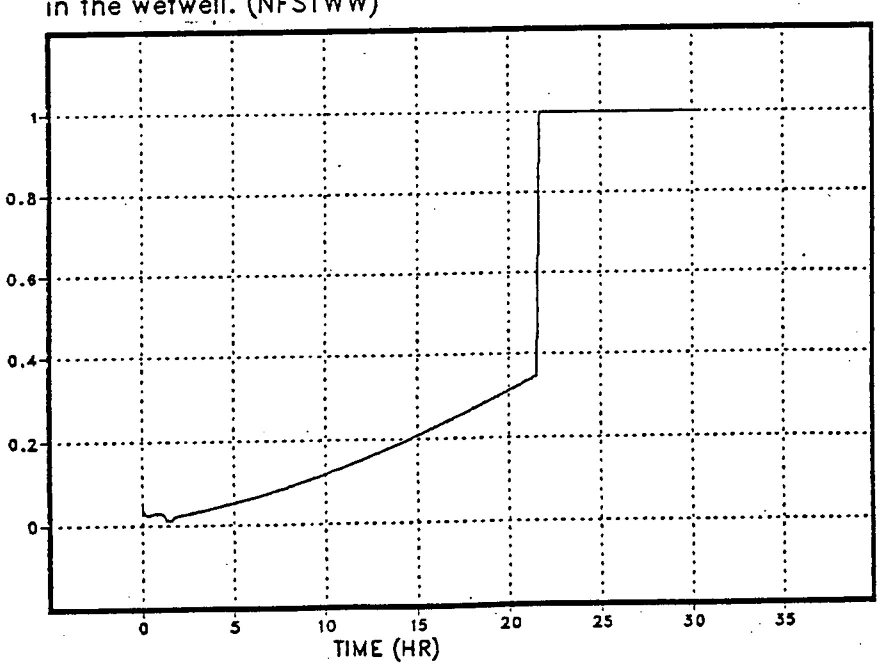
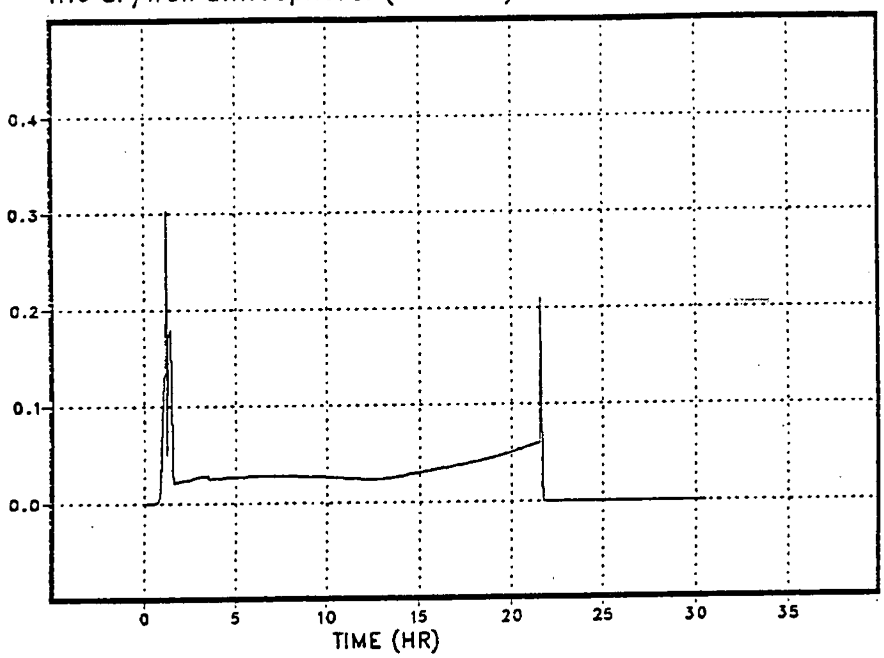


Figure 7.2-78

MAAP MNT SBLOCA W/ NO ADS
1 LPCI PUMP ONLY; FAIL TO DEPRESSURIZE

Mole fraction of hydrogen in
the drywell atmosphere. (NFH2DW)



Mole fraction of H2
in the wetwell. (NFH2WW)

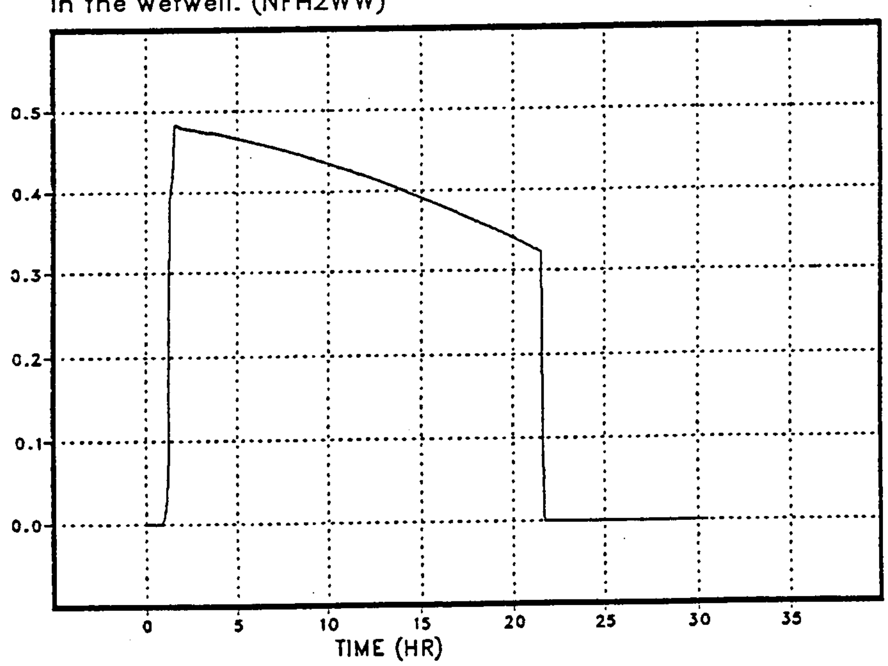
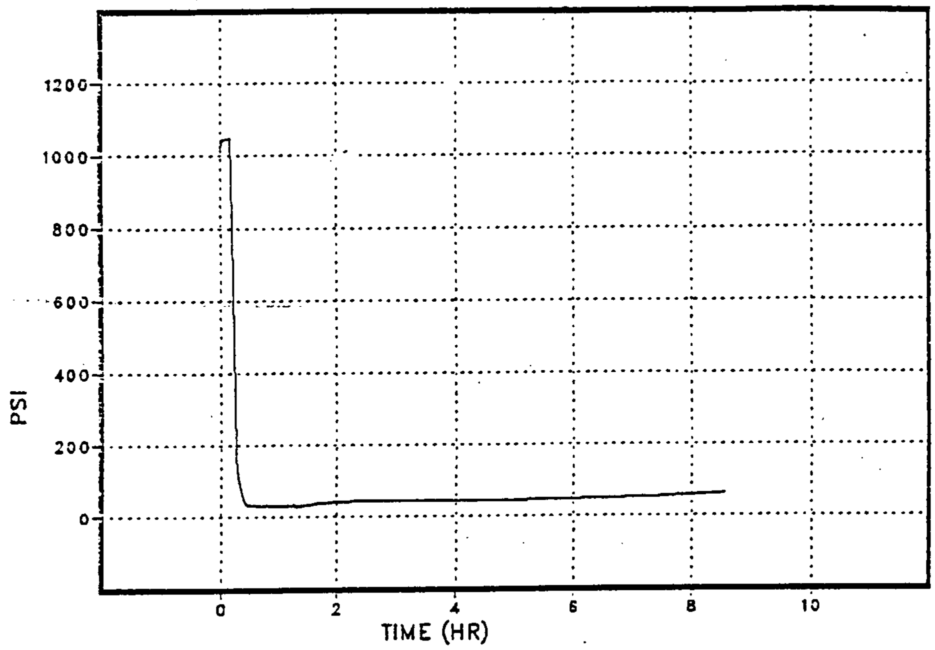


Figure 7.2-79

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT

Pressure in the
primary system. (PPS)



Maximum temperoture in
core. (T110P)

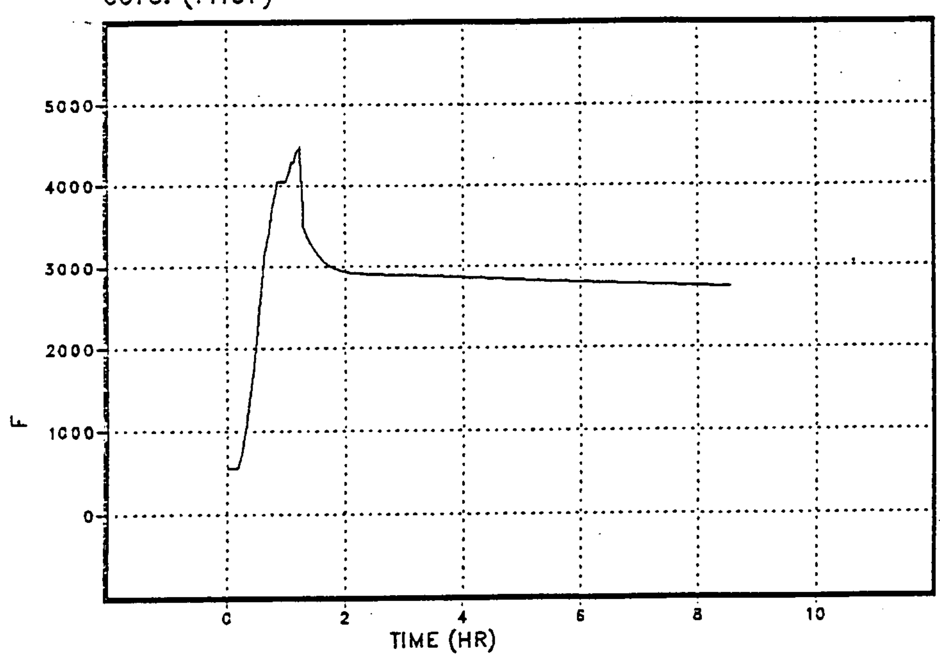
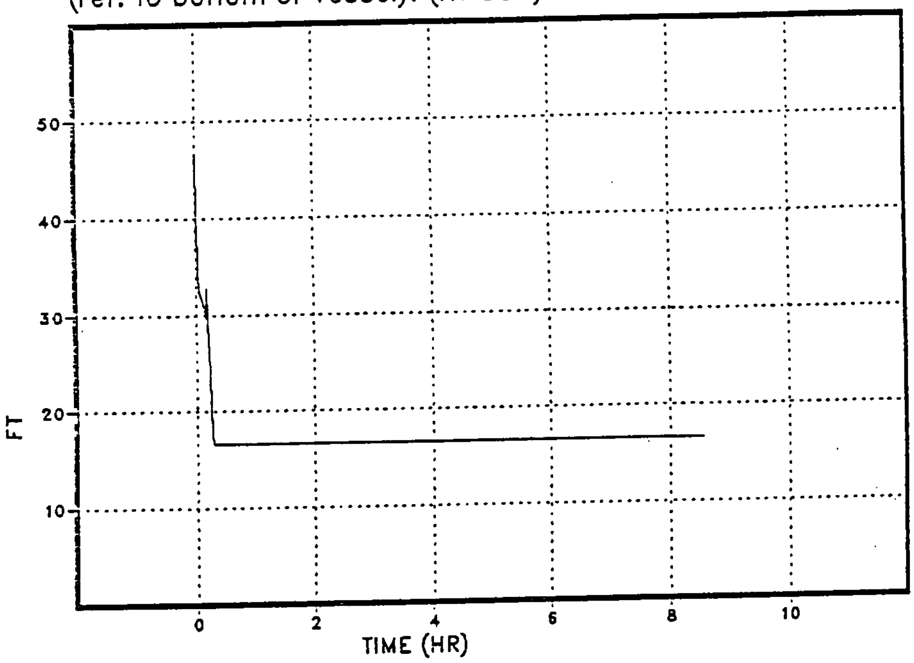


Figure 7.2-80

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

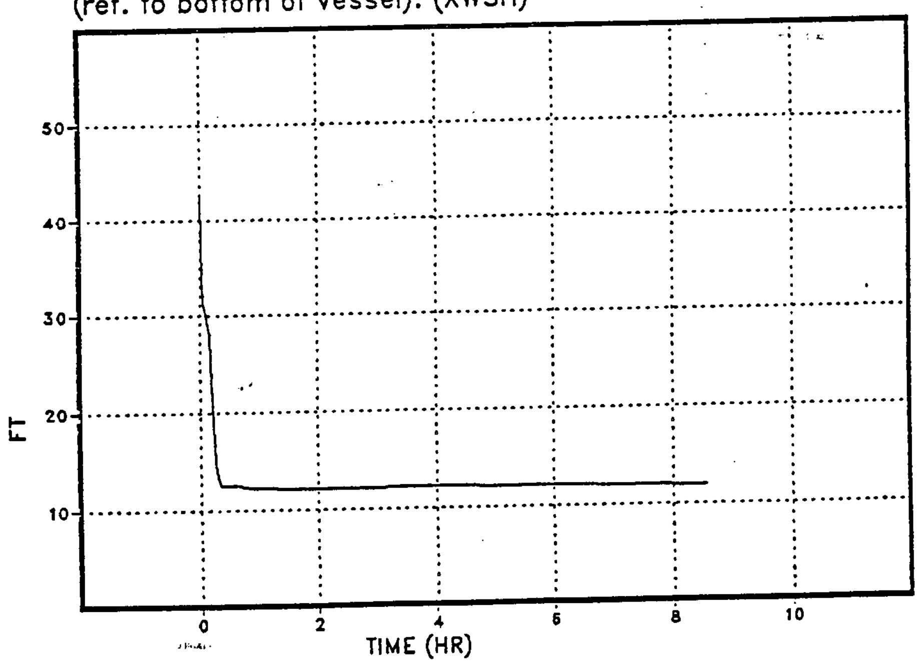
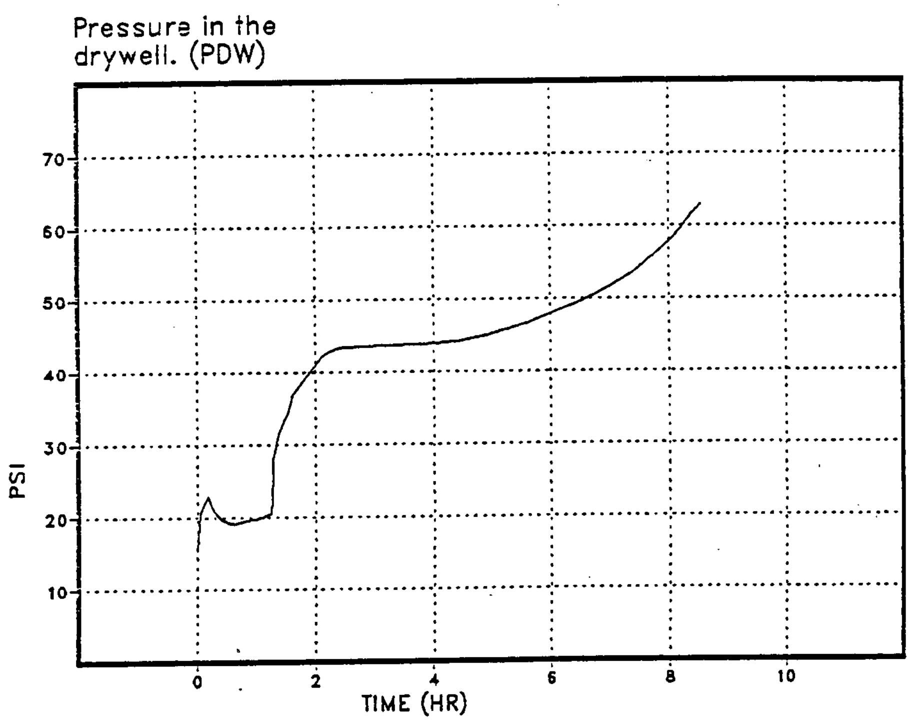


Figure 7.2-81

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT



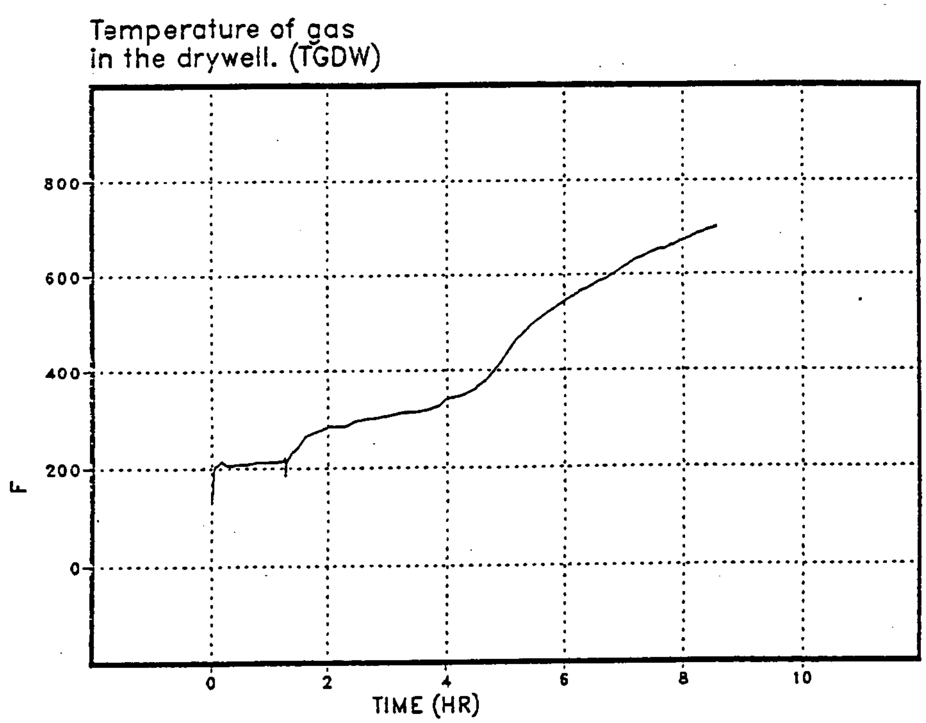
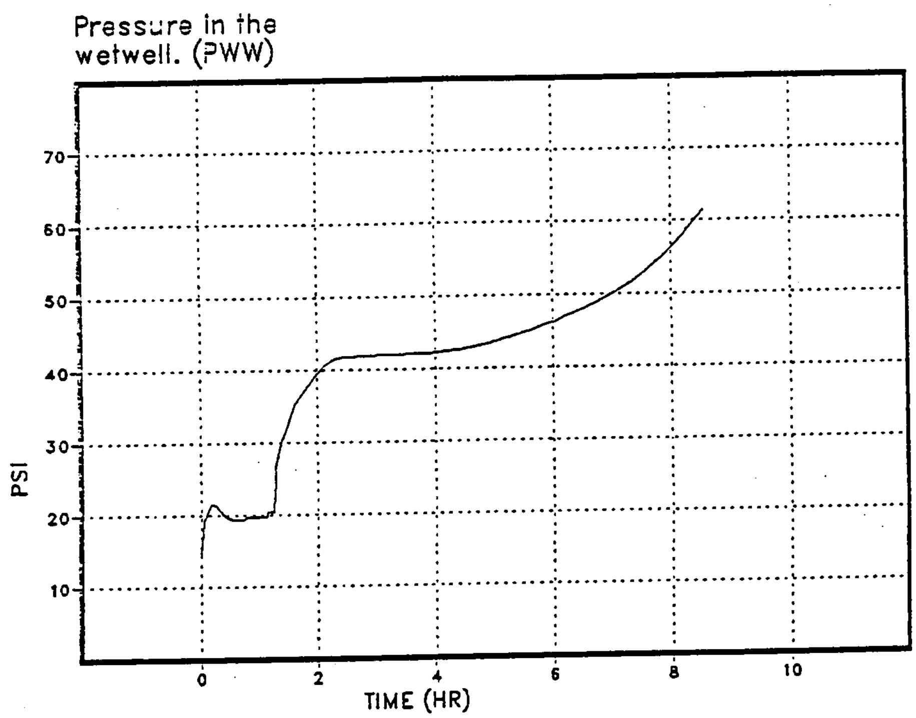


Figure 7.2-82

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT



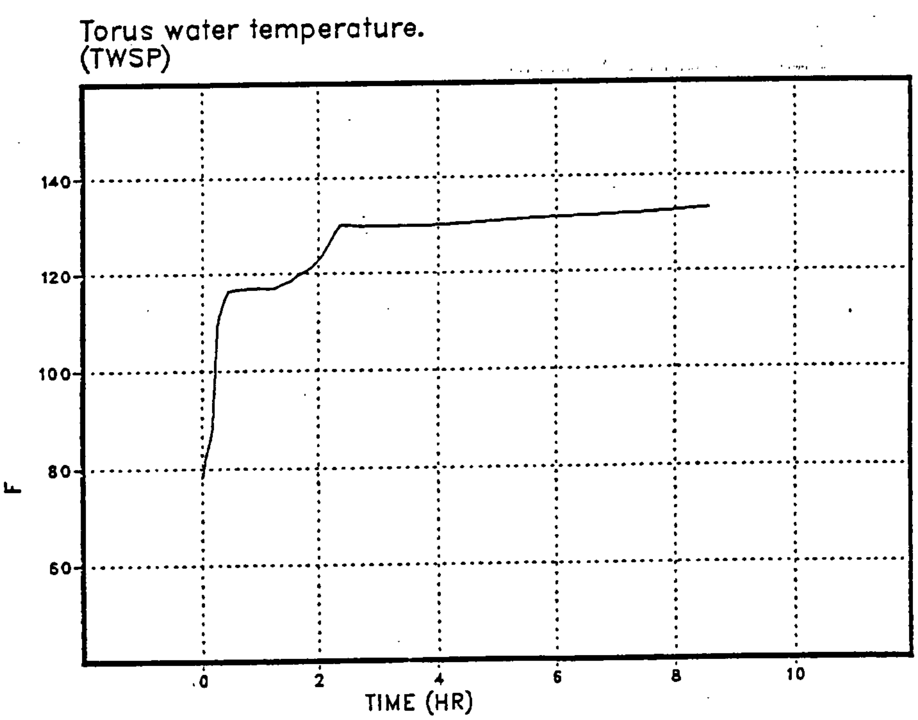


Figure 7.2-83

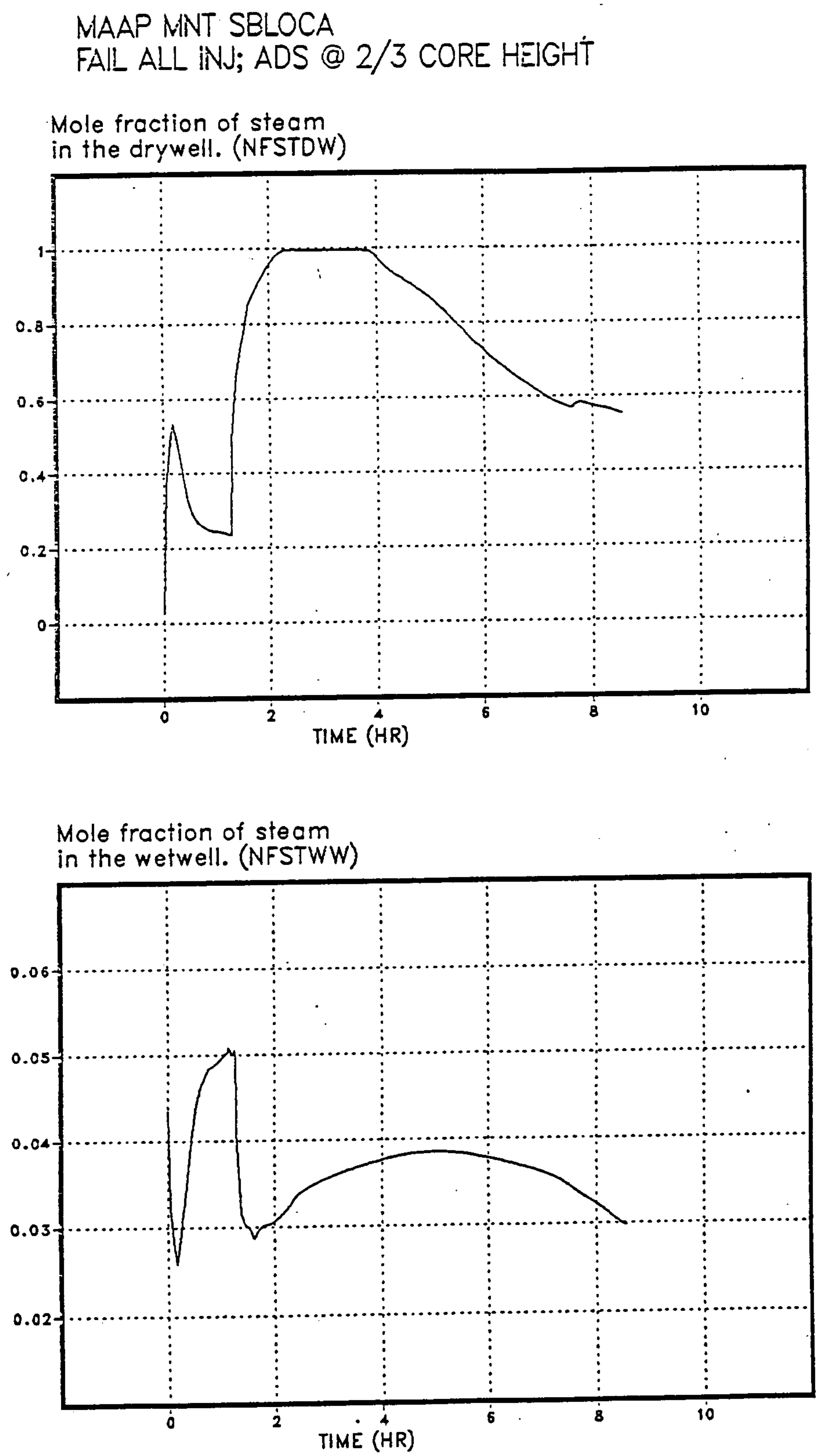
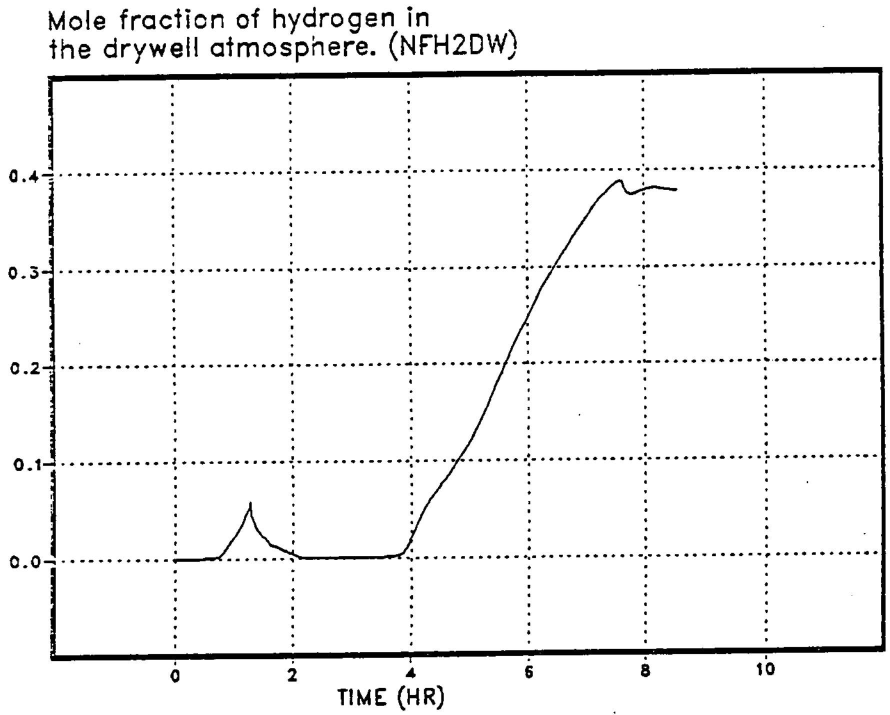

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT
Mole fraction of steam
in the drywell. (NFSTDW)
1
0.8
0.6
0.4
0.2
0
0
2
4
6
8
10
TIME (HR)
Mole fraction of steam
in the wetwell. (NFSTWW)
0.06
0.05
0.04
0.03
0.02
0
2
4
6
8
10
TIME (HR)

Figure 7.2-84

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT



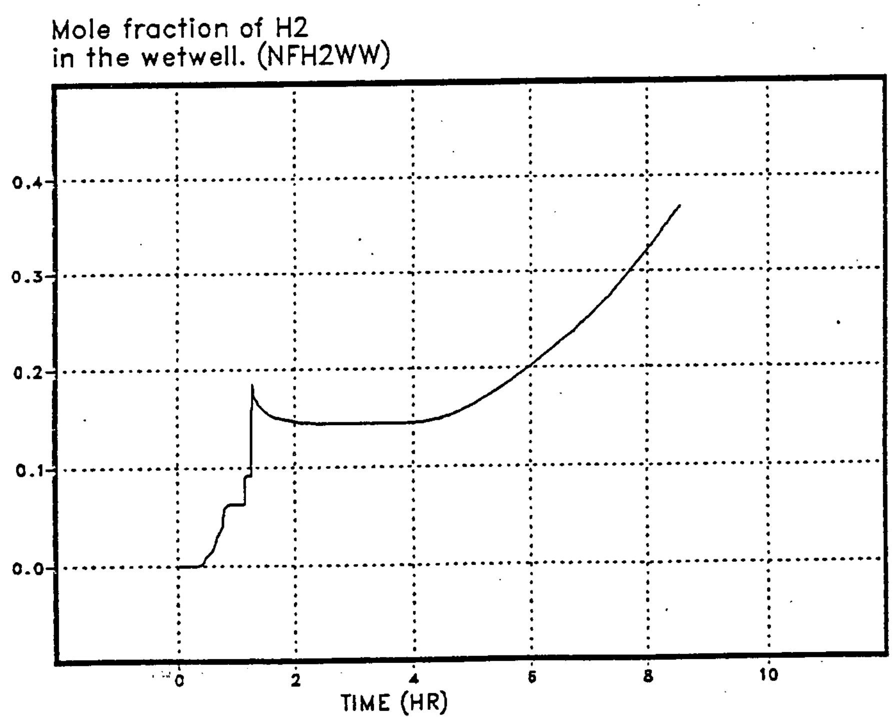
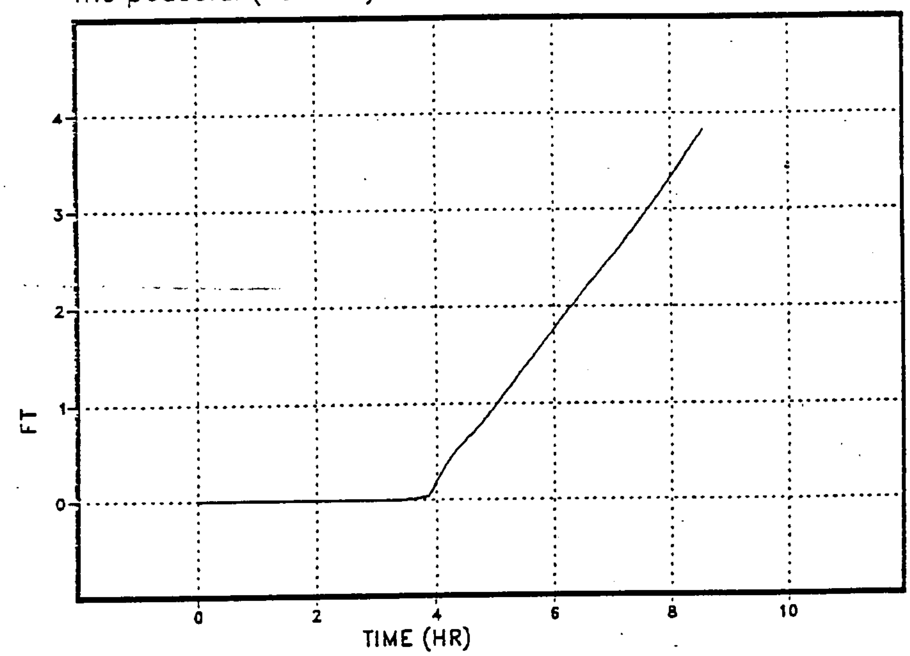


Figure 7.2-85

MAAP MNT SBLOCA
FAIL ALL INJ; ADS @ 2/3 CORE HEIGHT

Concrete ablation thickness in the pedestal (XCNPDP)



Height of water in the drywell. (XWDW)

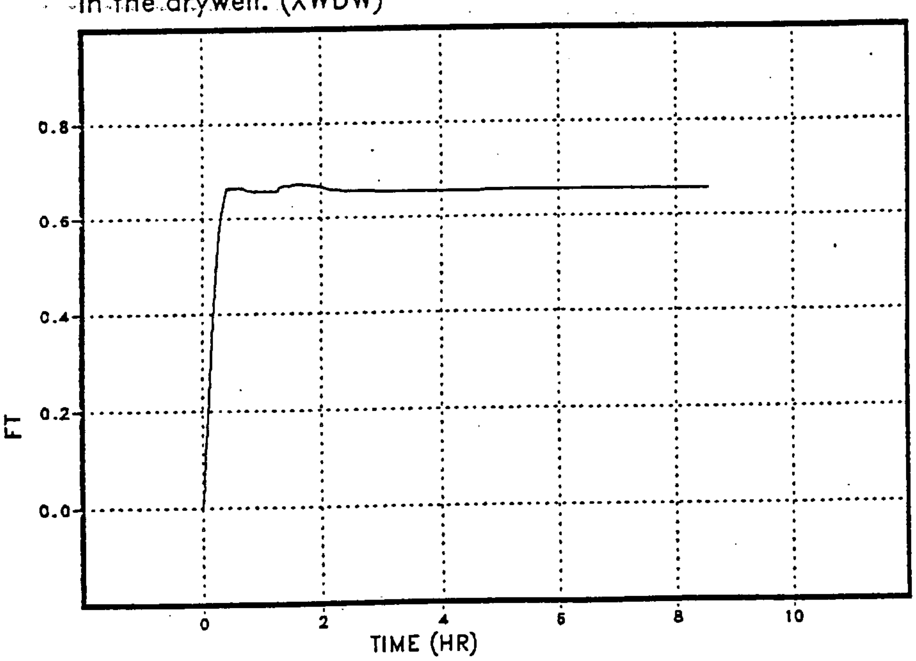
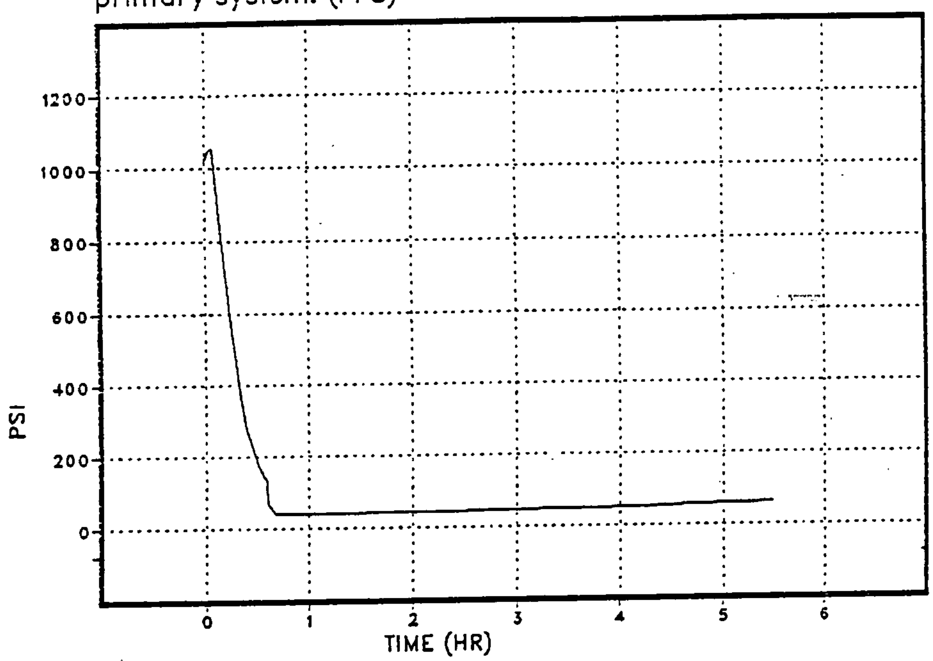


Figure 7.2-86

MNT MED LOCA W/ NO INJECTION
NO ADS

Pressure in the
primary system. (PPS)



Maximum temperature in
core. (T110P)

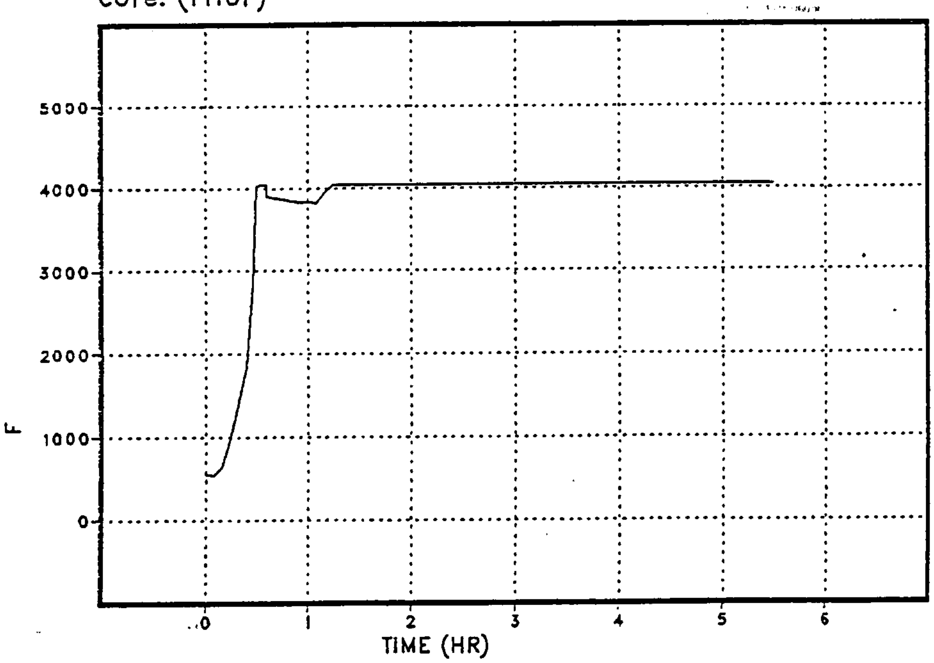
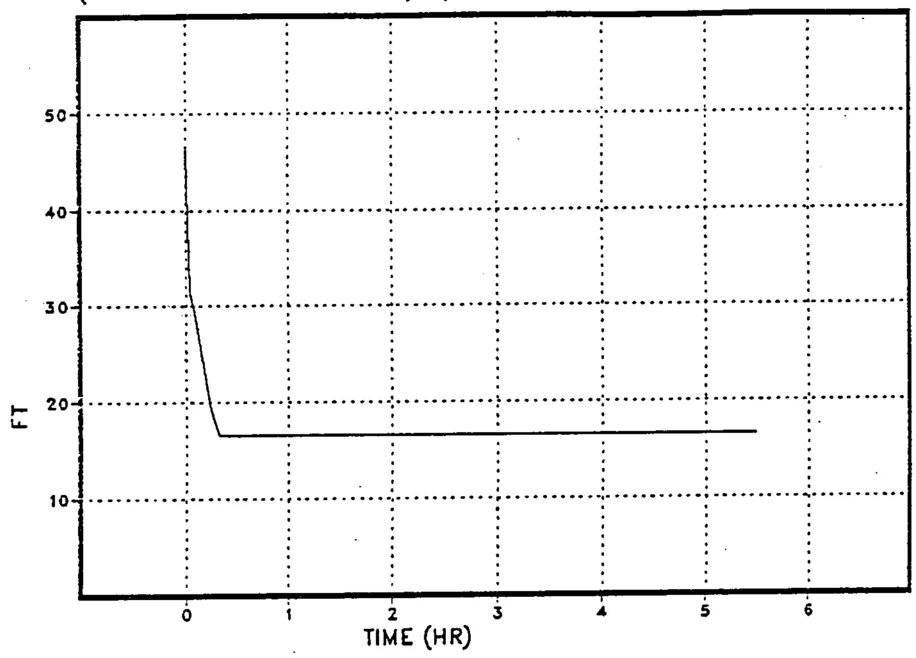


Figure 7.2-87

MNT MED LOCA W/ NO INJECTION
NO ADS

Boiled up height of water in the core
(ref. to bottom of vessel). (XWCOR)



Collapsed height of water in the shroud
(ref. to bottom of vessel). (XWSH)

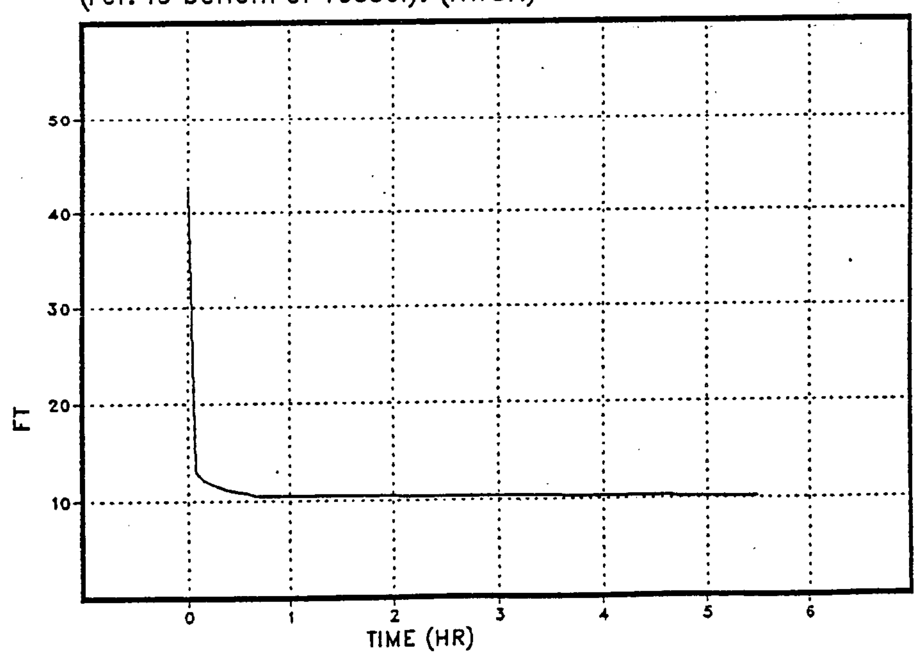
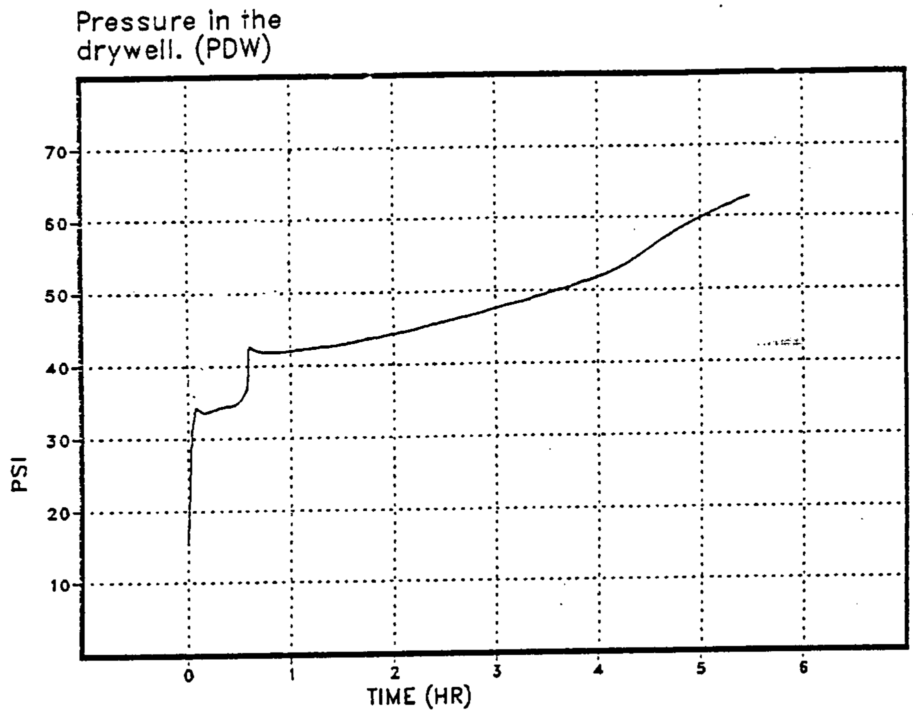


Figure 7.2-88

MNT MED LOCA W/ NO INJECTION
NO ADS



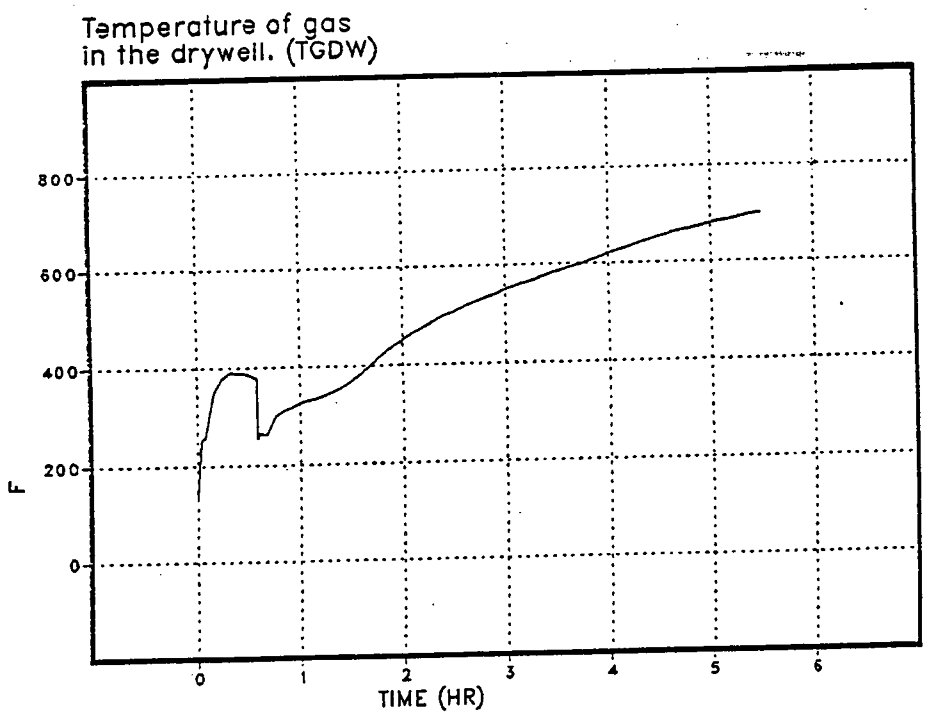
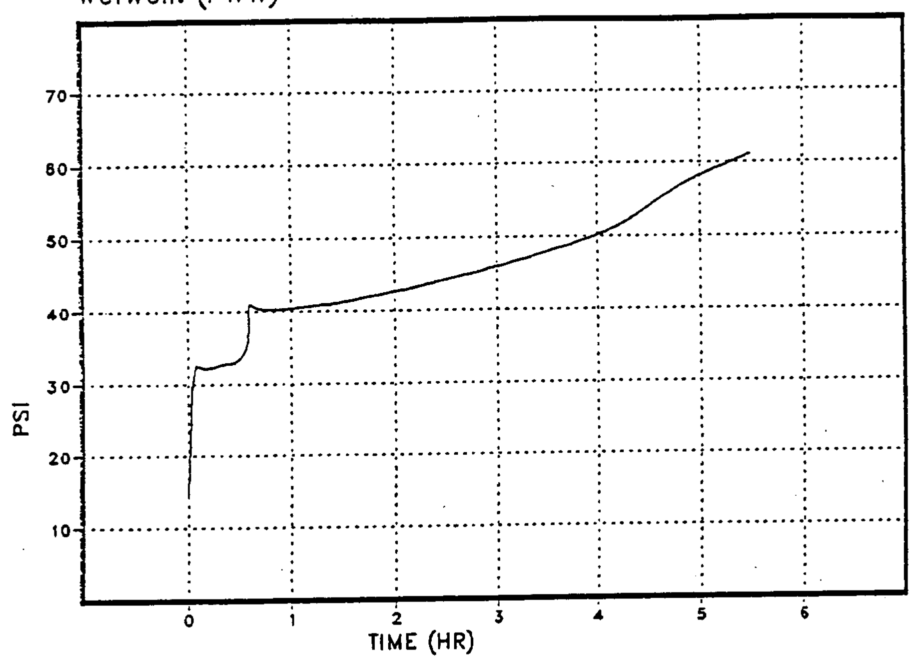


Figure 7.2-89

MNT MED LOCA W/ NO INJECTION
NO ADS

Pressure in the wetwell. (PWW)



Torus water temperature. (TWSP)

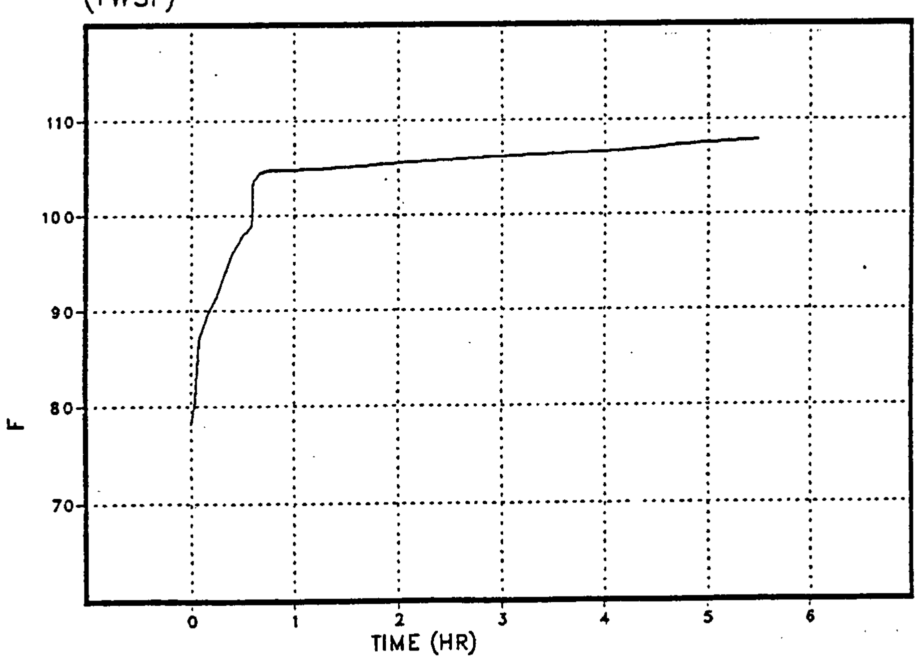
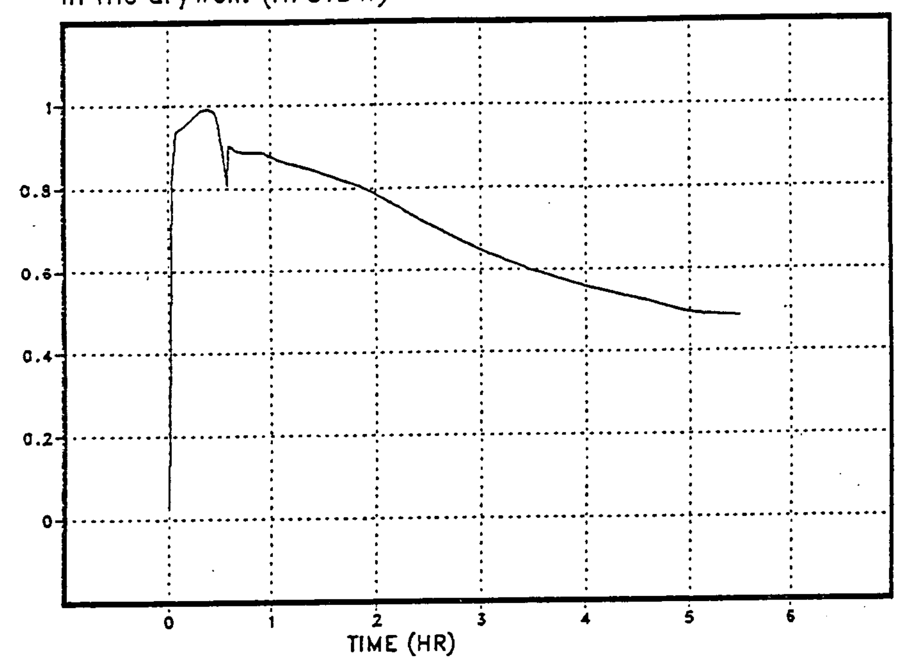


Figure 7.2-90

MNT MED LOCA W/ NO INJECTION
NO ADS

Mole fraction of steam
in the drywell. (NFSTDW)



Mole fraction of steam
in the wetwell. (NFSTWW)

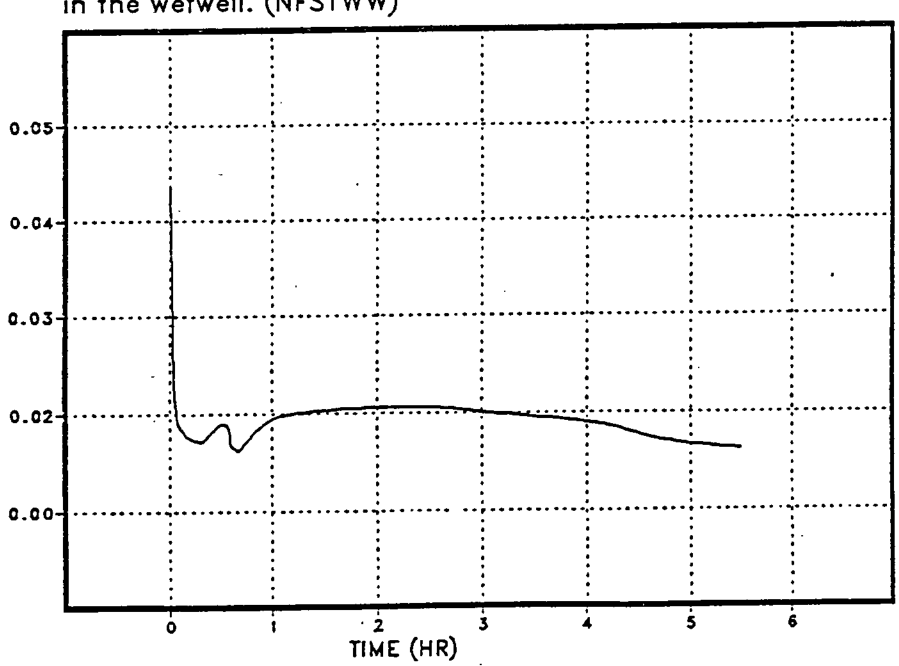
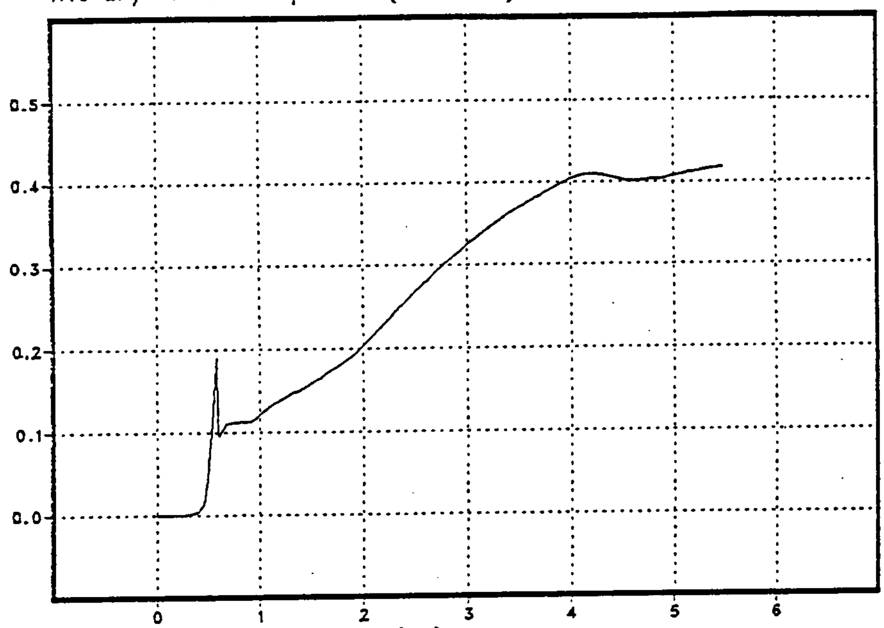


Figure 7.2-91

MNT MED LOCA W/ NO INJECTION
NO ADS

Mole fraction of hydrogen in
the drywell atmosphere. (NFH2DW)



Mole fraction of H2
in the wetwell. (NFH2WW)

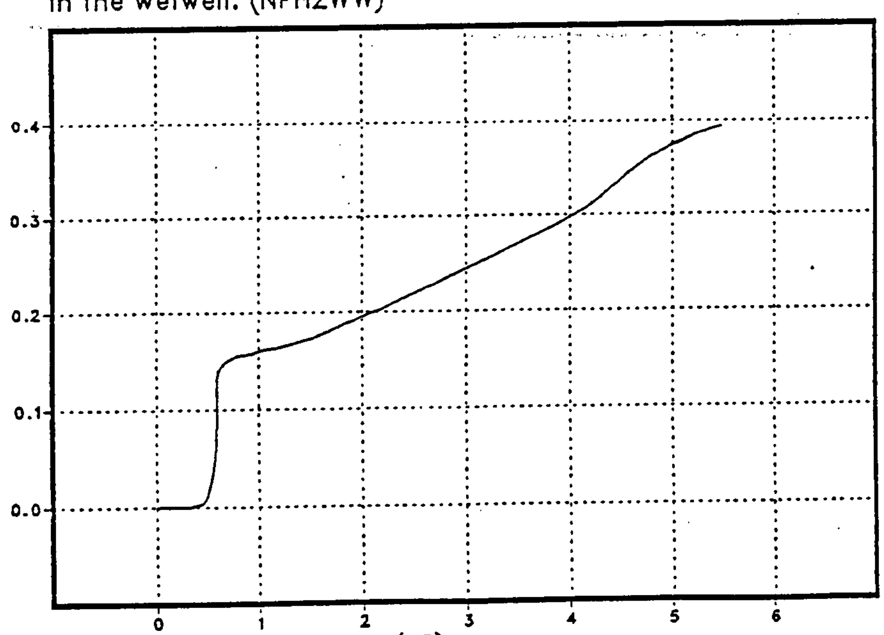
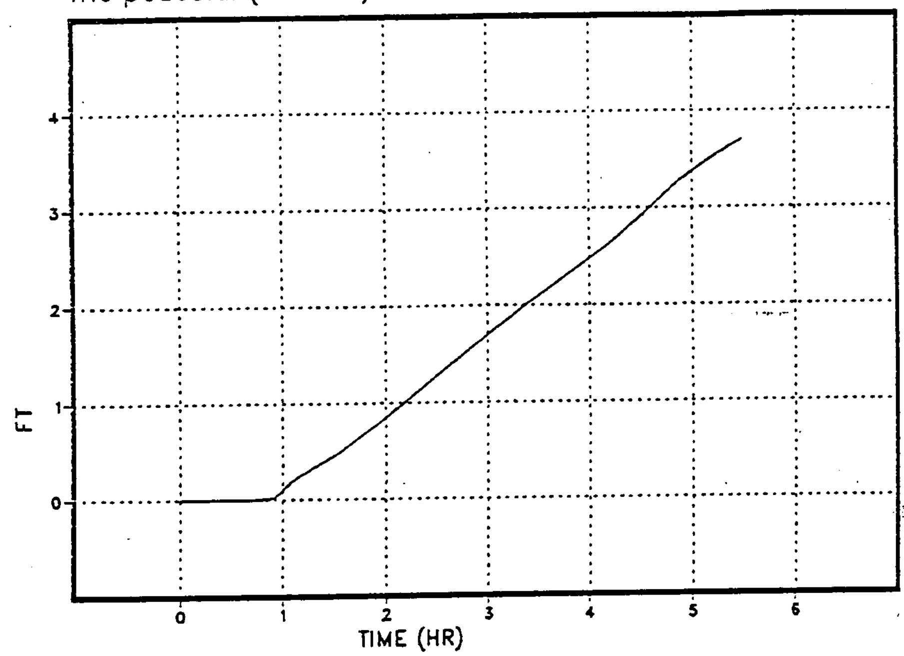


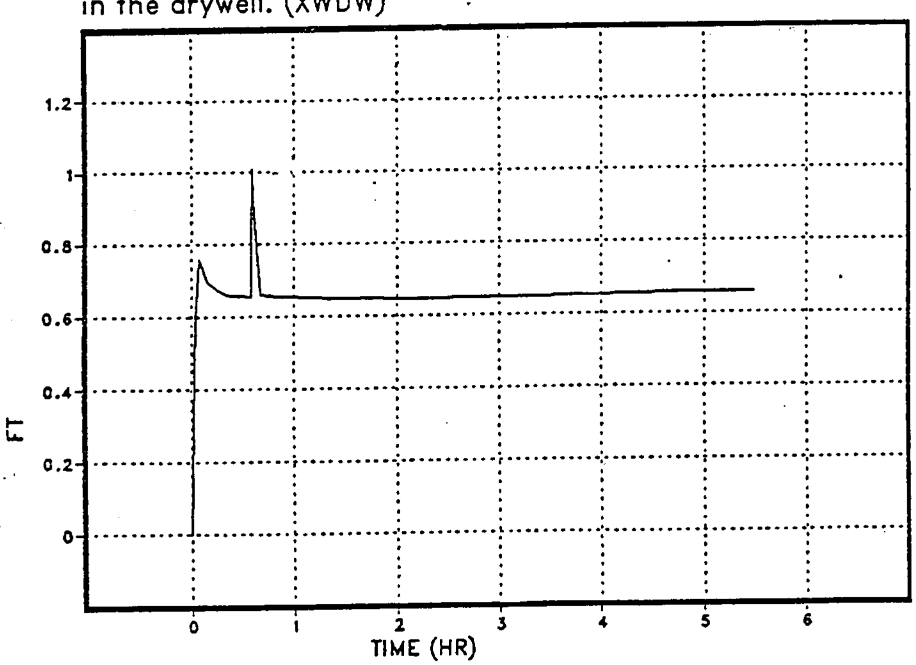
Figure 7.2-92

MNT MED LOCA W/ NO INJECTION
NO ADS

Concrete ablation thickness in the pedestal (XCNPDP)



Height of water in the drywell. (XWDW)



7.3 Reactor Building Equipment Survival After Containment Failure

A model of the reactor building was used in a variety of containment failure sequences to determine whether the conditions in the reactor building following containment failure are severe enough to disable the equipment located there. Of particular interest is the low pressure injection equipment for the RHR (LPCI) and core spray systems. With the exception of certain valves and piping, equipment for these systems is located in the RHR corner rooms on the lowest level of the reactor building, at the same elevation as the torus room.

At Monticello, the RHR corner rooms are well isolated from the rest of the reactor building, being separated from the torus room and the upper floors by reinforced concrete walls and ceilings. Because of this isolation, it was expected that conditions in these rooms might be mild enough following a containment failure that the equipment in those rooms would remain operable.

The reactor building model was developed based on plant drawings and a walkdown of the building. The model consists of nine nodes representing various areas of the building. Passageways which are normally open between these areas, such as open stairways and equipment hatches, are modeled as open flow paths of appropriate size between the nodes. Passageways which are normally closed but which may open if a sufficient differential pressure builds up, such as doorways and rupture panels, are modeled as failure junctions between adjoining nodes or between a node and the outside environment. For each failure node, the area of the flow path, the differential pressure required to open it, and the direction in which that pressure must be applied are specified; for junctions such as doors which can be forced open in either direction, but require different amounts of force depending on the direction, this is also specified. The nodes, open junctions, and failure junctions used in the model are shown in figure 7.3-1. The torus room and the four corner rooms on that level are each represented by a node, since these rooms are well isolated; the upper levels are divided into four large nodes, since they are connected by large open stairways and hatches. The outside walls on the refuel floor are expected to fail at a lower differential pressure than those in other areas, so this is where the failure junction to the environment is located.

Reactor building conditions were examined following releases from containment caused by (1) controlled venting of the containment, (2) containment overpressurization, (3) an unisolated LOCA outside of containment, and (4) an anticipated transient without scram (ATWS). Because the RHR and core spray pumps are able to operate in 100% humidity and at pressures far exceeding any that the reactor building can sustain, it is the temperature in the RHR corner rooms which determines whether the equipment can survive. The behavior of air temperature in the RHR corner rooms during these four types of event is summarized below.

Controlled Venting of the Containment:

The event modeled is a LOCA in one of the recirculation pump suction lines, resulting in core damage. When the vessel fails at one hour, LPCI is recovered and provides flow through the vessel breach to cover the debris, but containment cooling using the RHR system is not placed in service. When the drywell pressure reaches 56 psig, the operator begins using the drywell vent to control pressure between 46 psig and 56 psig. The temperature in the RHR rooms is unaffected during this sequence, and never rises above 90°F. [Figure 7.3-2.]

Containment Overpressurization:

Both a station blackout case and a loss of decay heat removal case (TW) were examined. In the station blackout, HPCI and RCIC alone remain operable until they fail due to battery depletion after four hours. The core melts through the vessel bottom at nine hours; the containment continues to heat up until it fails on overpressure (103 psig) around 24 hours. The gas temperature in the RHR rooms is unaffected until the containment failure, which is followed immediately by hydrogen burning in the reactor building; during this period the RHR room temperatures spike to a peak of 190°F, then drop for the remainder of the sequence, leveling out near 140°F. [Figure 7.3-3.] In the source term evaluation, no credit was taken for survival of equipment in the reactor building after the hydrogen burn.

The TW case begins with closure of the MSIVs, with feedwater providing injection throughout this accident. Steam from the core is channeled through the SRVs and into the suppression pool. When the pool reaches 142°F, the operator gradually depressurizes the reactor vessel to stay within the pool's heat capacity temperature limit curve. When the drywell reaches 70 psig, the operator loses pneumatic control of the SRVs, and the reactor vessel repressurizes. The containment pressure continues to rise until the drywell fails on overpressure at 66 hours. The gas temperature in the RHR rooms is unaffected until the containment fails; it then rises to 147°F for a period of approximately one hour, and falls thereafter. [Figure 7.3-4.] For loss of decay heat sequences, equipment in the RHR corner rooms was assumed to remain operational after the containment failed.

Unisolated LOCA Outside Containment:

The event modeled is a large break inside the steam chase in an unisolated main steamline. No injection is available to the reactor vessel. The reactor building fails immediately in the refueling area at the top of the building. In the RHR rooms, the temperature reaches a peak of 107°F for 15 minutes, then drops thereafter except for a 1°F increase when the vessel fails at 1 hour. The magnitude and location of this break make this case more severe than similar breaks in smaller lines. [Figure 7.3-5.]

ATWS:

The event analyzed is a turbine trip ATWS in which injection is lost after 26 minutes; at vessel failure, one LPCI pump begins to operate, providing cooling for the core debris. The drywell failure at 15 minutes produces a small temperature rise in the RHR rooms; the temperature then gradually rises and levels out at 125°F. At 1.5 hours, a hydrogen burn in the reactor building causes a spike of 190°F. At 1.7 hours, the vessel failure causes a spike of 250°F; within four minutes, the temperature drops just below 200°F and remains there for fifteen minutes. Temperature then drops and levels out at 140°F for the remainder of the sequence. [Figure 7.3-6.]

The highest sustained temperature during this sequence is 200°F for fifteen minutes. For ATWS cases, no credit was taken for survival of equipment in the reactor building after the containment failed.

In summary, the maximum sustained air temperatures in the RHR rooms during these containment failure sequences are:

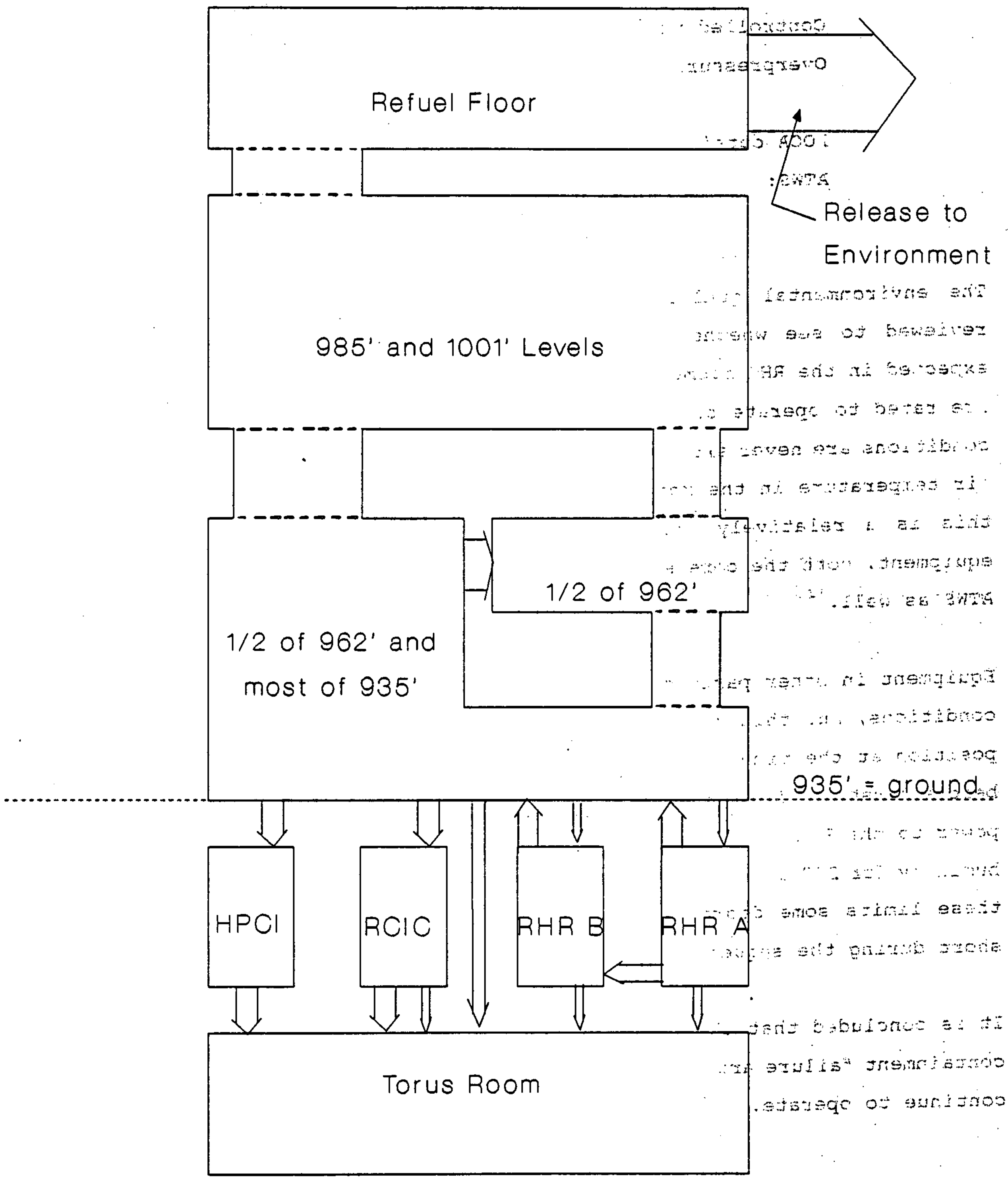
Controlled venting:		90°F	
Overpressurization:	TW:	147°F	
	SBO:	170°F	(estimated from spike)
LOCA outside containment:		107°F	
ATWS:		200°F	

The environmental qualifications of the core spray and RHR equipment were reviewed to see whether this equipment is likely to survive the conditions expected in the RHR rooms during these sequences. The RHR and core spray pumps are rated to operate continuously at 100% humidity, 66 psig, and 212°F. These conditions are never exceeded during any but the ATWS sequence; in that case, the air temperature in the room exceeds 212°F for less than five minutes. Because this is a relatively brief time in which to raise the temperature of the equipment, both the core spray and RHR pumps may continue to operate during the ATWS as well.

Equipment in other parts of the reactor building will not experience such mild conditions, but this consists mainly of valves which are either in the correct position at the time of containment failure or operate immediately afterwards, before harsh conditions have degraded the equipment. The cables supplying motive power to the RHR and core spray pumps are typically qualified to 340°F and 100% humidity for 200 days; it is assumed that if they do experience conditions beyond these limits some degradation of the insulation may occur, but they will not short during the sequences examined here.

It is concluded that the environmental conditions in the RHR rooms following a containment failure are sufficiently mild that the core spray and RHR systems can continue to operate.

Figure 7.3-1: MAAP Reactor Building Model.

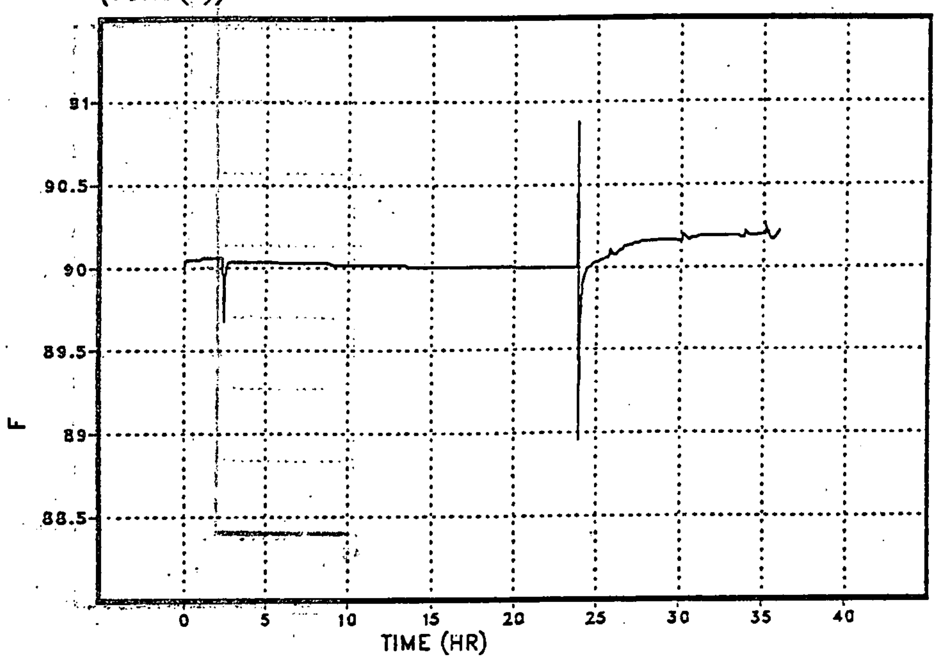


⟹ **denotes potential failure junction.**

Figure 7.3-2

MNT 3-06, RELEASE A6, L VB L
LBLOCA, NO INJ, FLOOD, VENT @56 PSIG

Gas temperature in building node 2.
(TGRB(2))



Pressure in building node 2.
(PRB(2))

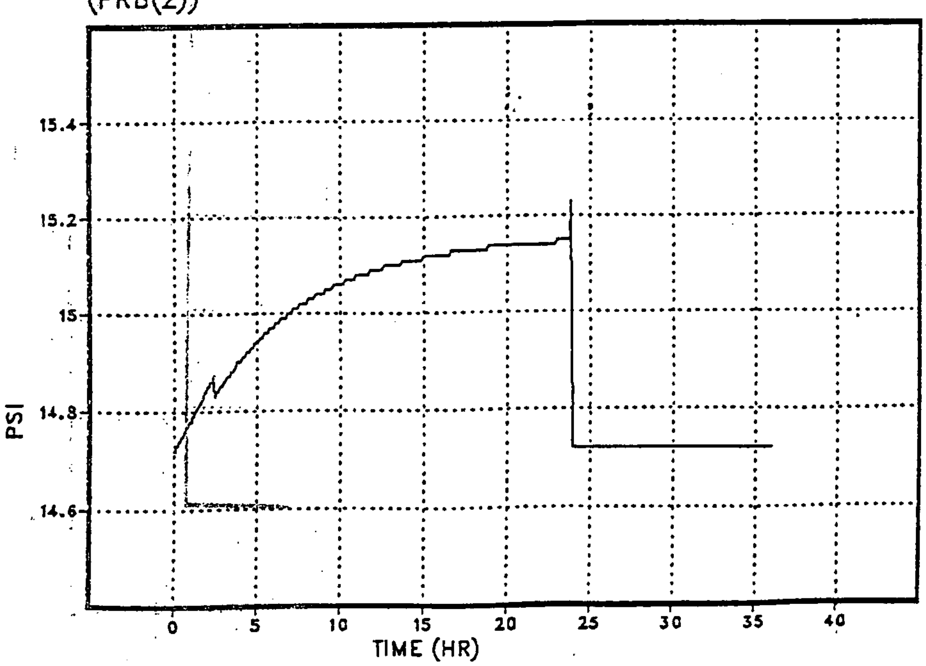
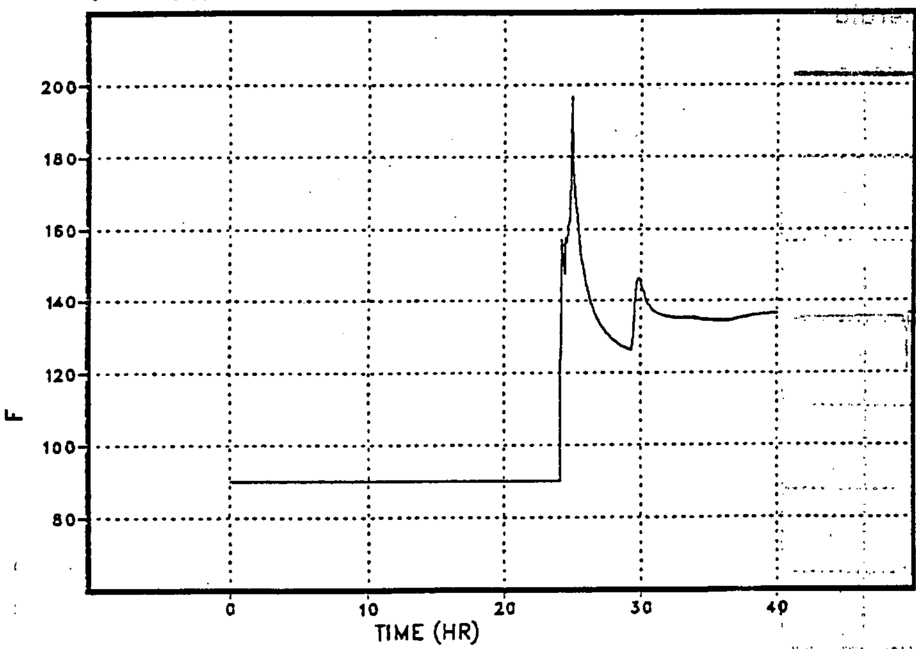


Figure 7.3-3

MNT CASE 1B-29, RELEASE C12
DAMAGE STATE H LM I, LONG TERM SBO

Gas temperature in building node 2.
(TGRB(2))



Pressure in building node 2.
(PRE(2))

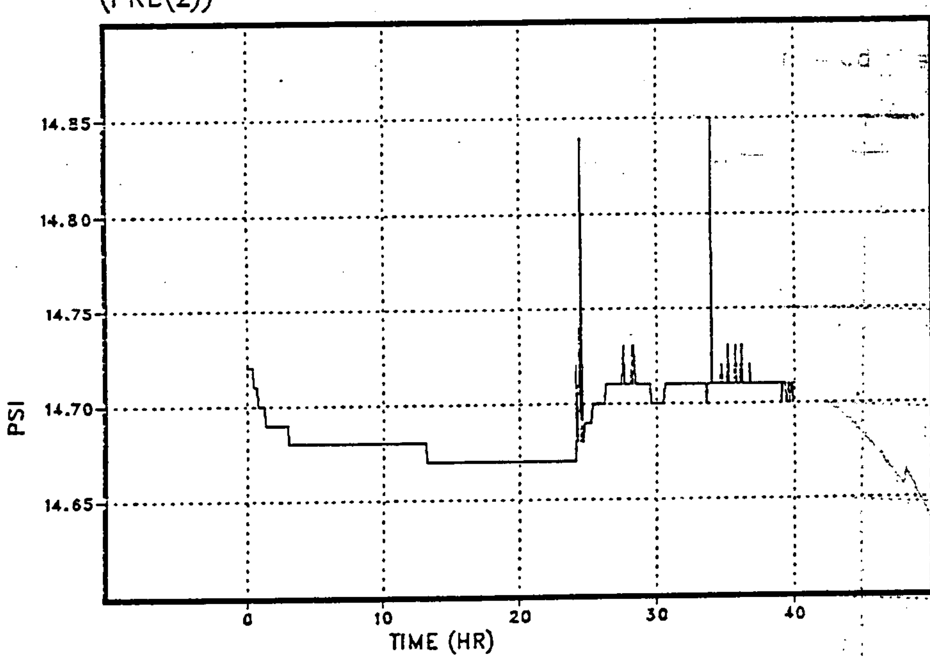
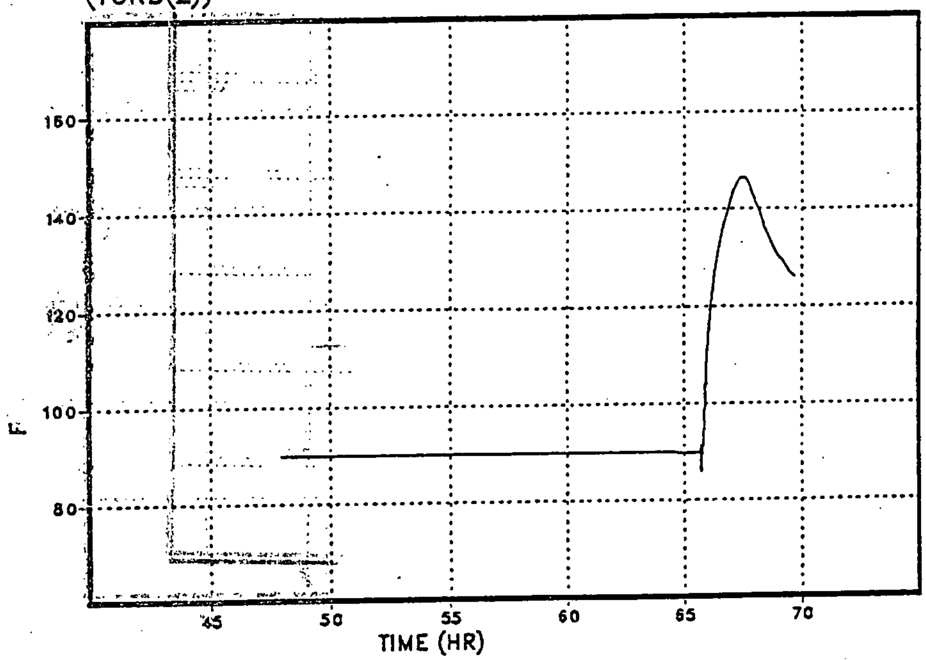


Figure 7.3-4

TW MSIV CLOSE; INJECTION FROM
CST'S ONLY; HCTL %% RESTART 1 %%

Gas temperature in building node 2.
(TGRB(2))



Pressure in building node 2.
(PRB(2))

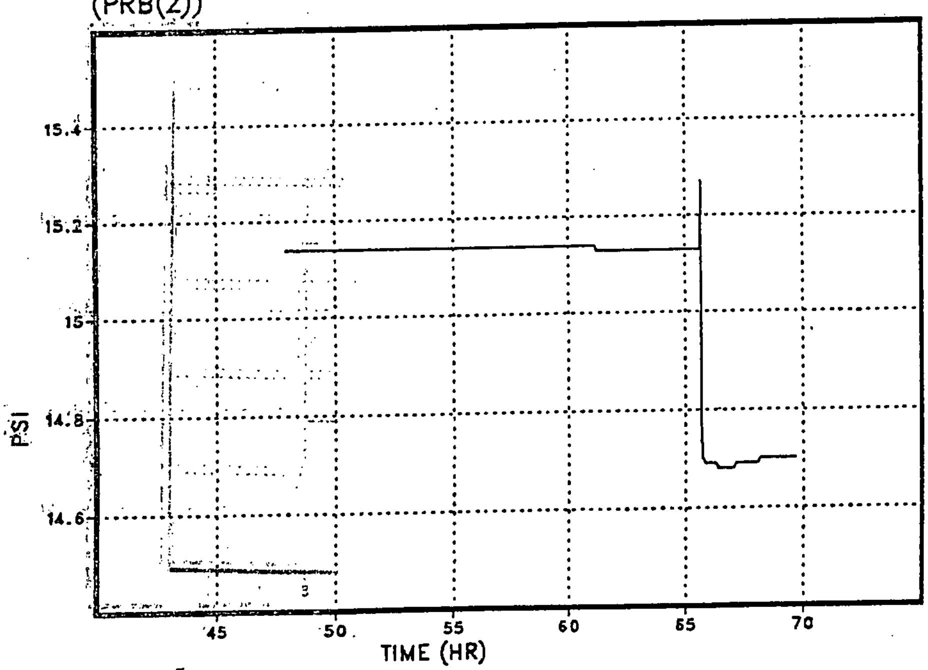
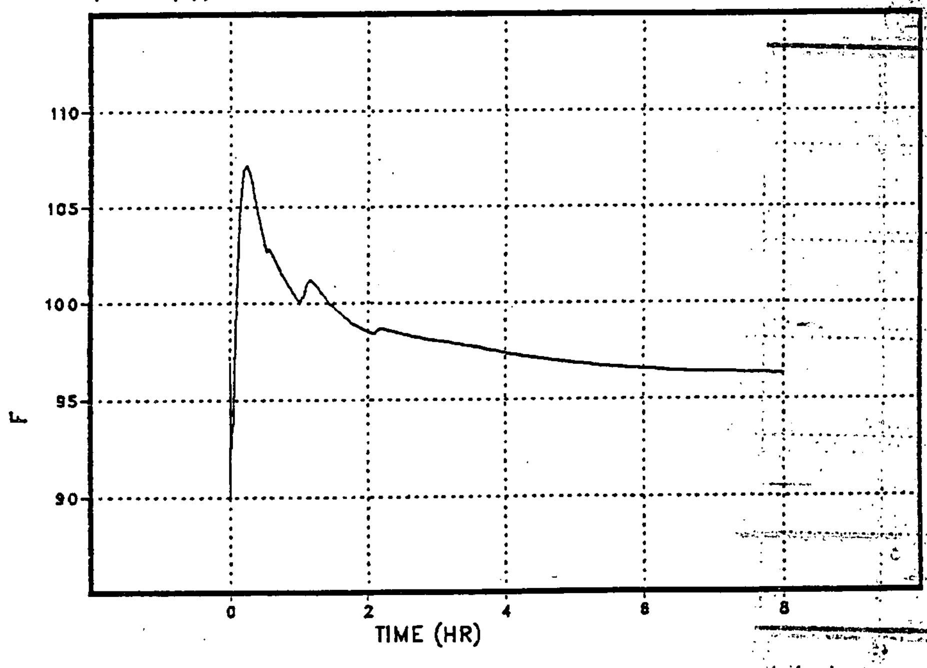


Figure 7.3-5

. MNT AOUT MODE E DAMAGE L CI E
AOUT IN STM CHASE, LO PR INJ FAILS

Gas temperature in building node 2.
(TGRB(2))



Pressure in building node 2.
(PRE(2))

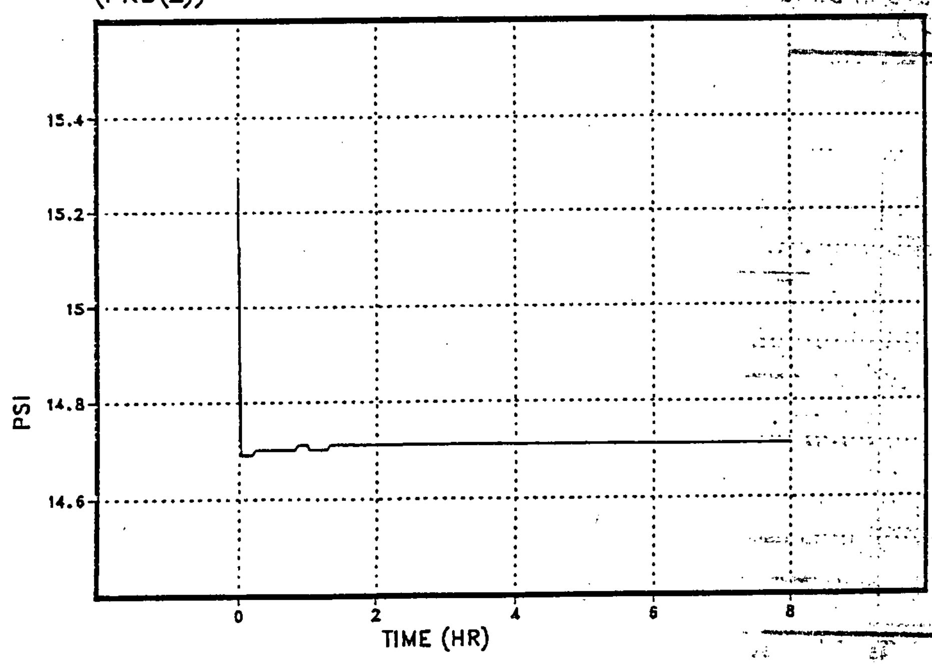
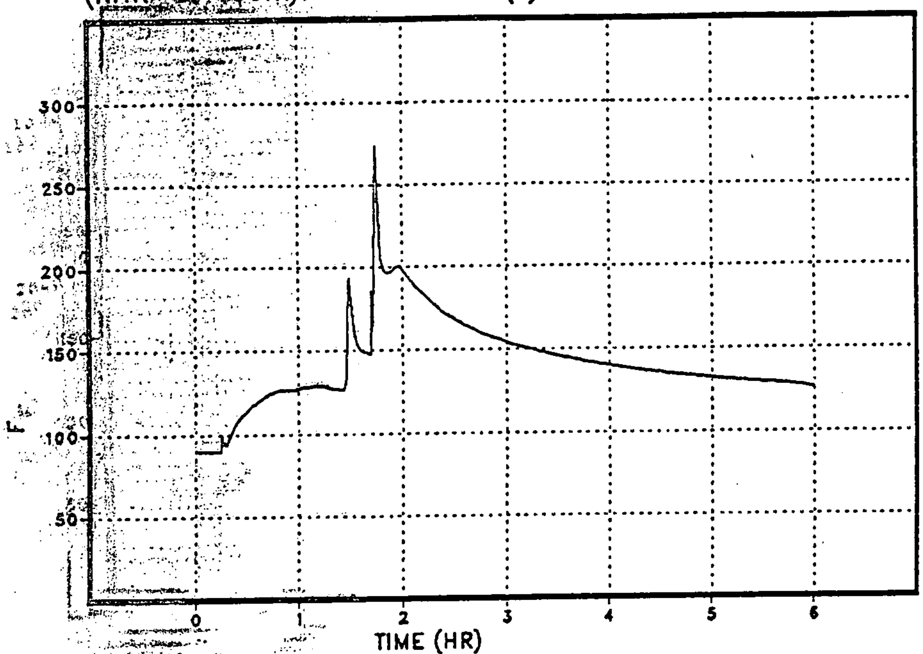


Figure 7.3-6

MNT TTW/BP ATWS; FW FOR .43HR
1 LPCI PUMP (CRD) @ RV FAIL

Gas temperature in building node 2
(RHR "B" room). TGRB(2)



Pressure in building node 2
(RHR "B" room). PRB(2)